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Generic Design Assessment – New Civil Reactor Build

Step 4 Structural Integrity Assessment of the Westinghouse AP1000® Reactor

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PREFACE

The Office for Nuclear Regulation (ONR) was created on 1st April 2011 as an Agency of the Health and Safety Executive (HSE). It was formed from HSE's Nuclear Directorate (ND) and has the same role. Any references in this document to the Nuclear Directorate (ND) or the Nuclear Installations Inspectorate (NII) should be taken as references to ONR.

The assessments supporting this report, undertaken as part of our Generic Design Assessment (GDA) process, and the submissions made by Westinghouse relating to the AP1000® reactor design, were established prior to the events at Fukushima, Japan. Therefore, this report makes no reference to Fukushima in any of its findings or conclusions. However, ONR has raised a GDA Issue which requires Westinghouse to demonstrate how they will be taking account of the lessons learnt from the events at Fukushima, including those lessons and recommendations that are identified in the ONR Chief Inspector's interim and final reports. The details of this GDA Issue can be found on the Joint Regulators' new build website www.hse.gov.uk/newreactors and in ONR's Step 4 Cross-cutting Topics Assessment of the AP1000® Reactor.

EXECUTIVE SUMMARY

This report presents the findings of the Structural Integrity assessment of the AP1000 reactor undertaken as part of Step 4 of the Health and Safety Executive's (HSE) Generic Design Assessment (GDA). The assessment has been carried out on the Pre-construction Safety Report (PCSR) and supporting documentation submitted by Westinghouse during Step 4.

The Step 4 assessment built on the assessments already carried out for Steps 2 and 3 and reviewed the safety aspects of the AP1000 reactor in greater detail, by examining the evidence, supporting arguments and claims made in the safety documentation. This has enabled me to make judgements on the adequacy of the Structural Integrity information contained within the PCSR and supporting documentation.

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 targeted and structured manner with a view to revealing any topic-specific or generic weaknesses in the safety case. To identify the sampling for the Structural Integrity an assessment plan for Step 4 was set out in advance. A number of items have been agreed with Westinghouse as being outside the scope of the GDA process and hence have not been included in my assessment.

My assessment has focussed on the nuclear safety related metal pressure vessels and piping and other pressure boundary components including:

- Categorisation and classification of systems, structures and components.
- Materials selection, design, fabrication.
- In-manufacture examination and testing.
- The analysis of structural integrity under normal load and faulted conditions (including fracture mechanics based analyses).
- Lifetime ageing of materials (including neutron irradiation embrittlement).

Nuclear pressure vessels and piping are designed to internationally accepted design codes and Westinghouse has designed the AP1000 against the American nuclear design code, ASME III. ND is familiar with its requirements and judges these to be generally acceptable for nuclear pressure systems.

However, there are a few critical components for which it is necessary to show that the likelihood of gross failure is so low that it can be discounted. In the UK we do not accept that the normal code requirements are sufficient to provide this level of confidence and we expect a higher level of demonstration of structural integrity. Westinghouse has accepted the need to make this demonstration in line with UK practice.

Westinghouse has designated these components as either Highest Safety Significance (HSS), for which there is no protection and failure is intolerable, or High Integrity (HI), for which failure would lead to severe core damage but effective containment will limit the off site consequences. Given their significance and the need for a demonstration against UK practice, I have concentrated on the demonstration of integrity for these components and I have satisfied myself that the process for identifying them is adequate.

The evidence to show that the likelihood of failure is so low that it can be discounted includes an avoidance of fracture demonstration which integrates fracture mechanics analyses, material toughness and qualification of manufacturing inspections. Westinghouse accepted the requirement to determine a limiting defect size and to demonstrate that the proposed inspection techniques were capable of detecting these with some margin. The R6 fracture mechanics

methodology was used to determine the limiting defect size and the ENIQ approach to confirm that defects of this size could be reliably detected with some margin.

Westinghouse has undertaken this work on a limited set of welds on the HSS and HI components which are believed to be representative of the more challenging areas in these components, and I am satisfied that this limited set is adequate for the purposes of making a demonstration within GDA.

Calculating limiting defect sizes using the R6 fracture mechanics methodology is consistent with existing UK nuclear industry practice, and I used an experienced UK contractor to review a small number of the Westinghouse calculations.

For the manufacturing inspection proposals Westinghouse prepared substantial, but nevertheless reduced scope, Technical Justifications (termed Inspection Plans) and subjected these to review by an independent and experienced, UK based, quasi Qualification Body. This body has confirmed that they believed that with a full technical justification it would be possible to qualify the proposed inspection, and I am satisfied with this process.

Westinghouse has submitted all the planned reports on avoidance of fracture, however a number of the important reports arrived later than had been originally planned and I have been unable to undertake a full assessment within the timescales allowed for GDA Step 4. Based on a high level review, I have sufficient confidence in the approach to conclude that it should be possible to provide a suitable demonstration for the safety case and thereby to support an Interim Design Acceptance Confirmation (IDAC). However a more detailed assessment post GDA Step 4 will be required to confirm that an adequate justification has been made before I am confident to support a DAC. A GDA Issue has been raised to support this ongoing assessment work post Step 4.

The design and construction of the remaining important vessels and components will be based on the normal requirements of the American nuclear design code ASME III which I judge to be generally acceptable.

I asked a UK contractor to review code compliance against ASME III on a number of the main vessels, and this identified a number of apparent discrepancies in the reports reviewed. Westinghouse is confident the vessels are code compliant and have provided additional supporting evidence to show this. This additional evidence gives confidence that the vessels are code compliant, and can therefore support an IDAC, however full resolution of the discrepancies will be needed before I can support a DAC. A GDA Issue has been raised to support this ongoing assessment work post Step 4.

Westinghouse did not decide on the material to be used for the reactor coolant pump casing until virtually the end of the assessment phase of GDA. As a consequence they were unable to supply technical reports on the casing design in sufficient time to allow me to undertake a full assessment of the pump casing. At this stage I have sufficient knowledge of the proposed design to be satisfied that the design and material choice are likely to be adequate and can therefore support an IDAC. However a more detailed assessment post GDA Step 4 will be required to confirm that an adequate justification has been made before I am confident to support a DAC. A GDA Issue has been raised to support this ongoing assessment work post Step 4.

Westinghouse has not yet demonstrated a 60 year fatigue life for the pressuriser surge line against the ASME code. This work is ongoing and Westinghouse is confident that they will be able to demonstrate code compliance. I am prepared to support an IDAC on the basis of this assurance; however, it will be necessary for Westinghouse to complete their assessments against the ASME code to confirm code compliance before I would be in a position to support a DAC. A GDA Issue has been raised to address this work post Step 4.

In terms of the containment vessel I am generally content with the analysis work that has been undertaken, but there is ongoing work to demonstrate the integrity against a thermal load case. In addition the containment vessel welds are not post-weld heat treated and Westinghouse has not, so far, adequately justified that the vessel is sufficiently tolerant of welding defects given the high residual stress which remains without post-weld heat treatment. Both these matters will need to be resolved before I can support a DAC, but I have sufficient confidence that the matters can be resolved to support an IDAC. A GDA Issue has been raised to address this work post Step 4.

The overall categorisation and classification scheme developed for the UK AP1000 is considered meet our expectations in other topic areas, but late in the assessment process I identified three areas where the classification scheme as applied to pressure equipment and tanks needed further justification. This was recognised too late in the assessment process for Westinghouse to provide the necessary supporting evidence and I have therefore taken the matter forward through a GDA Issue on Structural integrity Classification.

The GDA Issues discussed above are of particular significance and will require resolution before HSE would agree to the commencement of nuclear safety related construction of an AP1000 reactor in the UK. The GDA Issues are listed in Annex 2.

I have also identified several areas of a Licensee or site specific nature that do not need to be addressed as part of the GDA process but which will need to be followed up by any Licensee and these are listed in Annex 1 as Assessment Findings.

Some examples of my Assessment Findings are:-

- The Licensee shall define and justify the chemical compositions of the main forgings regardless of whether the composition is based on ASME III compositions or on more restrictive limits. The justification shall take into account start-of-life materials properties and through-life changes.
- The Licensee shall demonstrate that the damage correlation he will to use to determine the shift in RTNDT is suitable for the RPV materials. This needs to reflect on the current understanding of damage correlations and should be kept under review over the life of the plant as new data becomes available from surveillance specimens and from worldwide data.
- The Licensee shall review the upper shelf fracture toughness to be assumed in the fracture assessments of the low alloy steel forgings and their weldments to ensure that they have confidence that values can be reliably achieved during the manufacture of these components.
- The licensee shall demonstrate the protective coating applied to the containment vessel is capable of protecting it against extended exposure to the potentially corrosive chemicals to which it may be exposed.

Overall, based on the sample undertaken in accordance with ND procedures, 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 AP1000 reactor design. The AP1000 reactor is therefore suitable for construction in the UK, subject to satisfactory progression and resolution of GDA Issues to be addressed during the forward programme for this reactor and assessment of additional information that becomes available as the GDA Design Reference is supplemented with additional details on a site-by-site basis.

LIST OF ABBREVIATIONS

AGR	Advanced Gas-cooled Reactor
ALARP	As Low As Reasonably Practicable
ANL	Argonne National Laboratory
AVB	Anti Vibration Bar
BEZ	Break Exclusion Zone
BMS	(Nuclear Directorate) Business Management System
BPVC	Boiler and Pressure Vessel Code
CEGB	Central Electricity Generating Board
CDRM	Control Rod Drive Mechanisms
CSR	Component Safety Report
CRDM	Control Rod Drive Mechanism
CV	Containment Vessel
DAC	Design Acceptance Confirmation
DBA	Design Basis Accidents
DNBR	Departure from Nucleate Boiling Ratio
DSM	Defect Size Margin
DVI	Direct Vessel Injection
E-DCD	European Design Control Document
ELLDS	End of Life Limiting Defect Size
ENIQ	European Network for Inspection and Qualification
EoL	End of Life
EPRI	Electric Power Research Institute
GDA	Generic Design Assessment
HI	High Integrity (component)
HSE	The Health and Safety Executive
HSS	Highest Safety Significance (component)
IAEA	The International Atomic Energy Agency
ID	Internal Diameter
IDAC	Interim Design Acceptance Confirmation
IHP	Integrated Head Package
INES	International Radiological and Nuclear Event Scale
IOF	Incredibility of Failure

LIST OF ABBREVIATIONS

IP	Inspection Plan (Westinghouse name for an outline Technical Justification)
ISI	In-service Inspection
IVC	Inspection Validation Centre
LBB	Leak Before Break
LFCG	Lifetime Fatigue Crack Growth
LOCA	Loss of Coolant Accident
LTOPS	Low Temperature Operation Protection System
LWR	Light Water Reactor
MCL	Main Coolant Line
MSL	Main Steam Line
MDEP	Multinational Design Evaluation Programme
MSIV	Main Steam Isolation Valve
MTS	Main Turbine System
NACE	National Association of Corrosion Engineers
ND	The (HSE) Nuclear Directorate
NDT	Non-Destructive Testing
NNL	National Nuclear Laboratory
NSL	Nuclear Site Licensing
OD	Outer Diameter
OECD	Organisation for Economic Co-operation and Development
ORNL	Oak Ridge National Laboratory
PCSR	Pre-construction Safety Report
PCCWST	Passive Containment Cooling Water Storage Tank
PRHR	Passive Residual Heat Removal
PRZ	Pressuriser
PWHT	Post Weld Heat Treatment
PWR	Pressurised Water Reactor
P-T (limits)	Pressure-Temperature Limits
PXS	Passive Core Cooling System
QB	Qualification Body
QEDS	Qualified Examination Defect Size
RCL	Reactor Coolant Loop
RCP	Reactor Coolant Pump

LIST OF ABBREVIATIONS

RCS	Reactor Coolant System
RI	Regulatory Issue
RIA	Regulatory Issue Action
RO	Regulatory Observation
ROA	Regulatory Observation Action
RP	Requesting Party
RPV	Reactor Pressure Vessel
SAP	Safety Assessment Principle
SFR	Safety Function Requirements
SG	Steam Generator
SGS	Steam Generator System
SSC	System, Structure and Component
TAG	(Nuclear Directorate) Technical Assessment Guide
TAGSI	UK Technical Advisory Group on Structural Integrity
TIG	Tungsten Inert Gas
TJ	Technical Justification
TOFD	Time of Flight Diffraction
TQ	Technical Query
TSC	Technical Support Contractor
US NRC	Nuclear Regulatory Commission (United States of America)
WEC	Westinghouse Electric Company LLC
WENRA	The Western European Nuclear Regulators’ Association
UT	Ultrasonic Testing

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1 INTRODUCTION

1 This report presents the findings of the Step 4 Structural Integrity assessment of the Revision 2 AP1000 reactor PCSR (Ref. 12) and supporting documentation provided by Westinghouse under the Health and Safety Executive's (HSE) Generic Design Assessment (GDA) process. Assessment was undertaken of the PCSR and the supporting evidence derived from the Master Submission List (Ref. 14). The approach taken was to assess the principal submission, i.e. the PCSR, and then undertake assessment of the relevant documentation sourced from the Master Submission List on a sampling basis in accordance with the requirements of ND Business Management System (BMS) procedure AST/001 (Ref. 2). The Safety Assessment Principles (SAPS) (Ref. 4) have been used as the basis for this assessment. Ultimately, the goal of assessment is to reach an independent and informed judgment on the adequacy of a nuclear safety case. After the end of the GDA assessment period Revision A of the AP1000 PCSR (Ref. 13) was provided. This incorporated the Component Safety Reports (Refs 17 to 27) which had been available during GDA and were subject to assessment.

2 During the assessment a number of Regulatory Observations (RO) and Technical Queries (TQ) were issued and the responses made by Westinghouse assessed. Where relevant, detailed design information from specific projects for this reactor type has been assessed to build confidence and assist in forming a view as to whether the design intent proposed within the GDA process can be realised.

3 A number of items have been agreed with Westinghouse as being outside the scope of the GDA process and hence have not been included in this assessment.

2 NUCLEAR DIRECTORATE'S ASSESSMENT STRATEGY FOR STRUCTURAL INTEGRITY

2.1 Assessment Plan

4 The intended assessment strategy for Step 4 for the structural integrity topic area was set out in an assessment plan (Ref. 1) that identified the intended scope of the assessment and the standards and guidance that would be applied. This is summarised below.

5 The objective of the Step 4 assessment is to make a judgement on the adequacy of the claims, arguments and evidence in the area of structural integrity contained within the PCSR and Supporting Documentation. Assessment in Step 4 builds on the assessment carried out in Steps 2 and 3 and is oriented toward the evidence end of the spectrum of claims, arguments and evidence.

6 The overall bases for the start of assessment in GDA Step 4 are:

- PCSR (Ref. 12), the European Design Control Document (E-DCD), Ref. 15, and the Component Safety Reports (CSRs) for the structural Integrity components (Refs 17 to 27).
- Matters identified in GDA Step 3 that required further consideration and resolution within Step 4.

7 Within the Step 4 Plan the following generic HSE Commitments were required to be taken into consideration as part of the Step 4 structural integrity assessment:

- Consideration of issues identified in Step 3.
- Judging the design against SAPs and judging whether the proposed design reduces risks and is ALARP.
- Inspections of the Requesting Party's procedures and records.
- Independent verification analyses.
- Reviewing details of the design controls, procurement and quality control arrangements to secure compliance with the design intent.
- Assessing arrangements for moving the safety case to an operating regime.
- Assessing arrangements for ensuring and assuring that safety claims and assumptions are realised in the final design, building and construction.
- Reviewing overseas progress and issues raised by overseas regulators.
- Considering unresolved issues raised through the public involvement process.
- Resolution of identified nuclear safety issues, or identifying paths for resolution.

8 My GDA Step 4 assessment plan was based on my assessment of the 2009 PCSR, the European DCD and Westinghouse's responses to TQs and ROs contained in the Master Submission List and the design reference documentation. The 2009 PCSR was found to have significant shortfalls in terms of content and quality. Recognising the shortfalls with the 2009 PCSR, Westinghouse submitted Component Safety Reports (CSRs) for each of the main structural components in mid 2010. A replacement draft PCSR was supplied in December 2010. This was extensively restructured and enhanced the 2009 PCSR in order to address ND's concerns and incorporated the CSRs into the structural integrity

section. Westinghouse then submitted an approved PCSR in March 2011 which was too late for a meaningful assessment during GDA Step 4. However I was able to assess the CSRs and notwithstanding the GDA Issues raised within my assessment, I have no fundamental reasons to believe that Westinghouse cannot produce an adequate PCSR to support their GDA application, based on the information I have reviewed. I will however need to assess the revised PCSR, which Westinghouse must provide as part of a cross cutting GDA Issue.

2.2 Standards and Criteria

9 I have based my assessment of the structural integrity aspects of the AP1000 PCSR, the E-DCD and the CSRs primarily on the following:

- Safety Assessment Principles for Nuclear Facilities (Ref. 4); and
- Technical Assessment Guide - Integrity of Metal Components and Structures – T/AST/16 Issue 003 (Ref. 6).

10 For the SAPs the most relevant part is “Integrity of Metal Components and Structures” in Paras 238 to 279, involving Principles EMC.1 to EMC.34. Other key parts of the SAPs are “Ageing and Degradation” especially principles EAD.1 to EAD.4 and “Pressure Systems” especially EPS.4. A topic with some relevance to this assessment is “Safety Classification and Standards” in Paras 148 to 161, involving Principles ECS.1 to ECS.5. A list of these relevant SAPs is given in Table 1.

11 The assessment of the structural integrity area is on the basis of engineering practice and sound safety principles, rather than a numerical calculation of the likelihood of failure of components.

12 The AP1000 design is the outcome of many years of development and did not explicitly follow the approach to ALARP as practiced in the UK (e.g. SAPs Para. 93). As a consequence it is difficult to ‘back fit’ ALARP to the design at this stage although it is possible to examine individual important areas to determine if the situation is consistent with ALARP.

13 In carrying out my assessment, I have based my judgements of the technical aspects of structural integrity on the guidance provided on ALARP (e.g. Paras 14 and 93). I have interpreted the guidance to reach a judgement on the balance of all the factors which contribute to the structural integrity safety case.

14 Some components have a claim associated with them that gross failure is taken to be so unlikely it can be discounted. In assessing the arguments and evidence supporting this type of claim, I have applied the same basis of judgement as described above. For these claims of highest structural integrity, I have examined whether:

- the proposals meet a minimum level for such a claim;
- all that is reasonably practicable has been done.

2.3 Assessment Scope

15 Table 2 defines the scope of the Step 4 assessment, and this is based on Table 2 of the assessment plan (Ref. 1). Table 2 also lists any TSC reports commissioned.

2.3.1 Findings from GDA Step 3

16 The GDA Step 3 review of was carried out in 2009 and 2010 with the following specific aims:

- To improve knowledge of the design.
- Identify significant issues.
- Identify whether any significant design or safety case changes may be needed.
- Identify major issues that may affect design acceptance and attempt to resolve them.
- Achieve significant reduction in regulator uncertainty.

17 The results of the structural integrity Step 3 assessment are reported in Ref. 7. A total of thirteen Regulatory Observations (ROs) were pursued during Step 3 of which seven remained open at the end of Step 3 and therefore required work to achieve closure within Step 4. Two further ROs were raised at the end of Step 3.

2.3.2 Step 4 Structural Integrity Assessment

18 Table 2 shows that in general, the further work was carried out under new Actions to existing ROs, or new ROs that were established along with relevant Actions.

19 There was a substantial programme of work for Westinghouse (and implied assessment) for RO-AP1000-19 concerning avoidance of fracture of the highest integrity components which have been identified via RO-AP1000-18.

20 The response to RO-AP1000-19 was the process by which ND gained confidence that the integrity of the most important structural integrity components such as the reactor pressure vessel (RPV) could be demonstrated. It was recognised that the total scope and extent of this work required before reactor operation need not, and could not be completed within the timeframe of GDA Step 4. Therefore the aim of GDA Step 4 was for was to gain sufficient information on limiting defect sizes and inspection capability to enable all parties to be satisfied that the work to be carried out during the licensing phase would have a high likelihood of being able to achieve its purpose.

21 During Step 4 I also continued my assessment of the chemical composition of forgings and welds (RO-AP1000-21), irradiation embrittlement (RO-AP1000-22), fatigue usage for the pressuriser surge line (RO-AP1000-26), the basis for setting the RPV Pressure - Temperature limits (RO-AP1000-29) and the design of the Containment Vessel (RO-AP1000-30).

22 During Step 3 I identified some new areas for assessment :-

- Review of the documentary envelope (RO-AP1000-65) which included:
    - Equipment Specifications.
    - Design Specifications.
    - Analyses for loading conditions.
    - Welding procedures.
- Review of the accessibility for ISI (RO-AP1000-66).
- Operation of plant within safe limits (RO-AP1000-94).

- Review of the design of the Reactor Coolant Pump.

23 As my assessment progressed I also identified three new areas for assessment

- Review the need to take account of the requirements of NUREG 1.207 (effect of environment on fatigue crack growth).
- The acceptability of using an edition (for the most art) of a design code which is more than ten years old and has been superseded by later editions.
- The relevance of the recent Pressuriser heater leakage seen at Sizewell B to the AP1000.

Requirements for these new areas were discussed with Westinghouse using the established processes set out in the Interface Protocol.

2.3.3 Use of Technical Support Contractors

24 Table 5 of Ref. 1 identified 11 topics which were proposed for review by specialist contractors. All these have been performed as well as a few other topics which arose during the Step 4 assessment. The Contractors used were: EASL, Serco Assurance, Prof. Knott, TWI and NNL.

2.3.4 Cross-cutting Topics

25 There were a number of areas during the Step 4 assessment when there was a need to consult with other assessors. These are listed below:

- Categorisation and classification.
- Failure of pressure vessels, tanks and pipework (missiles and pipewhip).
- Transients used for fracture analyses.
- Pressure and temperature excursions within containment following large LOCA.
- Operational limits.
- Metrication.

2.3.5 Integration with other Assessment Topics

26 The need for coordination with other assessment areas was identified in Ref. 1 and the Table below summarises the coordination which occurred:

Structural Integrity Aspect / Assessment Area	Internal Hazards	Civil Engineering	Fault Studies / DBA	Reactor Chemistry	Mechanical Engineering	Phase 2 (site-licensing)
Common position on internal hazards for pressure boundary components and rotating components	✓					
Ensure common understanding on the potential for interaction between the containment vessel and the containment building.		✓	✓			
Consistent view on range of design basis loadings for pressure boundary components, including the containment vessel. Also advice to structural integrity that thermal-hydraulics analyses that provide pressure - temperature transients are based on acceptable methods			✓			
Ensure common understanding of reactor chemistry proposed by Westinghouse and any options that would be for the Licensee to decide.				✓		
Ensure common understanding of pressure relief arrangements for pressure boundary components (primary and secondary sides)					✓	
Need for coordination between ND assessors covering GDA and those involved with preparations for Phase 2 (site licensing)						✓

2.3.6 Out of Scope Items

27 In letter WEC000512N (Ref. 29) ONR identified a list of items proposed to be out of scope for GDA. The structural integrity items are tabulated below.

Out of Scope Item	Justification
Fracture Toughness testing for Avoidance of Fracture demonstration	Principles established, detailed specifications to be addressed during Phase 2.
Pre-Service Inspection and In-Service Inspection	To be addressed by the licensee organisation. GDA will consider accessibility issues.
Irradiation Damage Surveillance Programme	The detailed specifications will be addressed during Phase 2.

3 REQUESTING PARTY'S SAFETY CASE

28 Westinghouse provided Revision 2 of the PCSR in 2009 (Ref. 12) and this was revised and submitted in draft in December 2010 (Ref. 13) and as an approved document in March 2011 (Ref. 28), which was too late for meaningful assessment within GDA. However in the Structural Integrity area the revised PCSR is based closely on the UK Structural Integrity Classification Report (Ref. 16) and eleven Component Safety Reports (CSRs) (Refs 17 to 27) which were subject to review under GDA. These CSRs reference the European Design Control Document (E-DCD) (Ref. 15) for key design information and also a large number of supporting references.

29 At the highest level the approach is to allocate each structural component to a safety category dependent on the consequences of its gross failure and then assure its integrity by setting category dependent requirements which are more robust for the classes for which failure is least tolerable.

30 All the major structural component were either designed in the United States or under the direct control from there thus all the design calculations were carried out in imperial units and converted into metric units for documentation prepared for a European AP1000. Therefore documents prepared specifically of GDA are all in metric units but some documents which were requested during the assessment were only available in imperial units. In places in this report I quote directly from these reports and I have not converted measurements into metric units.

3.1 Structural Integrity Classification Methodology and Safety Case Claims

31 The Structural Integrity Classification scheme is an extended version of the AP1000 Safety Classification Scheme. Initially all components are assigned to 3 classes.

- Class 1: A structure, system or component that provides the principal means of fulfilling a Category A safety function. That is a safety function that is the principal means for maintaining nuclear safety whose failure has the potential for core damage or activity release to the environment within design basis limits.
- Class 2: A structure, system or component that provides the principal means of fulfilling a Category B safety function or is a significant contributor to fulfilling a Category A safety function. That is a safety function that makes a significant contribution to maintaining nuclear safety and where failure may reduce safety margins significantly, but will not result in Category A consequences.
- Class 3: A structure, system or component provided for any other safety purpose, including an ALARP measure. There are no special availability requirements.

32 However to align with accepted UK practice for determining the structural integrity requirement of passive structures, Class 1 structures were further divided into three groups to identify those components for which it would be necessary to demonstrate that the likelihood of gross failure was so low that it could be discounted.

33 For the UK design, the resulting structural integrity safety classes were identified as follows

Class 1 components:

- **Highest Safety Significance (HSS).** The gross failure of an HSS component is intolerable and there is no protection against the failure and it is not reasonably practicable to provide protection.
- **High Integrity (HI).** The failure of an HI component can lead to severe core damage but containment exists to limit the off-site consequences. In this case there would be a single line of protection but no redundancy.
- **Standard Class 1.** The failure of a Standard Class 1 component will result in only limited core damage.

Class 2 and Class 3 components:

- **Standard Class 2 and 3.** Failure of these components should not result in any core damage although there is the potential for contamination and worker dose. (These are unchanged from classes 2 and 3 in the AP1000 Safety Classification scheme.)

34 Consequences are considered in their broad sense and they "….*include the direct effects on plant safety, for example, a loss of coolant inventory or reactivity excursions, and the indirect effects from, for instance, missiles, blast, pipewhip, flooding and water/steam jets. This also needs to include any longer term effects on the availability of essential safety systems and instrumentation required to maintain plant safety in the faulted condition."* Leak before break arguments are not used alone to reduce the safety class of a component but can and do form part of the justification for the classification of a component.

35 The Class 1 equipment in the AP1000 Safety Classification scheme was reviewed and 18 components were identified at potentially HI or HSS. For each of these the consequences of failure were reviewed and a category determined and detailed in Ref. 16. The resulting classification is summarised in the table below.

Component	Category	Component Safety Report
Reactor Vessel	HSS	UKP-GW-GLR-005
Steam Generator Secondary Side	HSS	UKP-GW-GLR-010
Steam Generator Primary Side - Channel head	HSS	UKP-GW-GLR-010
Steam Generator Primary Side - Tube Sheet	HSS	UKP-GW-GLR-010
Pressuriser	HSS	UKP-GW-GLR-009
Reactor Coolant Pump Casing and Flywheel	Standard Class 1	UKP-GW-GLR-013
Welds between Reactor Coolant Loop (Hot and Cold legs) and RPV Safe Ends	HI	UKP-GW-GLR-014
Welds between Reactor Coolant Loop (Hot and Cold legs) and SG or RCP safe ends.	Standard Class 1	UKP-GW-GLR-014
Core Make-Up Tank	Standard Class 1	UKP-GW-GLR-012
Accumulator	Standard Class 1	UKP-GW-GLR-008
Passive Residual Heat Removal Heat Exchanger	Standard Class 1	UKP-GW-GLR-006

Component	Category	Component Safety Report
RPV Internals, Lower Core Support Structures, Core Barrel	Standard Class 1	UKP-GW-GLR-011
RPV Upper Internals	Standard Class 1	UKP-GW-GLR-011
Main Steam Lines Inside Containment	HSS	UKP-GW-GLR-015
Pipework in way of Containment Penetrations (Main Steam, Main Feedwater) – Outside Containment	Standard Class 1	UKP-GW-GLR-015
DVI Line	Standard Class 1	UKP-GW-GLR-005
Pressuriser Surge Line	Standard Class 1	UKP-GW-GLR-009
ADS System Piping and Safety Valve Piping	Standard Class 1	No CSR submitted
PRHR HX Inlet / Outlet Lines	Standard Class 1	UKP-GW-GLR-006

NB: Dissimilar metal welds between ferritic vessels (RPV or SG or PRZ) and austenitic safe ends are all HSS. Austenitic welds between safe ends and pipework are normally Class 1, but those between the RCL and the RPV safe ends are HI. The RCP is now declared to have a ferritic pump bowl but this late change in design is not addressed by these CSRs and has not been assessed. It has been raised in a GDA Issue (See Section 4.12 for discussion).

36 For the purposes of my assessment, I have regarded both the HSS welds and HI welds as falling within the ND SAPs category of Highest Integrity Components. Consequently, I have applied the principles EMC.1 to EMC.3 to both HSS and HI welds, although I have taken into account that there are differences between the consequences of failure for the two categories of weld.

3.2 Component Safety Reports

37 For components assigned to the Standard Class 1 structural integrity class Westinghouse has deemed that compliance with the ASME III (Ref. 30) provides sufficient assurance of integrity. A component safety report (CSR) has been provided for each of the Standard Class 1 components above which gives arguments and evidence to support the following three claims.

- Quality of Build - High quality is achieved through good design and manufacture.
- Good design - Achieved through compliance with ASME III.
- Avoidance of in-service degradation.

38 In the case of the only HI component, the reactor coolant loop, this has a CSR which includes the claims made for Class 1 components plus additional claims as follows:

- Forewarning of failure - provided by in-service inspection.
- Gross failure is remote. - achieved by a leak before break argument.
- The welds are tolerant to defects and are as defect free as practicable.

39 The last claim is supported by fracture assessment to determine the limiting defect size and qualified manufacturing inspections to confirm that these welds enter service as defect-free as practicable and certainly free from defects greater than about half the limiting size. This is an important aspect of the safety case in order to demonstrate compliance with SAPs EMC.1 to EMC.3.

40 For the HSS components the CSRs have been prepared following the TAGSI guidance with 4 safety case legs (Ref. 113). These are :-

- Good Design and Manufacture.
- Functional Testing.
- Failure Analysis including a Demonstration of Avoidance of Fracture.
- Forewarning of Failure.

41 The “Good design and Manufacture” leg is supported by arguments which show that the design complies with ASME III (Ref. 30) and has developed from existing PWR designs with the new features being introduced to simplify the design or improve it. Where possible, known degradation mechanisms have been designed out. The manufacturing is subject to high levels of QA and inspection throughout. The final non- destructive testing after post-weld heat treatment will be qualified with the ENIQ methodology (Ref. 112) with the intention that it will be capable of reliably detecting all defects significantly smaller than the limiting defect size determined in the fracture mechanics assessment; a factor of two being the target. Within GDA the work programme has included a representative sample of the welds with reduced scope technical justifications for the inspections.

42 “Functional testing” involves hydrostatic pressure testing both in the manufacturing shop and on site as part of the system hydrotest. Both tests are performed at 21.55MPa which is 1.25 times the design pressure of 17.24MPa. A range of non-destructive examinations are performed after the works hydrostatic test.

43 “Failure Analysis” includes an assessment of through-life degradation mechanisms and employs two types of fracture analyses. Firstly linear-elastic fracture mechanics is used to support the design as required by ASME III Appendix G, and secondly for the avoidance of fracture demonstration the elastic-plastic R6 (Ref. 50) approach has been employed as normally used in the UK. In the latter case the estimated end-of-life fracture toughness is used and crack growth is calculated using predicted through-life loadings to derive the End-of-Life Limiting Defect Size (ELLDS). Within GDA only a sample of representative and challenging welds have been assessed with R6.

44 “Forewarning of Failure” is provided by an extensive in-service inspection (ISI) programme, diverse systems to detect leakage and material surveillance programmes. The plant has been designed with accessibility for inspection in mind which should ensure that good quality inspections can be performed.

45 Component Safety Reports (CSR) were prepared for each of the components listed in the table above. In each case the Safety Function Requirements (SFR) were identified and a case developed following the structure outlined above. The cases for the HSS and HII components are summarised below.

3.2.1 Reactor Pressure Vessel

46 The Reactor Vessel Component Safety Report (Ref. 17) sets out the safety case for the RPV in claims, argument and evidence format. The following overview is derived from that report. Note that the Westinghouse report refers to this as the Reactor Vessel but in common with normal UK usage I will refer to it as the RPV.

3.2.1.1 RPV Function

47 The RPV is the high-pressure containment boundary used to support and enclose the reactor core. It maintains a volume of coolant around the core and the reactor internals control the flow through the core. The vessel interfaces with the reactor internals, primary loop piping, safety injection piping, and steel structures of the integrated head package (IHP) and is supported by RPV supports on the containment building's internal structures.

48 Inlet and outlet nozzles are provided to accommodate the flow of reactor coolant that circulates through the core to remove heat and transfer it to the steam generators. Direct Vessel Injection (DVI) nozzles are provided for the Passive Core Cooling System (PXS), which provides flow for a variety of accident conditions. To minimize the potential for draining the RPV and exposing the core, all penetrations are located above the core.

49 In addition to providing access to the inside of the vessel for refuelling and maintenance, the removable closure head also serves as the attachment point for the Control Rod Drive Mechanisms (CRDMs), the guide tubes for the in-core instrumentation, and certain other small penetrations. In order to support the Integrated Head Package (IHP) and accommodate lifting, supports and lift lugs are attached to the closure head.

3.2.1.2 RPV Design

50 The RPV consists of a cylindrical main section with a transition ring, hemispherical bottom head, and a removable flanged hemispherical upper head. The RPV (including closure head) is about 12.2m long and has an inner diameter at the core region of about 4.0m. The total weight of the vessel (including closure head) is approximately 352 tonnes. Surfaces, including the upper shell top surface, which can become wetted during operation and refuelling are clad to a nominal 5.6mm of thickness with Type 308L/309L stainless steel welded overlay. The RPV's design objective is to withstand the design environment of 17.2MPa and 343°C for 60 years.

51 The cylindrical section consists of two shells, the upper shell and the lower shell. The upper and lower shells, transition ring and the lower hemispherical head are fabricated from low alloy steel (ASME SA508, Grade 3, Class 1) and are welded together to make the RPV. To reduce irradiation damage to the welds (which are more vulnerable to irradiation damage) there are no welds within the active fuel region.

52 The removable flanged hemispherical upper head consists of a single forging, which includes the closure head flange and the closure head dome. The closure head is fabricated from the same low alloy steel forging as the rest of the RPV and similarly clad with austenitic stainless steel. It is attached to the vessel by a ring of 45 studs. Two metal o-rings are used for sealing the two assemblies and inner and outer monitor tubes are provided through the upper shell to collect any leakage past the o-rings.

53 There are no penetrations below the top of the core. This reduces the frequency of a loss of coolant accident (LOCA) from these penetrations that could allow the core to be uncovered. The core is positioned as low as possible in the vessel to limit re-flood time in

the event of an accident. During the beyond-design-basis severe accident core meltdown, the RPV bottom head retains the molten core and is cooled by flooding of containment to immerse the RPV.

3.2.1.3 RPV Safety Case

54 Ten Safety Case Requirements are identified for the RPV:

- The RPV is required to provide the highest reliability pressure boundary to contain the primary coolant, heat generating reactor core, and fuel fission products during normal and faulted design basis conditions for the design life of the plant.
- The RPV is required to provide support for the reactor internals and core to ensure that the core remains in a coolable configuration.
- The RPV is required to direct main coolant flow through the core by close interface with the reactor internals and flow skirt.
- The RPV is required to provide for core internals location and alignment.
- The RPV is required to provide support and alignment for the control rod drive mechanisms and in-core instrumentation assemblies.
- The RPV is required to provide support and alignment for the integrated head assembly.
- The RPV is required to provide an effective seal between the refuelling cavity and sump during refuelling operations.
- The RPV is required to support and locate the main coolant loop piping.
- The RPV is required to provide support for safety injection flow paths.
- The RPV is required to serve as a heat exchanger during core meltdown scenario with water on the outside surface of the vessel.

55 The RPV CSR demonstrates safety using the four HSS safety case legs in the way described above. In addition to the generic components of all HSS cases the following aspects of this case are of particular note:

- The design takes account of operating experience for other plant and identifies 15 known degradation mechanisms for which modifications have been made for the AP1000 to provide mitigation.
- The use of forgings is maximised to reduce the number of welds and in particular to ensure that there is no weld in the active core region.
- There are tight controls on the chemical composition of the base materials and the welds to minimise irradiation embrittlement.
- Fracture toughness tests will be performed on all ferritic base materials and welds.
- Surveillance specimens will be installed within the RPV to determine the effect of radiation on the RPV fracture toughness.

3.2.2 Pressuriser

56 The Pressuriser Safety Report (Ref. 21) sets out the safety case for the Pressuriser (PZR) in claims, argument and evidence format. The following overview is derived from that report.

3.2.2.1 Pressuriser Function

57 The PZR is designed to accommodate positive and negative surges through the surge line connecting the bottom head to a hot leg. During normal transient events, the pressure increases caused by insurges are controlled/mitigated by the PZR spray, such that the high pressure reactor trip setpoint is not reached. During pressure decreases (outsurges), automatic operation of electric heaters keeps the pressure above the low pressure reactor trip setpoint. The heaters are also energised on high water level during insurges to heat the sub-cooled surge water entering the PZR from the reactor coolant loop.

58 A screen at the surge line nozzle, as well as baffles in the lower section of the PZR, prevents cold insurge water from flowing directly to the steam/water interface. The screen and baffles also assist in mixing.

59 The PZR contains electrical heaters installed through the bottom head of the vessel. These heaters are removable for maintenance or replacement. The total heater capacity is selected to provide sufficient heating in the PZR during start-up operation to meet the RCS heat-up requirements.

3.2.2.2 Pressuriser Design

60 The PZR is a vertical, cylindrical vessel having hemispherical top and bottom heads constructed of low alloy steel, SA508 Grade 3 Class 2. Internal surfaces exposed to the reactor coolant are clad using austenitic steel. The AP1000 PZR is about 13.8 m long and has an inner diameter of 2.54 m. The minimum free internal volume for the PZR is 59.5 m^3. The upper head of the vessel is penetrated by one spray nozzle, two safety relief nozzles, four instrumentation nozzles and one temperature nozzle. One manway, four instrumentation nozzles, one sampling nozzle and one temperature nozzle are located in the cylindrical shell. The lower head is penetrated by one surge nozzle and 78 heater assemblies, which are inserted from the bottom head through heater sleeve assemblies.

61 To limit the number of welds required the surge nozzle is integrally forged with the bottom head and the safety/relief nozzle and spray nozzle are both integrally forged with the upper head.

62 The safety relief and surge nozzles terminate in Type 316 austenitic stainless steel safe ends, along with temperature and instrumentation nozzles. The spray nozzle terminates in an Alloy 690 safe end. The safe ends are of sufficient length to prevent damage to the transition weld on the nozzle during field welding. The surge, safety relief and ferritic nozzle weld preparations are buttered prior to final post weld heat treatment (PWHT). After final PWHT, the safe ends are welded to the buttered nozzles. The internal upper head spray nozzle region is first buttered and then stress relieved prior to welding the spray nozzle nipple. The dissimilar metal welds between the PRZ and the safe ends are HSS whilst the safe end to RCL welds are Class 1.

3.2.2.3 Pressuriser Safety Case

63 The PZR Component Safety Reports (CSR) demonstrates safety using the four HSS safety case legs in the way described above. It takes account of the following required safety functions.

- The PZR is required to maintain the integrity of the primary coolant pressure boundary during standby, normal operation and under design basis faulted conditions for the design life of the plant.
- The PZR is required to provide the point in the Reactor Coolant System (RCS) where system pressure is controlled during steady state operations and transients, to ensure minimum pressure requirements associated with core coolant boiling and departure from nucleate boiling limits are maintained.
- The PZR is required to provide the controlled volume from which the level of reactor coolant can be measured.
- The PZR is required to contain the water volume which is used to maintain RCS volume in the event of a minor system leak for a reasonable time without replenishment.

3.2.3 Steam Generator

64 The Steam Generator Component Safety Report (Ref. 22) sets out the safety case for the steam generator in claims, argument and evidence format. The following overview is derived from that report.

3.2.3.1 Steam Generator Function

65 During normal plant operation, the Reactor Coolant Pumps (RCPs) circulate pressurised reactor coolant through the RPV then through the Steam Generators (SGs). The SGs serve as heat exchangers by converting secondary water into steam from heat produced in the RPV. Reactor coolant flowing through tubes in the SG boils secondary water on the shell side to produce steam in the secondary loop that is delivered to the turbines. The steam is subsequently condensed via cooled water from the tertiary loop and returned to the SG to be heated once again. The reactor coolant is returned to the reactor vessel by the pumps to repeat the process.

3.2.3.2 Steam Generator Design

66 The SG shell comprises a transition cone, shell barrels, tubesheet, channel head, inlet nozzle, outlet nozzle, passive residual heat removal (PRHR) nozzle, main and start-up feedwater nozzles, trunnions, manways and elliptical head. The SG measures approximately 22.5m from steam outlet nozzle at its top to the flat, exterior portion of the channel head at its lower end. The inner diameter of the upper shell is 5.33 m, and that of the lower shell is 4.19 m. The maximum outer diameter of the SG is 5.58 m. All parts of the SG shell are low-alloy steel forgings. The ring forgings of the SG shell are connected by girth welds, and all welds on the surfaces of the shell, including nozzle attachments, are ground to remove discontinuities and stress raisers as well as to facilitate examinations and inspections.

67 The channel head forms the lower part of the SG. A single primary inlet nozzle connects to the 787 mm inner diameter (ID) hot leg reactor coolant system pipe. The channel head is divided into inlet (hot leg) and outlet (cold leg) primary chambers by a vertical divider plate that extends from the centre of the channel head to the tubesheet. The lower portion of these chambers is spherical and merges into a cylindrical portion, which mates to the tubesheet. Reactor coolant flow exits the SG through two 657 mm ID pump suction nozzles in the channel head. Two RCP casings are directly connected to and supported by the SG channel head at the pump suction nozzles. The interior surfaces of the channel head, primary nozzles, and primary manways are clad with weld deposited austenitic stainless steel of finished nominal thickness 5.9 mm (finished minimum clad thickness 3.9 mm).

68 The principal SG primary side components include the channel head, tubesheet, and tube bundle. These form part of the RCS pressure boundary, which provides a barrier against the release of radioactivity generated within the reactor. Each SG has 10,025 heat-transfer U-tubes which are welded to the tubesheet inside the channel head. The SG tubing is 17.5 mm outside diameter and 1.02 mm nominal wall thickness. The minimum U-bend radius is 82.6 mm. The tubes are fabricated of nickel-chromium-iron Alloy 690 and undergo thermal treatment following tube-forming operations.

69 Within the SG shell, the principal secondary side components include the feedwater ring, moisture separating equipment and tube supports. Support of the tubes is provided by ferritic stainless steel tube support plates. Holes in the tube support plates are broached with a hole geometry to promote flow along the tube and to provide an appropriate interface between the tube support plate and the tube. Anti-vibration bars (AVBs) are installed in the U-bend region of the tube bundle. The tube bundle is surrounded by a wrapper, forming an annulus for secondary side feedwater between the wrapper and the shell of the SG. The wrapper is supported by lugs attached to the inside of the SG Shell.

3.2.3.3 Steam Generator Safety case

70 The SG CSR demonstrates safety using the four HSS safety case legs in the way described above. In this assessment it takes account of the following required safety functions.

- The SG pressure boundary is required to maintain the integrity of the primary and secondary coolant boundaries during standby, normal operation and under design basis faulted conditions for the design life of the plant.
- The SG secondary side is required to provide a heat sink for the RCS during power operations and anticipated transients and under natural circulation conditions in accordance with component performance requirements (not required to provide safety-related safe shutdown of the plant).

3.2.4 Main Steam Line

71 The Main Steam Line Component Safety Report (Ref. 27) sets out the safety case for the Main Steam Line (MSL) in claims, argument and evidence format. The following overview is derived from that report.

3.2.4.1 Main Steam Line Function

72 The Steam Generator System (SGS) (and hence MSL) is required during start-up and operation at power, to remove the heat produced by the reactor core; and also in the initial phase of shutdown operation before the normal residual heat removal system can be connected, to remove the decay heat from the reactor core. The heat sinks are provided either by venting steam to atmosphere or by the turbine bypass system dumping steam to the condenser. Each MSL goes from its steam generator and passes through the containment boundary.

3.2.4.2 Main Steam Line Design

73 The MSL is the part of the SGS which transports and distributes steam to the main steam system and then to the Main Turbine System (MTS) during power generation and directly to the main condenser when the MTS is not available or following a large reduction in turbine generator load. The MSL is routed from the top of each of the two SGs dropping to a horizontal run below the operating deck. From there, the MSLs separately penetrate the side of the containment vessel and are routed through their own main steam and feedwater isolation valve compartment provided for each loop in the auxiliary building, to the turbine island. The MSL pipe is 965 mm nominal pipe diameter (outside diameter) with 44.2 mm minimum wall thickness, and is manufactured from ASME SA-335 Gr P11 which is seamless ferritic alloy-steel pipe for high temperature service. The MSL is fabricated from various pipe lengths using tungsten inert gas (TIG) welding.

3.2.4.3 Main Steam Line Safety Case

74 The MSL CSR demonstrates safety using the four HSS safety case legs in the way described above. In this assessment it takes account of the following required safety functions.

- The MSL is required to maintain the integrity of the secondary coolant pressure boundary during standby, normal operation and under design basis faulted conditions for the design life of the plant.
- MSLs are required to exhibit Leak Before Break (LBB) behaviour in the event of a through wall defect developing (defence in depth).
- The pipework in the vicinity of the containment penetrations is required to maintain the integrity of the pressure boundary during standby, normal operation and under design basis faulted conditions for the design life of the plant.
- The pipework in the vicinity of the containment penetrations above 50mm diameter is required to exhibit LBB behaviour in the event of a through wall defect developing (defence in depth). (This sentence is a verbatim extract from the CSR but Westinghouse has subsequently clarified that it is only the MSL pipework above 50 mm in diameter inside containment which is required to exhibit LBB behaviour).

3.2.5 Reactor Coolant Loop Piping

75 The Reactor Coolant Loop Piping Component Safety Report (Ref. 26) sets out the safety case for the reactor coolant loop (RCL) in claims, argument and evidence format. The following overview is derived from that report.

3.2.5.1 Reactor Coolant Loop Piping Function

76 During operation, the RCPs circulate pressurised water through the RCL and primary components. The pressurised water, which serves as coolant, moderator, and solvent for boric acid is heated as it passes through the core in the RPV. It next flows via the hot legs into the SGs where the heat is transferred to the SG secondary side water, and then is returned to the RPV via the RCPs and cold legs to repeat the process. The surge line connects the RCL to the PZR to provide a means of controlling RCS pressure within specified limits.

3.2.5.2 Reactor Coolant Loop Piping Design

77 The RCS includes two heat transfer loops, each of which contains a steam generator (SG). Attached to each SG are two RCPs, a single hot leg, and two cold legs for circulating reactor coolant between the RPV and the SGs. In addition, the RCS includes a pressuriser (PZR), interconnecting piping and valves, and instrumentation necessary for operational control and safeguards actuation. All reactor coolant system equipment is located in the reactor containment. The RCL piping forms part of the reactor coolant pressure boundary, which provides a barrier against the release of radioactivity generated within the reactor and is designed to provide a high degree of integrity throughout operation of the plant. The RCS piping comprises of the following:

- Two single piece seamless forged austenitic stainless steel (ASME SA-376) piping sections, known as hot legs. Each hot leg is connected via a safe end to an outlet nozzle on the RPV, and at the other end via a safe end to an inlet nozzle of each SG. The nominal wall thickness of the hot leg piping is 82.6 mm and the inner diameter (ID) is 787 mm.
- Four single seamless forged austenitic stainless steel (ASME SA-376) piping sections, known as cold legs. Each cold leg is welded via a safe end to an inlet nozzle on the RPV and at the other end via a safe end to the casing of a RCP; the RCP casings are in turn connected at the upstream end to SG outlet nozzles. The nominal wall thickness of the cold leg piping is 65 mm and the ID is 559 mm.
- A single piece seamless forged austenitic stainless steel (ASME SA-312) surge line connects one hot leg (Loop 1) to the PZR. The surge line is welded at one end to a nozzle in the hot leg, and at the other end to the PZR outlet nozzle safe end.

3.2.5.3 Reactor Coolant Loop Piping Safety case

78 The RCL CSR identified the following two safety functions for the RCL:

- The pipework is required to maintain the integrity of the RCS during standby, normal operation and under design basis faulted conditions for the design life of the plant.
- The pipework is required to exhibit LBB behaviour in the event of a through wall defect developing (defence in depth).

79 The RCL piping components have generally been determined to have a structural integrity classification of Standard Class 1, with the exception of the austenitic welds joining the RCL hot leg and cold leg to the safe ends of the RPV nozzles which are allocated the higher classification of HI. This HI classification has been conservatively

adopted since the RPV compartment module has not been designed against the internal pressurisation associated with a guillotine break of the RCS piping, and also because there is perceived to be some uncertainty regarding the effect of jet loading on the RPV internals.

80 As described in Section 3.2 above. the safety case for the standard class 1 welds is based on compliance with the ASME code whereas for the HI components the safety case has additional evidence:

- Forewarning of failure is provided by in-service inspection.
- A leak before break argument provides evidence that gross failure is remote.
- The HI welds are tolerant to defects and are as defect free as practicable.

4 GDA STEP 4 NUCLEAR DIRECTORATE ASSESSMENT FOR STRUCTURAL INTEGRITY

4.1 Categorisation and Classification of Structures, Systems and Components

4.1.1 Background, Summary of Step 3 Activities and Definition of Step 4 Actions

81 ND's SAPs (Ref. 4) recognise that the safety case for certain components and structures may need to be based on a claim that the likelihood of gross failure is so low that the consequences of gross failure can be discounted. These components are termed the 'highest reliability components'. As explained in the SAPs, this is an onerous route to constructing a safety case and measures over and above normal practice will be required to support and justify such a claim.

82 SAPs EMC.1 to EMC.3 identify those requirements which need to be applied over and above the normal principles applied to metal components and structures for the 'highest reliability' components', and Regulatory Observation Action RO-AP1000-19 'Avoidance of Fracture' was developed to capture these additional requirements.

83 RO-AP1000-18 'Categorisation and Classification of Structure, Systems and Components' was developed to identify those components to which these additional requirements should apply, i.e. those components where the likelihood of gross failure is claimed to be so low that it can be discounted.

84 The Step 3 Categorisation and Classification activities were focussed on obtaining a commitment from the RP to develop a systematic methodology to identify these 'highest reliability' components through Regulatory Observation Action RO-AP1000-18.A1, and Westinghouse's proposals were given in Ref. 68.

85 The Step 4 activities have focussed on the development and implementation of the methodology through Regulatory Observation Action RO-AP1000-18.A2 and ultimately the justification of the list of components whose likelihood of gross failure is so low that it can be discounted.

4.1.2 Justification of the List of Components Whose Likelihood of Failure Is So Low That It Can Be Discounted

86 This activity assesses the justification for the list of components whose likelihood of gross failure is discounted, referred to by Westinghouse as the HSS Components. The Step 4 Assessment Plan (Ref. 1) refers to this under activity AR09058-1 as 'Need to determine the final list of components with a conclusion for the basis for including or excluding specific components'.

4.1.2.1 Methodology

87 Westinghouse has undertaken a structured and systematic review of consequences of failure of the components listed below to identify what level of structural reliability is required and hence which components fall into the HSS Category, in the Structural Integrity Classification Report, Ref. 16. These are all Class 1 components from the Safety Categorisation and Classification of Systems report, Ref. 77.

- Reactor Vessel.
- Steam Generator Secondary Side.

- Steam Generator Primary Side.
- Pressuriser.
- Reactor Coolant Pump Casing and Flywheel.
- Reactor Coolant Loop (Hot and Cold legs).
- Core Make-Up Tank.
- Accumulator.
- Passive Residual Heat Removal Heat Exchanger.
- RV Internals, Core Support Structures, Core Barrel, Upper/Lower Supports.
- Main Steam Lines Inside Containment.
- Pipework in way of Containment Penetrations (Main Steam, Main Feedwater) – Outside Containment.
- DVI Line.
- Pressuriser Surge Line.
- Automatic Depressurisation System Piping and Safety Valve Piping.
- Passive residual Heat Removal Heat Exchanger Inlet / Outlet Lines.

88 The categorisation system adopted in Ref.16 is specific to the structural integrity safety case and is based on an approach adopted for structural integrity cases on the Advanced Gas Cooled Reactors (AGRs) operating in the UK. The systems breaks the Class 1 components into three categories:

- Highest Safety Significance (HSS)
- High Integrity (HI)
- Standard Class 1 Components

89 According to this methodology categories are assigned by establishing the level of structural reliability required from each component based on an understanding of the consequences of failure both in terms of the direct consequences for example loss of coolant inventory and indirect consequences such as missiles pipewhip, etc.

90 The highest reliability category, the HSS category, is assigned where the consequences of gross failure cannot be shown to be tolerable and where it is not reasonably practical to provide protection. The lowest reliability for a Class 1 component is Standard Class 1 and is assigned where the consequences of gross failure will lead to only ‘limited’ core damage. The intermediate reliability category, HI, is assigned where failure can lead to severe core damage, but effective containment limits the off-site consequences.

91 For HSS components the necessary reliability is assured by measures over and above normal practice support and justify compliance with SAPs EMC 1 to 3 for the ‘highest reliability components’. For the Standard Class 1 components reliability is essentially ensured by compliance with normal relevant design and construction codes and practice, for example ASME III, Ref. 30. The HI components are somewhere in-between with some but not all the additional measures for the ‘highest reliability components’, for example a defect tolerance study will be undertaken, but qualification of the

manufacturing inspections may not be necessary depending on the outcome of such studies.

92 The categorisation methodology is well known and well understood in the context of safety cases developed for AGRs in the UK. Its use in the context of a civil PWR is new to the UK. Ref. 16 rightly states that there is no prescriptive methodology, but I would make the following observations on the approach:

- The structural categorisation process used a multi-disciplinary expert review panel to identify the required level of structural reliability based on the criteria defined in Ref. 16. These reliability requirements are established by considering the consequences of gross failure in terms of levels of core damage and the International Radiological and Nuclear Event Scale (INES) Level. However, whilst the requirements are defined on the basis of these consequences, it was not clear to me that the reliability requirements had any links back to the fault studies/severe accident/PSA safety cases. For example although a weld may have been allocated an intermediate HI category by the expert panel, I do not believe that the implication of this reliability category is reflected in the fault studies/severe accident/PSA cases for the system containing that weld. In my opinion the approach cannot be used in isolation. Without a linkage back to the fault studies/severe accident/PSA cases then the overall safety case for the reactor cannot take account of the implied structural reliability and hence the structural integrity classification is not properly integrated into the safety case.
- There are very few components/welds in the intermediate HI category; in fact it is only the RPV Nozzle safe end to Main Coolant Loop welds which fall into this category. Thus components seem to logically fall into either the 'highest reliability' category or the Standard Class 1 category. Thus the sophistication of attempting to link gross failure to level of core damage and INES levels is largely redundant.
- The position of an HI component/weld in terms of the extent of the consequence arguments is not well defined. Ref.16 para. 3.4.2 on HI components/welds, states that a full justification of the claims for the consequence arguments should be provided, which may require sophisticated consequence analyses. However, Appendix B of Ref. 16 shows the RPV Nozzle safe end to Main Coolant Loop welds are included as HI welds largely as a precautionary measure because of uncertainty in some aspects of providing a consequence assessment, which appears to go against the principle established in the main text. From a regulatory perspective a HI component/weld is not an HSS component. As such the reliability of these components/welds is not sufficiently high that the consequences of failure can be discounted and the expectation is that it will be necessary to provide a full consequence assessment.
- In addition the position of an HI component/weld in terms of the Avoidance of Fracture Demonstration is not well defined. Para. 3.4.2 of Ref. 16 notes that a defect tolerance study will be required, and the subsequent inspection requirements will depend on the outcome of those studies. Targeted qualified inspections may be required, but there is no commentary on what size of defect would lead to the need to formally qualify the manufacturing inspection in practice. The Component Safety Report on the Main Coolant Loop (Ref. 26) states that the extent of inspection qualification will be tailored according to the calculated defect tolerance which is consistent, but non-specific. However, in terms of GDA the structural integrity underpin for the HI weld is exactly the same as if the weld had been an HSS weld in terms of the RO-AP1000-19

Avoidance of Fracture demonstration as both the fracture mechanics calculation and the inspection qualification work have both been undertaken.

93 Thus I have reservations about application of the categorisation methodology in the context of the AP1000 safety case in terms of the linkage into the rest of the safety case and the need and position of the intermediate HI category.

94 However, I believe that the methodology can still be effective in identifying those components where the safety case needs to be based on showing the likelihood of failure is so low that the consequences of failure can be discounted, and those components where there is a case to show that the consequences of failure are acceptable.

95 For GDA the uncertainty in the consequence argument and final structural integrity case for the HI welds can be addressed by looking at the evidence in terms of a full HSS case as the necessary elements of the Avoidance of Fracture demonstration are available, but with a fall back that it may be possible to rely on demonstration that the consequences of failure are acceptable. The final position on the HI welds will be a matter for the Licensee as they will be determining the extent of the inspection qualification and through life monitoring. However, if the HI category is retained for the Licensing phase it must be recognised that the expectation will be for a full assessment of the consequences of failure as the weld/component is not in the HSS category.

96 Thus for the purposes of GDA this is a satisfactory methodology, but for the post GDA a Licensee will have to review the approach to determine whether the methodology is appropriate in the context of the AP1000 safety case and whether the definition and use of an intermediate HI component/weld category is relevant and useful in terms of the overall safety case. This is taken forward as Assessment Finding **AF-AP1000-SI-01.**

4.1.2.1.1 Conclusions and Findings on Structural Integrity Classification Methodology

97 I am satisfied that the Structural Integrity Classification Methodology adopted by Westinghouse can be used as a basis for identifying those components where the likelihood of failure is so low that the consequences of failure can be discounted.

98 I have concerns with regard to the use of the intermediate HI component/weld category, but will primarily consider these as HSS welds in terms of the GDA demonstration, but with the fall back that it may be possible to rely on a consequences argument. The position of the HI component/weld category will need to be resolved during licensing phase, and I have raised the following Assessment Finding.

> ***AF-AP1000-SI-01:*** *The Licensee shall review the structural integrity classification scheme to determine whether the definition and use of an intermediate HI component/weld category is relevant and useful in terms of the overall safety case for the UK AP1000.*

99 This work shall be undertaken before the generic milestone of Install RPV as any change in the Structural Integrity Classification scheme beyond that point could be very difficult as the components will start to be installed, and any substantive changes could then lead to substantial delays and additional costs.

4.1.2.2 Assessment of Categorisation

100 The categorisation process took the form of an expert review panel using members knowledgeable on the AP1000 design, layout and transient response and the UK

classification methodology. The background and experience of the review panel is described in Ref. 16, and I am satisfied that it was fit for purpose.

101 The panel considered each of the Class 1 components identified above and output from their consideration is shown on the component assessment sheets shown in Appendix B of Ref.16. As a result of their deliberations the panel concluded that the following components should be treated as HSS (including the dissimilar metal safe end weld connections on the components where these exist):

- Reactor Vessel.
- Pressuriser.
- Steam Generator Secondary Shell, Tube Sheet and Channel Head.
- Main Steam Line Inside containment.

102 In addition the panel concluded that the Reactor Vessel Safe End to Main Coolant Loop welds should be classed as HI welds.

103 I have assessed the categorisation conclusions by looking in more detail at the component assessment sheets in Appendix B for a range of components that were considered to be Standard Class 1 components to ensure that the consequences of gross failure have been adequately addressed.

104 The assessments are based on a combination of identifying the existing deterministic consequence analyses available for failure of the component and expert judgement where deterministic consequence analyses were not available.

105 From my review it is clear that the Standard Class 1 components generally have a deterministic thermal hydraulic consequence analysis available, but that the indirect consequence analyses are not available and are generally based on expert judgement. For example on the Accumulators (Table B10 of Ref. 16) the LOCA resulting from the failure of an accumulator would be expected to be isolated by two check valves, but if these were not effective the failure would be bounded by a DVI line break, which is within the AP1000 design basis. However, the indirect missile/blast effects from disruptive failure of the accumulator shell has not been analysed using deterministic approaches, and expert judgement was used to conclude that the individual steel/concrete/steel accumulator compartment would prevent damage to essential safety systems in adjacent compartments or the redundant train of the passive core cooling components.

106 I therefore raised TQ-AP1000-1045 Ref. 9 to sample what evidence was available to support the expert judgements on the indirect consequence assessments for missile generation from failure of the Accumulator, containment of pipewhip from the Main Coolant Loop Pipework and missile generation from failure of the Reactor Coolant Pump bowl.

4.1.2.2.1 Missile Generation from the Accumulator and Reactor Coolant Pump Bowl

107 The TQ response acknowledged that disruptive failure of the major vessels is outside the generic AP1000 design basis and as no specific analysis had been undertaken, expert judgement had been used to conclude that disruptive failure would be contained within the compartment. The TQ therefore presented a missile impact assessment using the R3 Impact Assessment Procedure, Ref. 80, to confirm the judgements reached by the expert

review panel. This procedure was originally developed by UK's Central Electricity Generating Board and is suitable basis for undertaking this type of assessment.

108 The R3 analyses concluded that the compartment surrounding the accumulator should retain the missiles from a gross failure of the accumulator, but it could not fully discount the possibility of a worst case missile perforating the compartment ceiling. This was thought unlikely due to a number of conservatisms in the analysis, but further consideration was given to the consequences of a missile penetrating the compartment above, and it was concluded that this would not further escalate the consequences of failure. Although the impact analysis was relatively simplistic, I was satisfied that it should be conservative and hence the result supported the expert judgement.

109 In the case of the RCP bowl it was concluded that the failure would neither penetrate the compartment nor cause a consequential failure of the SG Channel Head. The analysis assumed that a gas filled RCP bowl would be bounding, but I was concerned that this may be non-conservative compared with high temperature pressurised water, and I also wished to check that compartment was sufficiently open to prevent a large over-pressurisation following catastrophic failure.

110 Westinghouse therefore provided supplementary information in Ref. 79 which accepted that the gas filled assumption was non-conservative, and if all the energy from the flashing off the water could be transferred into the velocity of the missile then the missile velocity could slightly exceed the penetration velocity of the concrete walls. Westinghouse argued that this was a very pessimistic assumption, but that in any case the steel-concrete composite wall would give enhanced impact resistance in any case compared with that of the traditional reinforced concrete assumed in the analysis. In addition Westinghouse provided three dimensional views of the RCP compartment to show that there are large vent paths through to containment so pressure build up due to steam flash off following catastrophic failure should not be a problem.

111 Again the impact analysis is quite simplistic, and it was of concern that an initial bounding assumption was subsequently found to be non-conservative, but on balance I accept that the analyses support the expert judgement on the indirect consequences of failure of the RCP bowl in terms of penetrating the compartment surrounding the RCP and that they should not penetrate the channel head.

112 The other potential concern on consequential damage from failure of the RCP bowl is the SG supports. The structural integrity classification document, Ref. 16, discounts the possibility of failing the SG supports as not credible, and I asked for supporting evidence as part of TQ-AP1000-1045, Ref. 9. Unfortunately the response to TQ-AP1000-1045, and supplementary information provided in Ref. 79 does not address the effect RCP bowl failure on the SG support.

113 Initially I thought this lack of a response was because the SG supports are remote from the RCP bowl, but in practice the supports which are remote from the RCP bowl only provide lateral restraint, and the vertical restraint is provided by a vertical support which runs directly between the two RCP bowls before it attaches to the SG lower head. It is therefore not clear to me that damage to this vertical support can be discounted as not credible without the supporting evidence.

114 I only became aware of this problem late in the assessment process and Westinghouse was unable to provide any supporting evidence in the remaining time available. The matter therefore needs to be addressed through a GDA Issue to ensure that the evidence can be provided to support the Standard Class 1 classification for the RCP bowl. I have

taken the matter forward through Action 3 of GDA Issue **GI-AP1000-SI-06** on Structural Integrity Categorisation and Classification for Westinghouse to provide the arguments and evidence to show that gross failure of the RCP bowl would not challenge the effectiveness of the vertical support for the SG. If it proves difficult to provide the necessary evidence, then Westinghouse would have to consider upgrading the classification of the RCP bowl to HSS and develop a case to show that the likelihood of gross failure is so low that it can be discounted.

4.1.2.2.2 Pipe-whip from Main Coolant Loop Pipework

115 I also sought evidence to support the argument that the penetration of the MCL pipework through the compartment walls would be adequate to contain the effects of pipe-whip from disruptive failure of the MCP pipework.

116 The TQ response notes that the MCL is fabricated to minimise the number of welds, and the only welds are where it connects onto the RPV, SG and RCP. This is a useful design feature, but the TQ response then focuses on the HI nature of the RPV nozzle weld rather than evidence to show that the primary shield wall will limit pipe-whip.

117 Thus no additional evidence was provided to show that the primary shield wall will be adequate to constrain the pipe failure. I am generally content that it should be possible to make a case for the primary shield wall to provide such constraint and will therefore accept the judgement of Westinghouse's expert panel that this is the case without further evidence for the purposes of moving to an IDAC however, it will need to be shown that this is the case before I would be prepared to support a DAC. This is being taken forward by the Internal Hazards assessment team as part of the broader GDA Issue relating to pressure part failure and pipe-whip (**GI-AP1000-IH-03**), see the Step 4 assessment of Internal Hazards (Ref. 205).

4.1.2.2.3 Conclusions and Findings on Assessment of Categorisation

118 I am satisfied that Westinghouse has undertaken the categorisation process in a thorough and systematic manner, considering all the necessary Safety Class 1 components.

119 It is clear that whilst deterministic thermal hydraulic analyses were generally available to consider the direct consequences of a disruptive failure, an element of expert judgement was required to reach a conclusion on the indirect consequences of failure as the necessary consequence assessments were not available. I challenged these judgements on a sample basis, and Westinghouse undertook additional analysis work which, although relatively simplistic, did support the original judgements of the review panel. However, there were two areas where the evidence was not provided to show that indirect consequences of failure were acceptable. These were the effect of RCP bowl failure on the SG vertical support and the ability of the primary shield wall to limit MCP pipe-whip.

120 In terms of the SG vertical support, Westinghouse will need to provide the evidence to support the Standard Class 1 classification for the RCP bowl. I have taken the matter forward through Action 3 of GDA Issue **GI-AP1000-SI-06** on Structural Integrity Categorisation and Classification.

GI-AP1000-SI-06: *Structural Integrity Categorisation and Classification. Actions 1 and 2 relating to this Issue are described in Section 4.9 on Generic Categorisation and Classification, but the key activity against Action 3 is to:*

Provide arguments and evidence to show that catastrophic failure of a reactor coolant pump bowl would not challenge the effectiveness of the vertical support for the steam generator.

121 The complete GDA Issue and associated action(s) are formally defined in Annex 2.

122 Evidence to show that the primary shield wall would be adequate to constrain MCL pipe-whip was not provided, and this will be taken forward as part of the broader GDA Issue relating to pressure part failure and pipe-whip (**GI-AP1000-IH-03**, see Ref. 205).

123 Thus for the purposes of GDA I am generally satisfied with the approach taken by Westinghouse to identify the HSS components. Further evidence is required to support the classification of the RCP Bowl, and that is being taken forward as a structural integrity GDA Issue, and further evidence is required to show that the primary shield wall will limit MCP pipe-whip and that is being taken forward through a broader GDA Issue of pressure part failure and pipe-whip. Providing the response to these GDA issues confirms the judgements already made by Westinghouse to date I am satisfied with the list of HSS components proposed by Westinghouse, i.e. the Reactor Vessel, Pressuriser, Steam Generator and Main Steam line Inside Containment, along with the HI weld of the Reactor Vessel Safe End to Main Coolant Loop weld.

124 Whilst the expert judgement is sufficient for defining the HSS boundary for the purposes of GDA, it will be necessary to address the indirect consequences of failure of the non-HSS components in a more formalised and systematic manner. An assessment of the indirect consequences of failure will be required for all non-HSS components, with arguments and evidence to support any claims being made on components or structures. This matter is being taken forward by the Internal Hazards assessment team as part of the broader GDA Issues, Pressure Part Failure (**GI-AP1000-IH-03**) and Internal Missile Safety Case (**GI-AP1000-IH-05**), see Ref. 205.

125 Post-GDA it will be necessary for the Licensee to review the structural integrity classification to remove the element of expert judgement in defining the HSS boundary by ensuring that the formalised assessments of the indirect consequences of failure of the Standard Class 1 and HI components/welds are fully reflected in the structural integrity classification scheme. I therefore raised the following Assessment Finding **AF-AP1000-SI-02**.

AF-AP1000-SI-02*. The Licensee shall review the structural integrity classification scheme to remove the element of expert judgement in defining the HSS boundary by ensuring that the formalised assessments of the indirect consequences of failure of the Standard Class 1 and HI components/welds are fully reflected in the structural integrity classification scheme.*

126 This work shall be undertaken before the generic milestone of Install RPV as upgrading any components from Standard Class 1 to HSS beyond that point could be very difficult as the components will start to be installed, and any substantive changes could then lead to substantial delays and additional costs.

4.1.2.3 Main Steam Line Classification

127 I have also looked in more detail at the reasoning behind the structural integrity classification of the Main Steam Lines (MSLs). Inside containment the lines are HSS components, whereas outside of containment in the main steam isolation valve compartment the MSLs (and main feed lines) are Standard Class 1.

128 The HSS categorisation within containment is based on the indirect consequences of failure. The guillotine failure of a single MSL is within the AP1000 design basis. However, although it was considered unlikely, the assessment concluded that it was not possible to preclude the potential of pipe whip from the casualty line to impinge and rupture the non-casualty line thus resulting in a double MSL break which is outside of the design basis and would lead to over-pressurisation of containment. Hence the MSL inside containment is classed as an HSS component. I believe this is a good example of the expert panel taking a conservative position on the classification as a result of being unable to preclude a certain scenario.

129 The case for the main steam line (and main feed lines) in the main steam isolation valve (MSIV) compartment being Standard Class 1 is complex due to the differences between the East and West MSIV compartments, as there is an additional concern with regard to consequential damage from the East MSIV compartment due to its location adjacent to the Main Control Room and a Class 1 electrical equipment room. There is therefore some discussion of including pipewhip restraints and jet barriers in the East MSIV compartment in Ref. 16, but these had yet to be finalised at the time Ref. 16 was written.

130 In addition I noted that whilst Ref. 16 accepted the possibility of a simultaneous failure of the main steam and main feed lines in the West MSIV, the equivalent section in the design transient document stated that this was not a credible scenario. I therefore raised TQ-AP1000-1219 to address this apparent anomaly and clarify the position with regard to the MSL in the MSIV compartments.

131 The response to the TQ identified that Ref. 16 was misleading and coincident failure of failure of a main steam and main feed line was not a credible event as this pipework in a break exclusion zone (BEZ) which provides high confidence that failure of a main steam line would not impact the main feedwater line. This statement appeared to be giving additional weight to the additional controls associated with the design fabrication and construction of BEZ pipework that was not reflected in the structural integrity classification associated with the UK AP1000. I discussed this matter with an Internal Hazards assessor, and it also appeared that the BEZ concept was also being applied to downgrade the measures that need to be claimed for the East MSIV compartment to protect the Main Control Room and a Class 1 electrical equipment room in the response to TQ-AP1000-1272.

132 I therefore sought additional clarification on whether any additional reliability was being sought from BEZ pipework that would necessitate an upgrading of the structural integrity classification of the main steam line and main feed line in the MSIV compartments. This additional clarification showed that no additional reliability should be claimed for the BEZ nature of these lines. A full consideration of the indirect consequences of failure must be considered on all necessary barriers and mitigation such as pipe whip restraints, and although a deterministic safety analysis for coincident steam and feed line break does not exist, the coincident failure in a single train is not considered to be limiting, and it is not credible to have coincident failures of both trains due to their separation.

133 This additional clarification was received very late in the assessment process in the form of informal e-mails. The re-assurance is sufficient to establish that no additional reliability should be claimed for the BEZ classification in the safety case compared to the Standard Class 1 structural integrity classification, but the position with regard to the claims needed to prevent or mitigate the effects of pipework failure in the MSIV compartments still needs to be properly established. This work included with the Internal Hazards GDA Issue, **GI-AP1000-IH-03** 'Provide substantiation to support claims and arguments made within the area of pressure part failure', and in particular Action 2 which explicitly considers the claims made in relation to pipework failure in the MSIV compartments, see Ref. 205. The complete GDA Issue and associated action(s) are formally defined in Annex 2.

134 In addition there is a need to ensure that case to classify this pipework as Standard Class 1 from a structural integrity perspective includes consideration of coincident failure of a main steam line and main feed line from a thermal hydraulic safety analysis perspective, and I have raised an Assessment Finding **AF-AP1000-SI-03** to address this matter.

4.1.2.3.1 Conclusions and Findings on Main Steam Line Classification

135 I am satisfied that the main steam line has been classified as HSS within containment. I am generally satisfied that the main steam and main feed lines are classified as Standard Class 1 outside of containment, but the Licensee will need to ensure that a case is available to demonstrate that coincident failure of a main steam and main feed line will not prove limiting, and have raised Assessment Finding **AF-AP1000-SI-03.**

> ***AF-AP1000-SI-03.*** *The Licensee shall ensure that the case for categorising the main steam line and main feed line in the main steam isolation valve compartment as Standard Class 1 components includes explicit evidence that coincident failure of a main steam line and main feed line will not be limiting from a thermal hydraulic safety analysis perspective.*

136 This work shall be undertaken before the generic milestone of Install RPV as although this pipework is not a long lead time item, it is important that the case for all Structural Integrity Classification has been fully established by that point in the construction phase as upgrading any components from Standard Class 1 to HSS beyond that point could be very difficult, as the components will start to be installed, and any substantive changes could then lead to substantial delays and additional costs.

4.1.3 Conclusions and Findings Relating to Categorisation

137 I am broadly satisfied with the process for identifying the components of HSS whose likelihood of failure has to be demonstrated to be so low that it may be discounted. However, I have reservations about the intermediate category called HI but will primarily consider these as HSS welds in terms of the GDA demonstration. Post GDA a Licensee will need to review whether the HI classification is relevant and useful for the safety case.

138 In addition the categorisation process has included an element of expert judgement. This has been adequate for to define the boundary for the purposes of GDA, but post GDA it will be necessary for the Licensee to remove the element of expert judgement by ensuring that the formalised assessments of the indirect consequences of failure of the Standard Class 1 and HI components/welds are fully reflected in the structural integrity classification scheme.

4.2 Avoidance of Fracture - Margins Based in Size of Crack-Like Defects, Integration of Material Toughness Properties, Non-Destructive Examinations During Manufacture and Analyses for Limiting Sizes of Crack-Like Defects

139 This activity continues the assessment which followed from Step 3 Regulatory Observation RO-AP1000-19.

4.2.1 Background, Summary of Step 3 Activities and Definition of Step 4 Actions

140 This activity is AR09058-2 on the Step 4 Action Plan (Table 2) and continues the assessment which followed from Step 3. For those components whose likelihood of failure is deemed to be so low that it may be discounted, ND's expectations based on the SAPs were set down during GDA Step 3 in RO-AP1000-19 and the associated Action RO-AP1000-19.A1 and A2. SAPs EMC1-3 are particularly relevant and are also listed in the table below.

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: the metal component or structure should be as defect-free as possible; 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.

141 SAP EMC.2 makes clear that the safety case should include a comprehensive examination of relevant scientific and technical issues, taking account of precedent when available. This is also emphasised by SAPs Paragraph 243 (Ref. 4) which states that *'Discounting gross failure of a component is an onerous route to constructing a safety case and there must be measures over and above normal practice that support and justify the claim'.*

142 Avoidance of failure by propagation of crack-like defects is based on a 'defence-in-depth' approach of:

1. defect tolerance confirmed by fracture analyses to determine limiting defect sizes and the absence of significant crack-like defects based on NDT examinations at the end of the manufacturing process; and

2. material toughness offering good resistance to propagation of crack-like defects - underpinned by minimum material toughness requirements in equipment specifications.

143 The basic logic of this approach is to underwrite the claim that the component enters service with either no crack-like defects or at least defects sufficiently small for there to be a substantial margin to the limiting defect size. This approach depends on a number of supporting strategies which are discussed in the three subsequent paragraphs.

144 **Limiting Defect Size Analyses:** All relevant materials are ductile and so the fracture analyses need to make use of elastic-plastic fracture mechanics methods. Limiting loading conditions need to be analysed using conservative materials properties which take account of uncertainties in the data as specified in SAP EMC.33 (Ref. 4). There also needs to be a realistic allowance for any potential crack growth in service.

145 **Materials Toughness:** There needs to be a basis for a conservative (lower bound) value of fracture toughness for end of life conditions as specified in SAP EMC.33 (Ref. 4). In some cases (e.g. shells of RPV, steam generators, pressuriser), this might be based on worldwide data, with minimum requirements in the component Equipment Specification to ensure the specific materials of manufacture are within the worldwide dataset.

146 **Manufacturing Inspections:** The concept is that manufacturing examinations be qualified to detect, with high confidence, defects of a size somewhat less than the size which could cause failure during service. The difference in size of defect that could cause failure and the size which can be detected with high confidence is referred to here as a defect size margin.

147 Westinghouse set down proposals for addressing RO Actions RO-AP1000-19-A1 and A2 in letter WEC00101N (Ref. 180). The plan was to:

- Identify which components would have an Incredibility of Failure (IOF) claim in the safety case.
- Specify the minimum toughness of the IOF components.
- Determine the size of defects of structural concern in the IOF components using the R6 approach.
- Use the ENIQ methodology to qualify the manufacturing inspection for the IOF components.

It should be noted that this letter uses the term "Incredibility of Failure (IOF)", which was current at the time it was written and refers to a claim that the likelihood of gross failure is deemed so low it can be discounted. As explained in Section 3.1 above, for the purposes of my assessment I have treated both HSS and HI welds as falling within the definition in the SAPs for Highest Integrity Components. .

148 Subsequently, on 5 March 2010, ND issued RO-AP1000-19.A3 (Ref. 10) to clarify its expectations. The action was:-

'*Westinghouse to execute a programme of work to establish a procedure for qualification of manufacturing examinations. The programme of work should include sufficient supporting analysis work (especially fracture mechanics analyses for limiting defect sizes) within GDA Step 4 to enable a judgement on the likelihood of success when fully implemented; full implementation being after the end of GDA Step 4.'*

The qualification of manufacturing inspections is expected to apply to components identified by work under RO-AP1000-18."

149 This action was clarified in the RO Action with the words *'The procedure for qualifying manufacturing inspections will be executed to completion beyond GDA Step 4. Within GDA Step 4, HSE-ND expects a sufficient scope of analysis work to be completed. This applies particularly to fracture mechanics analyses for limiting defect sizes. The work done within GDA Step 4 should be sufficient to enable a judgement to be made on whether the manufacturing inspections are likely to be capable of reliably detecting defects of concern (that is a fraction of the calculated limiting defect sizes); and so whether the qualification activities beyond GDA Step 4 are likely to be successful.'*

4.2.2 Overview of Position Reached at the end of GDA Step 4

150 The work conducted under RO-AP1000-19.A3 was a very substantial programme of work. The activities have not always been straightforward partly because the stress analysis data needed for the fracture assessments was not readily available in some cases, and partly because Westinghouse was not always familiar with ND's expectations in this area. In addition the process for addressing the qualification of the manufacturing examinations took longer than anticipated. As a result there were a number of significant delays in delivering the planned reports.

151 All the planned reports against RO action RO-AP1000-19.A3 have now been supplied. Westinghouse provided a number of drafts of their reports as the assessment process progressed in order to aid the understanding of their approach, however, the majority of the reports were not available in a final issued form until the very end of, or in many cases after, the agreed assessment period for GDA. Given this late delivery it was recognised that trying to fully assess these reports at the same time as producing the Step 4 report would represent an unacceptable risk to the delivery of the GDA Step 4 report. Consequently a strategy was developed to undertake a high level assessment of the reports in order to come to a judgement on whether it was likely that an adequate case could be made on the avoidance of fracture, and use this assessment as the basis for coming to a conclusion on whether to support an IDAC. A more detailed assessment will then be undertaken post Step 4 to confirm that an adequate justification had been made in order to come to a conclusion on a DAC. A GDA Issue has been raised to support this ongoing assessment work post Step 4, **GI-AP1000-SI-01**. The complete GDA Issue and associated action(s) are formally defined in Annex 2.

152 The following subsections (4.2.3 to 4.2.5) address my assessment of the three key aspects: the prediction of limiting defect sizes for crack-like defects; qualified non-destructive examinations during manufacture; and the derivation of material fracture toughness.

4.2.3 Extent of Work Programme on Fracture Mechanics and Qualified Manufacturing Inspections

153 As discussed in Section 4.1, Westinghouse has identified four components on the AP1000 where it needs to be shown that the likelihood of gross failure is so low that it can be discounted. Westinghouse refers to these as the HSS components (Ref. 16), but for the purposes of this assessment they will be judged against the criteria set out in ND's SAPs for the Highest Reliability Components.

154 The HSS components are (with specific parts as necessary):

- Reactor Pressure Vessel (RPV).
- Pressuriser (PZR).
- Steam Generator (SG) (channel head, tubesheet and secondary side shell).
- Main Steam Line (MSL) (inside containment).

155 In addition to the HSS components identified above, the Main Coolant Loop (MCL) piping welds between the MCL and the Reactor Vessel nozzle safe ends have been identified as HI welds. As discussed in Section 4.2.1, I have decided to assess these welds against the SAPs requirements for Highest Reliability Components.

156 ND accepted that it was not necessary to provide a justification for every weld or location on these components within the GDA timeframe, but a reasonable range of locations would need to be analysed on each component in order to come to a judgement on the acceptability of the design (Ref. 56). Westinghouse therefore developed a process to systematically rank welds in order to identify those which should be further analysed in terms of fracture mechanics calculations and technical justifications for the manufacturing inspections (Ref. 163).

157 The approach considered the likely defect tolerance of the weld using existing stress and fracture analyses, and ranked the inspectability of the weld using a group of NDT experts from the UK and USA. The combination of the defect tolerance ranking and inspectability ranking provided the overall ranking of the weld locations. This overall ranking, and a grouping of the welds based on component and location led to Ref. 163 identifying 12 welds where fracture mechanics calculations and an Inspection Plans (IP) would be needed to show that the design was acceptable. IPs are reduced scope technical justifications, these are discussed in more detail in Section 4.2.5 below. It was originally intended that IPs would be prepared for all these welds however it soon became apparent that there was significant overlap in the inspection requirements and thus it was agreed that completing IPs for the seven welds shown in the table below would be sufficient to give high confidence within GDA that adequate manufacturing inspections could be performed.

Weld Number	Component	Weld	Fracture Mechanics Assessment	NDT Inspection Plan
1	RPV	Lower Shell to Upper Shell	Yes	Yes
2		DVI Nozzle to Upper Shell	Yes	Yes
3		Inlet Nozzle to Safe-End (dissimilar metal weld)	Yes	Yes
4	SG	Lower Shell Barrel A to Tubesheet	Yes	Bounded by IPs for 1 &7
5		Main Feedwater Nozzle to Shell	Yes	Bounded by IP for 2
6		Inlet Nozzle to Safe-End (dissimilar metal weld)	Yes	Bounded by IPs for 3 &7

7	PZR	Upper Head to Upper Shell	Yes	Yes
8		Upper Shell to Middle Shell	Yes	Bounded by IP for 7
9		Manway to Shell	Yes	Yes
10		Surge Nozzle to Safe-End (dissimilar metal weld)	Yes	Yes
11	MSL	SG Main Steam Nozzle to Pipe	Yes	Yes
12	MCL	SG Inlet Nozzle Safe-End to Pipe	Yes	Bounded by IPs for 3 &10

158 I have sampled the approach and am generally satisfied with the methodology used for the ranking process and that it has been applied in a thorough and systematic manner. There is inevitably a degree of judgement in the process as the input data from the existing stress and fracture calculations is only a surrogate for the site specific calculations, the NDT ranking relies on expert judgement and there is an inevitable degree of judgement in the scoring and banding in the overall ranking process.

159 For example I did identify one potential anomaly. The SG Channel Head to Tubesheet weld is not included within the above table whereas Lower Shell Barrel A to Tubesheet weld is included. The thinner Lower Shell Barrel A to Tubesheet weld was considered bounding due to higher stress and fatigue levels, which is reasonable as both welds were given the same inspectability ranking. However, in practice the thicker section and clad nature of the Channel Head welds is more difficult to inspect than the thinner Lower Shell Barrel. This is reflected in the NDT ranking numbers of Table 9 in Ref. 163, but is not translated into the actual inspectability ranking due to necessarily coarse nature of the ranking boundaries as shown in Figure 8 of Ref. 163. Hence the decision is correct based on the process, but there could be a potential concern if the limiting defect sizes were small thus leading to detailed questions on the inspection qualification for the two regions. This is discussed further in Section 4.2.4.6.2.

160 Nevertheless I am generally satisfied that the methodology developed in Ref. 163 is a reasonable basis for identifying the most limiting locations in the design. I am therefore content that it has identified a generally representative and sufficient set of limiting locations for the purposes of providing a demonstration for GDA.

161 It should be noted that the dissimilar metal welds between the low alloy vessel nozzles and the austenitic stainless steel safe ends on the RPV, PZR and SG nozzles associated with the MCL pipework have been considered as HSS welds, and a representative weld from each of these vessels has been included with the 12 welds put forward for further detailed consideration. The weld ranking process assigned the dissimilar metal welds a relatively high ranking in terms of defect tolerance and defect detectability, so their inclusion on the basis of the weld ranking is understandable.

162 The MCL and Surge Line pipework welds are generally Class 1 although the welds to the RPV safe ends are classified as HI. However the dissimilar metal welds connecting the safe ends to the main vessels are all HSS. There is a question on whether the dissimilar metal welds should be considered as part of the vessel for the purposes of demonstrating that likelihood of gross failure in the vessel is so low that it can be discounted.

163 The decision has not been explicitly discussed, but I assume that Westinghouse took this decision to include the dissimilar metal welds within the HSS boundary because they

were unable to discount the possibility of the welds threatening the integrity of the vessel as Annex C of Ref.16 notes that the failure mode is a defect running into the RPV. The decision goes beyond the position previously adopted in the UK, and I am not aware of any residual concerns with the approach previously adopted in the UK. Hence I consider this to be a cautious decision by Westinghouse and will judge the evidence on that basis.

164 Note the position on the RPV nozzle to safe end dissimilar metal weld is made more complex by the piping welds between the RPV nozzle safe ends and the MCL having been identified as HI welds. Thus the welds beyond the nozzle safe ends are HI welds and Westinghouse has chosen to show that these welds are tolerant of defects. Hence the logic would then be to include the dissimilar metal weld in the RPV nozzles within the HI category and demonstrate defect tolerance in any case, and I will take this factor into account in judging the evidence.

4.2.4 Fracture Mechanics Analyses

4.2.4.1 Background

Fracture mechanics analyses have been provided for the twelve welds identified as being representative of the limiting welds in the components whose gross failure has been discounted. The analyses determine the limiting defect sizes for the welds and the through life fatigue crack growth from an initial a crack size that can be detected and sized with high confidence. The objective is to show a margin between the limiting defect size and the detectable defect size with an allowance for through life fatigue crack growth. In practice the dominant defect size parameter is usually the through-wall extent which is most significant both in terms of the limiting size for fracture mechanics and the size of defect which can be reliably detected and characterised.

165 The fracture mechanics analyses are one of the fundamental requirements identified in RO-AP1000-19 (Ref. 10), 'Avoidance of Fracture – Margins Based on Size of Crack Like Defect' that would be need to be addressed in order to show that the Highest Reliability Components are tolerant of defects. The need for the Safety Case to show that the Highest Reliability Components are defect tolerant is in line with EMC.1 of the SAPs for the Integrity of Metal Components (Ref. 4), and EMC1 is one of the three SAPs, EMC.1 to 3, which specifically apply to Highest Reliability Components over and above the normal integrity principles in order to be able to show that the likelihood of gross failure for these components is so low that it can be discounted.

166 Step 3 activities in this area were focussed on ensuring that Westinghouse was willing to propose suitable work packages to implement a method of achieving and demonstrating integrity consistent with UK practice, and Step 4 has focussed on ensuring that these work packages deliver the necessary assurance. Thus there are no matters to carry forward from Step 3.

167 The work is presented in a series of individual reports, a methodology report Ref. 52 and fracture assessments for the pressuriser, RPV and steam generators in Refs 53, 54 and 55 respectively.

168 Westinghouse has adopted the RO-AP1000-19 (Ref. 10) terminology in their fracture assessments. The limiting defect size is termed the End of Life Limiting Defect Size (ELLDS); the crack size that can be detected and sized with a high confidence is the Qualified Examination Defect Size (QEDS); and the through life fatigue crack growth the Lifetime Fatigue Crack Growth (LFCG). The margin between the ELLDS and the QEDS

plus LFCG is termed the Defect Size Margin (DSM), and Westinghouse has worked to a minimum DSM of 2.0 in line with approaches previously adopted in the UK.

Written as an equation: DSM = ELLDS / (QEDS + LFCG)

4.2.4.2 Extent of the Fracture Mechanics Analyses

4.2.4.2.1 Choice of Locations for Fracture Assessments

169 The fracture assessments have been undertaken at the twelve HSS/HI welds which Westinghouse considers the locations to be representative of the most onerous locations. As previously stated, whilst there will inevitably be a degree of judgement in this selection process, I am content that these are a generally representative and sufficient set of limiting locations for the purposes of providing a demonstration for GDA.

170 Westinghouse recognises that further work will be required post GDA to extend the scope of this programme and a wider range of weld locations will need to be considered by the fracture assessments during the licensing phase in order to confirm that the limiting locations have indeed been considered. This will be taken forward in Assessment Finding **AF-AP1000-SI-04.**

4.2.4.2.2 Parent Forgings

171 Westinghouse has focused on providing fracture assessments for defects at weld locations and have not provided fracture assessments at parent forging locations as part of GDA. This is on the basis that the welds are most likely to contain structurally significant defects and the assessment of the parent forgings will be bounded by the weld locations, see Paragraph 3.3.1.1 of Ref. 17, the Component Safety Report for the RPV.

172 In terms of the limiting defect sizes it is acknowledged that the parent forgings generally have a higher material toughness than the weld material and as they also do not contain residual stresses from the welding process, the parent material will have a larger limiting defect size at a given location than the weld. Thus in principle focussing on the welds will give the limiting case.

173 The exception to this is in vulnerable locations in the parent material, for example the nozzle crotch corners or RPV belt line regions.

174 Based on my previous experience of PWR fracture assessments I am content that the parent forgings are unlikely to be limiting in terms of the fracture assessments and limiting defect sizes do not need to be calculated during the GDA process. However, a selection of parent material fracture assessments will have to be undertaken during the Licensing Phase to confirm that these regions are not limiting.

175 This need for this work is recognised by Westinghouse, (for example Para. 3.3.1.1 of Ref. 17, the Component Safety Report for the RPV or Section 20.A.2.3.1.1 of the PCSR), and this is taken forward in Assessment Finding **AF-AP1000-SI-04.**

4.2.4.2.3 Fatigue Crack Growth

176 Westinghouse has undertaken fatigue crack growth calculations at all the identified locations to give a prediction of crack growth over the 60 year life of the plant from a postulated defect size that can be reliably detected and sized.

177 The calculations use existing transient definitions and cycle numbers and are useful in showing the likely fatigue crack growth over the life of the plant. This can then be taken into account when setting the qualified examination defect size (QEDS) for the inspection qualification.

4.2.4.2.4 Postulated Defect Description

178 The fracture assessments, Refs 53, 54 and 55 are based on semi-elliptical surface-breaking defects with a range of aspect ratios 2:1, 6:1 and 10:1, i.e. a 10mm deep defect with a 2:1 aspect ratio would have a length of 20mm whereas a 10:1 aspect ratio would have a length of 100mm. The postulated defects are orientated both parallel and transverse to the axis of the weld, and postulated at the most loaded position, for example the inner surface for a cold thermal shock and outer skin for a hot thermal shock. Crack tip loading at the deepest point and surface are considered.

179 The aspect ratios in the fracture assessments have been the subject of significant discussion and iteration. Westinghouse originally used an aspect ratio of 2:1 which was consistent with the aspect ratios used for the inspection qualification process, with a very limited consideration of a 10:1 aspect ratio for sensitivity purposes. The 2:1 aspect ratio was consistent with postulated defects orientated transverse to the weld typically assumed in previous defect tolerance demonstrations seen in the UK, and I am satisfied with that choice for defects orientated transverse to the weld. However, it is not consistent with the 10:1 aspect ratio typically assumed in for postulated defects orientated parallel to the weld to allow for a difficulty in the welding process leading to an extended defect.

180 As a consequence Westinghouse revised their fracture assessments to include a more extended 6:1 defect parallel to the weld, and retained the 2:1 aspect ratio transverse to the weld, but with some assessments still using the 10:1 aspect ratio if that had been considered in the sensitivity study. There is an element of judgement in setting this ratio, and whilst I have not undertaken a review of the likelihood of a weld in a nuclear pressure component containing defects with a length greater than six or ten times its depth, I am aware that aspect ratios of 6:1 are commonly assumed in nuclear pressure vessel design codes.

181 The aspect ratio has an effect on the crack tip loading, and a larger aspect ratio increases the applied stress intensity factor for a given defect depth at both the ends of the crack tip on the surface and at the deepest point. Thus, whilst setting the aspect ratio of the postulated defect is considered in terms of the likelihood of the welding processes leading to defects with a particular aspect ratio, it is also an integral part of the margins within the overall demonstration of fracture, be they margins embedded within the methodology or explicit margins such as the target DSM. As such it is difficult to consider the choice of aspect ratio in isolation from the margins.

182 I am content for Westinghouse to have used an aspect ratio of 6:1 for defects orientated parallel to the weld axis for the purposes of GDA. on the basis that this aspect ratio is in common use in nuclear pressure vessel design codes and GDA is concerned with gaining confidence in the design. However, the more extensive fracture assessments which will be undertaken post GDA to support the Pre-operation Safety Report (POSR) will need to consider the effect of using a 10:1 aspect ratio compared with a 6:1 aspect ratio. The assessments will need to show that a 10:1 aspect ratio defect would not lead to a unacceptably large reduction in the DSM in the overall demonstration of fracture i.e.

to show that there is no 'cliff edge' effect in using a 10:1 aspect ratio. This is taken forward as Assessment Finding **AF-AP1000-SI-05**.

183 The final fracture assessments, Refs 53, 54 and 55 reflect the revised aspect ratios, but it should be noted that the fracture methodology report, Ref. 52, has not been updated to reflect this change. I do not consider this to be material to the overall case presented for GDA.

184 It should be noted that the shorter aspect ratios are still used for the inspection qualification process. For a defect depth predicted by fracture analysis with a 6:1 aspect ratio, the use of this depth but an aspect ratio of 2:1 will be conservative for inspection qualification and I am content with the use of the shorter aspect ratio in this context.

4.2.4.2.5 Conclusions and Findings Relating to Extent of Fracture Mechanics Analyses

185 Westinghouse has undertaken a series of fracture mechanics analyses and technical justifications for the manufacturing inspections for the twelve welds identified as being representative of the limiting welds of the components whose gross failure has been discounted. I am satisfied that a representative set of limiting weld locations have been defined for the purposes of GDA.

186 There are two assessment findings which should both be completed before the generic milestone Install RPV. This is because it would be extremely difficult to make substantive changes once the components start to be installed which could then lead to substantial delays and additional costs. In practice the findings will need to be completed earlier to match the programme for demonstrating avoidance of fracture.

> ***AF-AP1000-SI-04:*** *The Licensee shall undertake fracture assessments on a wider range of weld locations on the HSS Components in order to demonstrate that the limiting locations have been assessed. The Licensee shall also undertake fracture assessments on the vulnerable areas of the parent forgings in order to demonstrate that the limiting locations have been assessed.*
>
> ***AF-AP1000-SI-05:*** *The Licensee shall undertake fracture assessments to show that a postulated defect with a 10:1 aspect ratio defect would not lead to an unacceptably large reduction in the Defect Size Margin (DSM) in the overall demonstration of fracture i.e. the Licensee shall demonstrate that a 10:1 aspect ratio would not lead to a 'cliff edge' effect on the DSM.*

4.2.4.3 Loading Conditions

187 The fracture assessments take into account thermal hydraulic loading, external mechanical loading and weld residual stress. The load cases are described in the individual fracture assessments (Refs 53, 54 and 55).

4.2.4.3.1 Thermal Hydraulic Loading

188 Westinghouse proprietary software is used identify the limiting load case/time step combinations through a range of crack depths from 10% to 90% of wall thickness, on the basis of the crack tip stress intensity factor (Ref. 52). All transients are considered, and the analysis takes account of the direct thermal hydraulic loads and the external mechanical loading from pipework and supports.

189 Once the limiting load/time step combinations have been identified the loading conditions are extracted, weld residual stress and crack face pressure stress introduced and the combined load case assessed using the R6 defect assessment procedure (Ref. 50).

190 The identification of the limiting load case/time step combination is essentially a screening process as it is based on the linear elastic stress intensity factor rather than a two parameter elastic-plastic fracture mechanics approach taking into account plastic collapse. It also does not account for the residual stresses. The final calculation of the limiting defect size takes account of the complete load combination, but the question is whether the screening process is able to identify the limiting transient combination and the limiting time step for that transient/crack depth.

191 I am satisfied that the screening process should identify the limiting transient, but consideration of the elastic stress intensity factor alone is not a rigorous approach to determine the limiting timestep for a two parameter elastic-plastic fracture mechanics approach. I therefore asked a Technical Support Contractor to undertake a comparative study to determine whether the limiting time step had been correctly identified.

192 The comparative study is discussed under Section 4.2.4.6.3 and some differences have been indentified. This work arrived too late in the assessment process to be fully considered within this Step 4 report. I am satisfied that these differences should not be material to the overall conclusions on limiting defect size for the purposes of moving to an IDAC, but I will need to resolve the differences before I would be prepared to support a DAC, and will address these through GDA Issue **GI-AP1000-SI-01.**

193 However, irrespective of these differences it is clear that a more sophisticated approach will be required to indentify the limiting time step for the transient/crack depth combinations for the more extensive fracture assessments that will be undertaken post-GDA against Assessment Finding **AF-AP1000-SI-04**. The Licensee will have to develop such an approach, and this results in Assessment Finding **AF-AP1000-SI-06**.

194 The fatigue crack growth from the thermal and pressure transients has also been assessed using Westinghouse proprietary software taking into account the direct thermal hydraulic loads and the external mechanical loading from pipework and supports in a similar manner (Ref. 52). The bounding stress intensity factors are then calculated for a range crack depths, and these are then used for the fatigue crack growth analysis. I was unable to sample the method in detail, but in principle I am satisfied with the approach that has been taken.

195 The limiting thermal hydraulic cases are generally the most severe Category D transients (faulted conditions), although there are some situations where the where it has been a Category C or indeed Category A/B transient that has been limiting.

196 In general I am satisfied that an appropriate range of transients and transient definitions have been considered in the fracture assessments. However, it is of note that the RPV fracture assessment (Ref. 54) did not include a consideration of the Level D thermal transients. I questioned why this was the case, and Westinghouse responded in Ref. 57 that they believed that the Core Makeup Tank Injection Test, the limiting Level A/B transient, would envelope the most severe Level D transients including the Large LOCA and Large Feedwater Break Transients.

197 This is unexpected and I was unable to confirm Westinghouse's assertion as the RPV fracture assessment (Ref. 54), and the supporting evidence in Ref. 57 arrived too late to be assessed in detail for the GDA Step 4 assessment report (explained further at Section 4.2.4.7). The detailed assessment of this evidence and the fracture assessments will

therefore occur post Step 4 under GDA Issue **GI-AP1000-SI-01** and as part this work a more detailed review of this transient will be undertaken involving the ND Fault Studies team.

4.2.4.3.2 External Mechanical Loading

198 The mechanical loading includes loadings from studs, deadweight, thermal expansion, etc. The approach to applying the mechanical loads is generally as expected but the fracture assessment reports arrived too late to be assessed in detail.

199 A more detailed review of these loads will be undertaken as part of the detailed assessment of the fracture assessments post Step 4 under GDA Issue **GI-AP1000-SI-01.**

4.2.4.3.3 Residual Stress

200 The fracture analysis methodology report (Ref. 52) includes the residual stress assumptions that have been made for each of the different weld types as discussed below.

Low Alloy Steel Welds for the RPV, PZR and SG

201 The low alloy welds are stress relieved, and weld residual stress recommendations from Ref. 51 were originally used in the fracture assessments equating to 170 MPa for longitudinal flaws in a circumferential weld and 113 MPa for a circumferential flaw in a circumferential weld (Ref. 53). Westinghouse found these levels of residual stress caused difficulties with some of the fracture assessments, and then adopted what they considered to be a more realistic uniform residual tensile stress of 55 MPa. This value was based on a historical recommendation taken from Table II.7.1 of the R6 Procedure for the Assessment of the Integrity of Structures Containing Defects (Ref. 50). The value of 55 MPa has been in general use in the UK for fracture assessments of this type, and I am satisfied with the use of a residual stress of 55MPa or greater.

Stainless Steel Welds in the Main Coolant Loop Pipework

202 The stainless steel welds in the main coolant loop pipework are not stress relieved, and residual stress recommendations from Ref. 51 have been assumed. Two distributions are provided; one for axial flaws and one for circumferential flaws. Both have yield magnitude residual stresses on the outer surface of approximately 148 MPa, reducing towards the inner surface with a residual stress of 0 MPa in the case of the axial flaws and 30 MPa in the case of circumferential flaws (Figure 4-1 of Ref. 55). I am satisfied with the distributions as the values have been taken from a well established source.

Dissimilar Metal Welds in the Main Coolant Loop Pipework

203 The residual stress distribution in dissimilar meld welds is complex due to the differing thermal expansivities of the materials. The welds themselves are not stress relieved. Ref. 52 provides separate residual stress profiles for the nickel-based dissimilar metal welds at both ambient temperature and normal operating temperature for both the MCL

pipework and the pressuriser surge line. Profiles are provided in both the hoop and axial directions.

204 The distributions are based on information from a variety of sources which include both analytical work and residual stress measurements on representative mock-ups. The distributions for the RPV Nozzle to Safe End dissimilar metal weld have been subject to review through a Technical Support Contract (TSC) looking at the overall fracture assessment of the dissimilar metal weld (Ref. 58), but the RPV fracture assessment report (Ref. 54) arrived too late for the results from the TSC work to be fully considered within the GDA Step 4 assessment report (explained further at Section 4.2.4.7).

Main Steam Line Welds

205 The welds in the ferritic steel pipework of the MSL are stress relieved, and a uniform residual stress of 60 MPa has been used based on a historical recommendation taken from Table II.7.1 of the R6 Procedure for the Assessment of the Integrity of Structures Containing Defects (Ref. 50). The value of 60 MPa has been in general use in the UK for fracture assessments of this type, and again I am satisfied with the use of a residual stress of 60 MPa.

4.2.4.3.4 Conclusions and Findings Relating to Loading Conditions and Residual Stresses

206 The approaches used by Westinghouse to derive loading conditions and residual stresses are generally acceptable, however the Westinghouse fracture assessment reports arrived too late in the assessment process to be fully assessed during Step 4 from the perspective of the loading conditions.

207 This detailed assessment will occur post Step 4 through GDA Issue **GI-AP1000-SI-01** (see Section 4.2.4.7), and will consider the:

- Identification of the limiting time step from the thermal hydraulic loading using the linear elastic stress intensity factor rather than a two parameter elastic-plastic fracture mechanics approach.
- The justification for not considering Level D thermal transients in the RPV fracture assessment.
- A review of the external mechanical loading applied in the fracture assessments.

208 In addition the more extensive fracture assessments that will need to be undertaken post GDA against Assessment Finding **AF-AP1000-SI-04** will require a more sophisticated approach to identifying the limiting time step for the transient/crack depth combinations, and this has resulted in Assessment Finding **AF-AP1000-SI-06**

> ***AF-AP1000-SI-06**: The Licensee shall use a robust methodology for indentifying the limiting time steps for use in the more extensive fracture assessments that will be undertaken post GDA.*

209 This should be completed before the Generic Milestone for Installation of the RPV. This is because it would be extremely difficult to make substantive changes once the components start to be installed which could then lead to substantial delays and additional costs.

4.2.4.4 Fracture Assessment Methodology

210 Westinghouse has used the R6 Procedure for the Assessment of the Integrity of Structures Containing Defects, Ref. 50 to calculate the limiting defect sizes for the welds. The R6 Procedure was originally developed by the Central Electricity Generating Board (CEGB) in the UK, and is currently at Revision 4.

211 The R6 Procedure has been used in UK based nuclear safety cases for many years, and an independent review by the Advisory Committee on the Safety of Nuclear Installations (Ref. 62) concluded that it was a soundly based approach with extensive validation. I am therefore satisfied that it is an appropriate methodology for calculating the limiting defect sizes in the welds.

212 Westinghouse has chosen to use the software based implementation of the R6 Procedure, R-Code (Ref. 63) to undertake the limiting defect size calculations. The stress distributions used in the assessments are taken from the existing elastic finite element stress analyses of the components, and resolved into primary and secondary loading. The residual stresses are then added as an additional secondary load set. Recognised stress intensity factor solutions and plastic collapse solutions from R-Code are then used to undertake the limiting defect size calculation. All the fracture assessments are based on this standard approach. I am generally content with such an approach and the use of R-Code to implement the R6 Procedure.

213 The fatigue crack growth has been calculated using Paris Law crack growth equations from ASME XI, Ref. 66, with transients applied in sequence based on the total number of transients specified for the 60 year design life. This is a standard approach and I am satisfied with the method used.

214 As discussed in Section 4.2.4.3.1 on the thermal transients, the R6 Procedure to determine the limiting defect size has only been applied at what is believed to be the limiting time step from a consideration of linear elastic stress intensity factor alone. This is not a rigorous approach, and a more sophisticated approach will be required to identify the limiting time steps for the post GDA fracture assessments, which is addressed by Assessment Finding **AF-AP1000-SI-06.**

4.2.4.5 Material Properties

215 The lower bound material properties used in the fracture assessments are presented in Ref. 52. The material properties are considered further in Section 4.2.6, and I am satisfied that Ref. 52 provides a suitable basis for calculating the limiting defect sizes for GDA.

216 It is notable that all the fracture assessments are based on initiation toughness, and this includes the more severe, faulted and accident, Level C/D transients. Previous fracture assessments seen in the UK have invoked ductile tearing for these more severe transients and SAP EMC.34 (Ref. 4) allows for a limited degree of stable tearing to be invoked for the severe faulted and accident transients providing there is valid fracture toughness data available. The approach taken by Westinghouse could therefore be seen as introducing an additional degree of conservatism into some of the results. However, a number of the limiting cases are driven by the normal operation Level A/B transients for which it would not have been acceptable to invoke ductile tearing and hence there is no additional conservatism in such cases.

4.2.4.6 Fracture Assessment Results

217 The following table summarises the results from the fracture assessments.

Component	HSS Weld	QEDS (mm)	QEDS + LFCG (mm)	ELLDS (mm)	Wall Thickness (mm)	Ref.
RPV	Lower Shell to Upper Shell	12.5	21.1	48.6	216.3	54
	DVI Nozzle to Upper Shell	12.5	21.9	57.3	286.3	54
	Inlet Nozzle to Safe-End (dissimilar metal weld)	6.0	6.8	15.0	59.4	54
SG	Lower Shell Barrel A to Tubesheet	5.4	33.69	67.56	96	55
	Main Feedwater Nozzle to Shell	15.0	34.45	95.35	122	55
	Inlet Nozzle to Safe-End (dissimilar metal weld)	10.0	13.89	43.43	113	55
PZR	Upper Head to Upper Shell	11.75	14.40	28.83	64	53
	Upper Shell to Middle Shell	15.0	17.87	43.43	114	53
	Manway to Shell	11.75	-	62	114	53
	Surge Nozzle to Safe-End (dissimilar metal weld)	5.0	5.0	12.09	55.4	53
MSL	SG Main Steam Nozzle to Pipe	7.5	7.52	14.82	49	55
MCL	SG Inlet Nozzle Safe-End to Pipe	6.9	8.64	17.30	88	55

218 All of the above are based on an aspect ratio of 6:1, apart from the PZR main shell welds which are based on an aspect ratio of 10:1. Values quoted are applicable to the minimum defect size margin quoted for each weld.

4.2.4.6.1 Assessment of Results

219 The fracture assessment reports, Refs 53, 54 and 55, arrived very late in the GDA Step 4 assessment process, and were much later than had originally been envisaged. Therefore, whilst Westinghouse has submitted all the planned fracture assessment reports within GDA, it has not been possible for ND to undertake a full assessment of these reports within the timescales allowed for GDA Step 4.

220 Westinghouse did, however, submit earlier drafts of the fracture assessment reports which has allowed me to gain an understanding of their approach and to enable me to commission comparative studies by a EASL. I therefore considered it reasonable to undertake a high level review of the submitted reports in conjunction with the understanding of their approach and the results for the comparative studies to come to a view on whether the defects sizes can be used as the basis for the overall Avoidance of Fracture demonstration in terms of an IDAC.

221 The next section summarises my assessment of whether the defect sizes can be used for the purposes of supporting an IDAC. However I will need to undertake a more detailed assessment of the fracture assessment reports in order to confirm that I am satisfied with the results in terms of supporting a DAC and this is covered under GDA Issue **GI-AP1000-SI-01.**

4.2.4.6.2 High Level Review

222 In general I am satisfied with the work presented in the fracture assessments but my high level review has identified a number of points of note that have needed to be taken into account in my judgement.

SG Lower Shell Barrel A to Tubesheet weld

223 The through-life fatigue crack growth predicted for the SG Lower Shell Barrel A to Tubesheet weld is large. This leads to a relatively small QEDS defect depth of 5.4mm which has been taken through to the inspection qualification process for the SG Lower Shell Barrel A to Tubesheet weld. As stated in Section 4.2.3, this weld was analysed on the basis that it would bound the SG Channel Head to Tubesheet weld. However, since the SG Channel Head to Tubesheet weld is thicker as well as being clad internally, inspection is more difficult. If this weld were to have a similarly small QEDS value, it might cause difficulty for the inspection qualification process and cause this weld to be limiting for avoidance of fracture rather than the SG Lower Shell Barrel A to Tubesheet weld currently assumed.

224 This therefore raises the question as to whether a separate fracture assessment and inspection qualification process is now required for the SG Channel Head to Tubesheet weld for the purposes of the GDA demonstration.

225 The small QEDS value is driven by the high level of fatigue crack growth. This may be due to overly conservative transient definitions, but this is not known. Looking in detail at Tables 4 and 5 of the Weld Ranking report (Ref. 163), you find that the ASME III fatigue usage factor ‘U’ is around three times higher on the SG Lower Shell Barrel A to Tubesheet compared with the SG Channel Head to Tubesheet weld. Although there is not a direct correlation between the fatigue usage factor and the amount of fatigue crack growth, it is reasonable to use this as an indicator that the fatigue crack growth rates will be less severe on the Channel Head side of the Tubesheet. I am therefore satisfied that the QEDS for the SG Channel Head to Tubesheet weld will be significantly larger than the 5.4mm deep QEDS required for the SG Lower Shell Barrel A to Tubesheet. I am therefore content to judge for the purposes of supporting an IDAC that it is unlikely that the SG Channel Head to Tubesheet weld would be limiting in terms of the overall demonstration of avoidance of fracture, but will progress the matter further during the detailed assessment of the reports under GDA Issue **GI-AP1000-SI-01** before accepting this for a DAC.

MCL Weld

226 The limiting defect size calculated for the MCL is based on an allowable fracture toughness of 286MPa√m, which is much higher that than the 182MPa√m recommended in the Methodology Report (Ref. 52). Westinghouse argues that the higher value is compatible with the gas tungsten arc welds used on the MCL. However, the argument is

caveated by a statement that the welds are ‘usually’ performed using this process, which raises the obvious question as to whether there are circumstances in which the welding process would differ (Section 4.6.6.2 of Ref. 53). The 286MPa√m is a relatively high toughness when set in the context of initiation toughness for austenitic stainless steel, and is more representative of the values that are normally assumed after ductile tearing has been invoked. However, as the limiting defect size (ELLDS) is calculated for the Large Steam Line Break, which is a Level D transient (Table 5-6 of Ref. 55), it would be permissible to invoke ductile tearing. I therefore judge that it is unlikely that the net effect will undermine the overall demonstration for the MCL, but I will need to consider this aspect further during the detailed assessment of the reports.

227 The MCL fracture assessment is included to underpin the HI claim made for the MCL piping welds between the MCL and the RPV nozzle safe ends. However, the fracture assessment is provided for the piping weld between the MCL and the SG Primary Inlet nozzle safe end. This is a function of the weld ranking process identifying that the MCL loading combination is most onerous at the SG Inlet Nozzle.

228 I have not looked in detail at the loading sets, but I am satisfied that Westinghouse has gone through a rigorous process in identifying the limiting areas. Thus I accept that undertaking the MCL weld assessment at the SG inlet nozzle is an adequate and conservative surrogate for an assessment of the MCL welds on the RPV nozzles.

RPV Lower to Upper Shell Weld

229 The assessment of the RPV Lower to Upper Shell weld has been undertaken using transient stresses from a section called ‘ASN12’ located in the Upper Shell at the upper shell geometry transition. This location is ‘several inches’ higher up in the RPV than the lower to upper shell weld, but taking the transient stresses from this location is considered to be conservative by Westinghouse due to the bending stresses induced by the thickness change and high transient stress as the section used is directly below the DVI nozzle (Section 4.5.1, Ref. 54). I understand that this location was chosen due to the availability of stress information, and I am satisfied that the location should be conservative in terms of the applied stresses.

230 However, I note that the effect of irradiation embrittlement on the lower to upper shell location has not been addressed, and Westinghouse has identified this as an open item which will be addressed in the next revision of the document, Section 2.3 of Ref. 54.

231 The GDA fracture assessments have focussed on the welds, and I am generally content with this approach. However, I recognised that as the belt line region of the RPV is forged as a single piece, there will be a need to check that the fracture assessment of the welds surrounding the belt line forging bounds the forging itself at the location of greatest irradiation embrittlement. I therefore raised TQ-AP1000-682 (Ref. 9) for Westinghouse to explain how the fracture analysis work on the welds above and below the belt line forging would be bounding for the belt line forging. The response explains that the preliminary end-of-life nil ductility temperature predictions for a weld in the beltline region would be higher than for the belt line forging material, and the fracture assessments of the welds closest to the beltline region of the RPV would therefore use end-of-life fracture toughness properties for the weld material.

232 Thus despite the re-assurance provided by the response to TQ-AP1000-682 (Ref. 9), the limiting defect size depth of 48.6mm calculated for the RPV Lower to Upper Shell Weld is potentially non-conservative as it has been based on start-of-life fracture toughness

properties, and there is no evidence to show that a defect in the weld would be bounding for the belt line forging.

233 The failure to address the effect of irradiation embrittlement on this weld is an important omission. I am unable to reliably determine the extent of any non-conservatism on the calculated defect sizes without a re-assessment of fracture assessment for the weld using end-of-life properties. I also recognise that the transient loadings which have been applied to this weld are conservative for the location of the weld, and there would be scope to remove this conservatism if necessary, thus making it very difficult to judge the final effect on the defect sizes.

234 I will address the effect of irradiation embrittlement on this weld as part of the detailed assessment of the fracture reports; however, I judge that the final differences in the defect sizes are unlikely to be so large so as to invalidate the case being presented. I therefore believe it is reasonable to use the defect sizes quoted to date for the RPV Lower to Upper Shell weld in the overall avoidance of fracture justification for the purposes of supporting an IDAC, but with the proviso that these will need to be confirmed during the detailed assessment of Ref. 54.

4.2.4.6.3 TSC Comparative Studies and Review Work

235 I commissioned EASL to undertake comparative studies on the basic fracture assessments undertaken by Westinghouse, and a review of the assessment of a dissimilar metal weld. The comparative studies were designed to test Westinghouse's approach to determining the limiting time step for the transient and their application of the R6 Procedure. The review work was intended to provide an overview of the approach taken by Westinghouse to the assessment of a dissimilar metal weld.

Comparative Studies

236 The comparative studies were undertaken on the main shell welds on the Pressuriser and RPV, Refs 59 and 61. For expediency the EASL work had to be undertaken on either draft or non-final versions of the Westinghouse work. This led to an apparent discrepancy in the PZR comparative study that was not borne out in practice by the final Westinghouse report and the EASL work was therefore supplemented by an internal file note, Ref. 60.

237 A summary of the comparative study results is shown below:

Location and Load Case	WEC limiting defect depth	Timestep	EASL limiting defect depth	Timestep
Pressuriser Head to Shell Weld Inner surface circumferential defect 6:1 aspect ratio Load Case - Small Steam Line Break	41.0 mm	1610 secs	42.8 mm	1610 secs
RPV Lower Shell to Upper Shell Weld Inner surface circumferential defect 2:1 aspect ratio	104.5mm	2550 secs	148.0mm	2550 secs

Location and Load Case	WEC limiting defect depth	Timestep	EASL limiting defect depth	Timestep
Load Case – Core Make Up tank high pressure injection and heated drain down test				
RPV Lower Shell to Upper Shell Weld As above, but with the limiting time-step established by the EASL.			60.8mm	2028 secs

238 As can be seen the comparative study on the Pressuriser Head to Shell weld showed good agreement with the Westinghouse work in terms of the limiting defect size and the identification of the limiting timestep.

239 However, the work on the RPV Lower Shell to Upper Shell Weld indicates apparent discrepancies in both the limiting defect sizes and the identification of the limiting timestep. The EASL work indicates that the Westinghouse assessment could be conservative based on the same limiting timestep, and EASL suggests that the differences may be down to a limited range of plastic collapse solutions being available in R-Code, Ref. 63 which do not allow for the bending component in the primary load. (R-Code is the software implementation of the R6 Procedure, Ref. 50, and includes a range of stress intensity factors and plastic collapse solutions that are not connected to the R6 Procedure itself, and do not necessarily cover all situations.)

240 Of more concern is that EASL has identified a different limiting timestep, and that the limiting defect size at that timestep is significantly smaller than the limiting defect size calculated by Westinghouse. This could indicate a problem in indentifying the limiting time step from a consideration of linear elastic stress intensity factor alone as discussed in Section 4.2.4.3.1.

241 Taking the EASL results at face value, the Westinghouse limiting defect sizes could be underestimated by 40%. If this were found to be the case, and similar underestimates were found in other locations, then it would have an effect on the overall avoidance of fracture justification. It would not necessarily preclude a justification, but further work would be required. However, the earlier work undertaken by EASL indicates that the difference does not apply in all cases, and that there can be good agreement between the results in other areas. It is therefore important to fully understand and reconcile the differences shown in the RPV assessment.

242 The EASL report on the RPV arrived later than originally planned. This was because the Westinghouse reports and information needed to undertake the comparative study arrived much later than anticipated. It was therefore not possible to reconcile the EASL results and the Westinghouse results within the GDA Step 4 timeframe.

243 Thus at this point in time there is some uncertainty about the limiting defect sizes calculated by the Westinghouse fracture assessments. I do not believe that the discrepancies are likely to be so significant that they would ultimately undermine an avoidance of fracture justification, and I am therefore satisfied that the approach can be used in the overall avoidance of fracture justification for the purposes of supporting an IDAC. However, the results will have to be reconciled during the detailed assessment of

the Westinghouse fracture assessments before I would be prepared to support a DAC based on the Westinghouse approach. This will be addressed as part of GDA Issue **GI-AP1000-SI-01**.

Dissimilar Metal Weld Review

244 The EASL work is presented in Ref. 58, and the review was undertaken on the nozzle to safe end dissimilar metal weld on the RPV. The objective was to review the approach taken by Westinghouse rather than undertake a comparative study.

245 EASL's report identifies a number of areas where clarification is required and a number of areas of concern with regard to the assessment approach. Unfortunately the EASL report arrived later than originally planned because the Westinghouse report needed to undertake the review arrived much later than anticipated. As a consequence it has not been possible to obtain the necessary clarification and understanding on the assessment approach to come to a final conclusion on the approach taken by Westinghouse. Overall the EASL report does question the validity of the Westinghouse assessment, but I consider that it is premature to take such a view without obtaining further clarification from Westinghouse on the rationale for some aspects of their approach.

246 As stated in Section 4.2.3, I consider the assessment of the dissimilar metal welds on the nozzles as part of the HSS welds to be a cautious decision by Westinghouse, and such approaches go beyond the approach previously adopted in the UK where the attached pipework is not in the HSS/HI category. Thus any concerns with regard to the approach need to be considered in this context. I do acknowledge that there is a difference for the dissimilar metal welds in the RPV nozzles as the attached pipework is considered to be a HI weld. However, I recognise that Westinghouse may decide to develop a consequences case for this weld and thus remove it from the HI category in any case (comment in Table B7 of Ref. 16). Thus the overall significance of the case for the dissimilar metal welds is not as high as the other HSS welds, and I therefore consider that it is reasonable to accept the Westinghouse approach from the perspective of supporting an IDAC with the uncertainties identified by EASL. However, assuming that Westinghouse retains these welds in the HSS category, then the EASL comments will have to be reconciled during the detailed assessment of the Westinghouse fracture assessments before I would be prepared to support a DAC based on the Westinghouse approach. This will be addressed as part of GDA Issue **GI-AP1000-SI-01**.

4.2.4.7 Overall Conclusion on the Fracture Assessment Results

247 The fracture mechanics analyses calculate limiting defect sizes for the welds using the R6 Procedure (Ref. 50) and undertake through life fatigue crack growth calculations from an initial crack size that can be detected and sized with high confidence. The approach is consistent with those previously adopted by Licensees in the UK.

248 As previously stated, the Westinghouse fracture assessment reports arrived too late to allow ND to undertake a full assessment of the reports within the timescales allowed for GDA Step 4. I have therefore undertaken a high level review of the fracture assessment reports and commissioned comparative studies from EASL in order to come to a view on whether the defects sizes can be used as the basis for the overall Avoidance of Fracture demonstration in terms of an IDAC.

249 The high level review and the EASL studies have identified a number of important matters that will need to be considered further, but overall I have confidence in the approach being taken by Westinghouse and judge that it is unlikely that these matters would undermine the overall Avoidance of Fracture demonstration. I therefore conclude that it is reasonable to assume that the limiting defect sizes calculated by Westinghouse can be used as the basis for the overall Avoidance of Fracture demonstration in terms of an IDAC.

250 However, I will need to resolve these matters and undertake a more detailed assessment of the fracture assessment reports in order to confirm that I am satisfied with these limiting defect sizes in terms of a DAC. I have therefore raised Action 1 of GDA Issue **GI-AP1000-SI-01** for Westinghouse to support the ongoing assessment of the fracture assessment reports post GDA Step 4 in order to confirm that I am satisfied that these limiting defect sizes can be used in the Avoidance of Fracture demonstration for a DAC.

*GDA Issue Action **GI-AP1000-SI-01.A1:** Westinghouse to support the assessment of the fracture assessment reports post GDA Step 4.*

The main activity shall involve making adequate responses to questions arising from ND assessment of documents submitted during GDA Step 4 or in response to this Action.

251 The complete GDA Issue and associated action(s) are formally defined in Annex 2

4.2.5 Qualified Non-Destructive Examinations During Manufacture

252 Westinghouse has agreed that the manufacturing inspections for the HSS and HI components will be qualified in accordance with the ENIQ methodology (Ref. 112). The size of the qualification defect must be smaller than the limiting defect size determined through fracture mechanics and fatigue analysis by a significant margin, which is currently intended to be a factor of two.

253 The main requirements of the ENIQ methodology are:-

- A Technical Justification (TJ) is prepared which provides evidence from trials, inspection and modelling to support the claimed capability.
- A group of independent experts, known as the Qualification Body (QB), review the TJ and identify, if considered necessary, blind trials to confirm the predicted capability. Alternatively they may reject the TJ or ask for revisions at this stage.
- Once the TJ is considered acceptable and the blind trials are successfully completed the Qualification Body approves the procedure for use.

254 The process Westinghouse adopted for GDA was a streamlined version of this using an expert from SERCO's Inspection Validation Centre (IVC) as a quasi Qualification Body. The TJs, which for this purpose were called Inspection Plans (IP), gave a full description of the inspections but provided less supporting evidence for the claimed capability than would normally be expected. Seven IPs were produced which covered a range of representative welds for the main vessels as discussed in Section 4.2.3.

255 All the inspection plans were reviewed by the IVC and revised as required and when they were satisfied they provided a statement that, in their judgement, the proposed procedure would be capable of being qualified. This review did not include physical work or trials. The IPs do not include a full review of all the parameters which could affect the inspection

nor do they provide advice to the QB on test pieces for open and blind trials. However they do contain a review of the more important variables which influence the inspection and they provide detailed evidence on the likely capability. Consequently they are an adequate basis for our assessment of whether, when fully developed, the inspections are likely to be capable of successful qualification.

4.2.5.1 Inspection Plan for RPV Upper Shell - Lower Shell Weld

4.2.5.1.1 Overview of RPV Upper Shell - Lower Shell Weld Inspection Plan

256 The IP for the ultrasonic inspection of the RPV lower shell to upper shell weld is set out in Ref. 164. This is a substantial document which also describes the inspection plan for the RPV DVI nozzle to shell weld which I discuss in the next section.

257 This weld joins the cylindrical forging which forms the main body of the RPV (lower shell) to the perforated ring forging (upper shell) which contains the nozzles. Both shell forgings are made of SA-508 Grade 3 Class 1 ferritic steel and the weld is a narrow gap, submerged arc weld with ferritic steel filler. The forgings have an inside diameter of 4038 mm and are 213 mm thick. Prior to the qualified inspection the weld will have undergone post weld heat treatment and the inside surface of the vessel will have been clad with austenitic stainless steel with a thickness of approximately 5.6 mm.

258 This IP was prepared in advance of the fracture mechanics assessment and an estimated Qualification Examination Defect Size (QEDS) of 25 mm x 50 mm was used, the reconciliation when the actual QEDS was determined is discussed in Section 4.2.5.4 below. Four types of planar defect were assumed; lack of sidewall fusion, interbead lack of fusion, parallel cracking and transverse cracking. For each type of defect the morphology (rough or smooth), the potential range of tilt and skew and the location relative to the weld were defined.

259 The object of the inspection was to *'detect, characterise and size planar manufacturing flaw indications within the inspection volume'*. The specific requirements are to achieve highly reliable detection and classification (planar or volumetric) of defects larger than the QEDS and to measure the through wall size with an accuracy of ± 5 mm. The defect length is to be measured with an accuracy of ± 20 mm for surface-breaking defects and with a slightly lower accuracy for embedded defects.

260 For this inspection the ultrasonic operator will be a certified Level II or Level III inspector who will have been further qualified for this inspection by blind trials on suitable qualification test pieces and by other means if required by the Qualification Body.

261 The proposed automated inspections will be from both the internal clad surface and the external unclad surface and are claimed to be based on the Sizewell B manufacturing inspections and those in successful industry test trials. They will use 0°, 45°, 60° and 70° pulse-echo probes from both surfaces supplemented by 50° pulse-echo and 45° and 50° tandem from the internal surface and a 40° pulse-echo probe from the external surface. In addition Time of Flight Diffraction (TOFD) will be used to aid sizing and characterisation of defects.

262 As required by ENIQ (Ref. 112) the IP reviews the influential parameters (those that could affect the inspection results) and identifies which of those are essential parameters, i.e. those which are able to significantly influence the results of the inspection.

263 Under the ENIQ guidance “Physical Reasoning” would be used to justify why each of the essential parameters selected for the inspection were appropriate to meet the aims of the

inspection. Because the IP was not a full TJ only a subset of the essential parameters were considered. Nevertheless this included all the most important parameters. For instance, for each probe the wave mode, type, size, frequency, beam angle, focal range and sensitivity were reviewed and justified.

264 In any inspection it is important to reliably decide if an indication originates from a defect or from a benign source since reporting of large numbers of benign indications as defects can result in masking of a real defect. A skilled ultrasonic operator will reliably identify all real defects and make few “false calls”. The IP sets down criteria to guide the operator in sentencing an indication. In line with current good practice these criteria are based mainly on pattern recognition rather than signal amplitude. Essentially the philosophy is to analyse the features in the ultrasonic indication and decide whether they are characteristic of a genuine (planar or volumetric) defect or whether they are characteristic of a benign source such as a known geometric feature.

265 Since the data interpretation is not rule based and relies on the operator using his skill, experience and the guidance provided to interpret and sentence indications it is essential, as recognised in the IP, that the operator is trained and qualified on representative specimens. As it is an important and clearly identified role of the QB to ensure that the operators are adequately trained and qualified for the inspection I have not raised this as an assessment finding.

266 Under a section entitled “Accommodation of Key Influential Parameters” the impact of surface roughness and undulations, cladding defect morphology and defect misorientation are discussed in a realistic manner. Based on this review it is judged that the most difficult credible planar defect, greater than the QEDS to detect, would be the same as identified for Sizewell B, namely a planar defect 25 mm through wall, with 0° skew and 8° - 10° of tilt positioned 1/3 way through wall. In the Sizewell B trials this defect was detected with both the 70° probes and the tandem probes. Given the similarity in the proposed inspection techniques and the weld to that at Sizewell B, the IP concludes that a similar defect would also be detected in the AP1000. I consider this to be a reasonable judgement.

267 The IP provides experimental evidence from the validated manufacturing inspections for Sizewell B, the qualified in service inspections of the Ringhals RPV, the UKAEA Defect Detection Trials and the ASME-qualified Wesdyne RPV shell weld procedure to support the claimed capability.

268 As explained above the IP was not intended to be a full TJ. In a number of areas the discussion was sparser than I would expect in a full TJ and certain sections were not included. The main aspects not included in full were:

- Parametric studies of the parameters.
- Evidence to support the selected inspection hardware and software.
- Advice on test pieces required for open and blind trials.
- A systematic review of the evidence and arguments presented in the other sections to support the claimed inspection capability. (This was covered in the IP but a more detailed analysis would be required in a TJ.)

4.2.5.1.2 IVC Review of RPV Upper Shell - Lower Shell Weld Inspection Plan

269 The IVC reviewed this IP and provided a Validation Certificate GDA_AP1000_VC1.(165) which concludes that *"It is IVC's Judgement that the proposed NDT techniques defined in Reference 1, when fully developed, will meet the inspection requirements defined in the same document and will be capable of being formally qualified*."

270 During my assessment I was given the opportunity to see the questions raised by the IVC, the responses given by Westinghouse and the agreement reached. In my view the questions were both pertinent and robust; in short I believe that the IVC acted as an effective quasi-Qualification Body within the constraints of GDA.

4.2.5.1.3 Assessment of RPV Upper Shell - Lower Shell Weld Inspection Plan

271 The proposed inspection is broadly similar to the automated inspections carried out at Sizewell B and for ISI at Ringhals and recognises the need to achieve near specular reflection from defects of the most likely orientation combined with the need to detect defects of less likely orientations and morphologies to provide some strength in depth.

272 Whilst this inspection is consistent with similar inspections carried out in the UK I understand that it is more thorough than is likely to be required in the USA.

273 The Inspection Plan has embraced the ENIQ methodology appropriately for GDA but it is recognised that it is not a full TJ. It is therefore an assessment finding that any Licensee will need to prepare a Technical Justification for this Weld (and the other HSS and HI welds). This is addressed by Assessment Finding **AF-AP1000-SI-07.**

274 I noted that whilst the defect types (morphology and orientation) were specified there was no justification that these were the only types that needed to be considered. This justification will need to be either in the TJ or the safety case. This is addressed by Assessment Finding **AF-AP1000-SI-08.**

275 The use of an Inspection Qualification Body is an essential part of the ENIQ methodology which has worked well in GDA; this process therefore needs to be carried through into the NSL phase which I have therefore captured as Assessment Finding **AF-AP1000-SI-09.**

276 Overall I am satisfied that Westinghouse has demonstrated that they understand how to prepare a Technical Justification which meets ENIQ requirements and that it is likely that a full TJ with its supporting blind trials is likely to be capable of demonstrating that defects 25 x 50 mm in extent can be reliably detected, characterised and sentenced.

4.2.5.2 Inspection Plan for RPV DVI Nozzle - Shell Welds

277 Within Ref. 164 Westinghouse also provided an IP for the Direct Vessel Injection (DVI) nozzle to shell weld which is a somewhat smaller (184 mm ID) nozzle inset into the upper shell. The IVC also validated this IP and the next few paragraphs summarise the IP and my assessment of it.

278 The defect specification and the required inspection performance were identical for the upper shell to lower shell weld discussed above. The consideration of the influential and essential parameters were also common to both inspections but due to the different geometry the probes and access surfaces were different.

279 As before, pulse-echo inspection will be performed from the inside and outside surface with a range of probe angles. In addition 0°, 15°, 20°, 30 ° and 45° probes will be deployed from the nozzle bore. These are used to ensure near normal incidence on defects aligned with the fusion face and to maximise coverage in the complex geometry.

280 This results in a total of 19 probe/surface combinations for defect detection and detailed coverage diagrams are shown for each. Despite this number of probes there is a 114° zone around the circumference where embedded defects close to the inside surface will not be interrogated with a beam having the desired less than 20° misorientation to the normal to the defect face. In fact it could be up to around 30° misalignment for the worst defect tilt. Arguments are presented which predict some detection capability in this case.

281 Achieving an adequate inspection procedure for this complex geometry has required some imagination in the use of probes with non standard angles. The thorough use of coverage diagrams to identify the areas of reduced capability was essential in this process. Despite the poor coverage in some areas I believe that it is unlikely than adding even more probes to the inspection would significantly improve the capability. I am therefore satisfied that the proposed inspection is As Low As Reasonably Practicable (ALARP) for a Standard Class 1 weld.

4.2.5.3 Other Qualified Non-destructive Examinations

282 In addition to the two IPs described above, five other IPs were completed and supplied for assessment in four reports (Refs 182-185). These covered the following welds:

- RPV Inlet Nozzle to Safe End Weld.
- PZR Upper Head to Upper Shell Weld.
- PZR Shell and Manway Weld.
- PZR Surge Nozzle to Safe End weld.
- Main Steam Nozzle to Pipe Weld.

Unfortunately, these IPs were supplied very late in the assessment period and I have therefore only been able to carry out a high level review of these documents at this stage. However, a more detailed assessment of the inspection plans post GDA Step 4 will be required to confirm that an adequate justification has been made before I am confident to support a DAC. This will be carried out under Action 2 of GDA Issue **GI-AP1000-SI-01.**

283 Each of these inspection plans were subject to review by the IVC as the quasi Qualification Body and for each they provided validation certificates (Refs 186 to 189). In each case they confirmed that they believed that when fully developed the inspection technique was capable of being formally qualified. However in the case of the RPV Inlet Nozzle to Safe End Weld, there was a caveat that they judged that insufficient evidence had been provided to support the capability for embedded transverse defects.

4.2.5.4 Reconciliation Following Completion of Fracture Mechanics Assessments

284 The Inspection Plans were developed using a QEDS based on the best judgement of the fracture mechanics experts prior to completion of their detailed analysis. The fracture mechanics assessments have since been completed and have, in several cases, resulted in a reduction in the QEDS as shown in the table below. The QEDS value derived from fracture mechanics is half the limiting defect size (ELLDS).

Component	Weld	QEDS assumed in Weld Ranking Report	QEDS derived from Fracture Mechanics Assessment	QEDS used for NDT Technical Justification
RPV	Lower Shell to Upper Shell	25	12.5	25
	DVI Nozzle to Upper Shell	25	12.5	25
	Inlet Nozzle to Safe-End	6	6	6
SG	Lower Shell Barrel A to Tubesheet	15	5.4	N/A
	Main Feedwater Nozzle to Shell	15	15	N/A
	Inlet Nozzle to Safe-End	Added after weld ranking report prepared.	10	N/A
PZR	Upper Head to Upper Shell	15	11.75	10
	Upper Shell to Middle Shell			N/A
	Manway to Shell	15	11.75	10
	Surge Nozzle to Safe-End	5	5	5
MSL	SG Main Steam Nozzle to Pipe	10	7.5	7.5
MCL	SG Inlet Nozzle Safe-End to Pipe	10	6.9	N/A

285 In TQ-AP1000-1255 (Ref. 9) I asked Westinghouse to provide a commentary on whether they judged that that the inspection was capable of being qualified against the new smaller QEDS values. In response they confirmed that they were and gave a justification which appears reasonable. At my suggestion they also invited the IVC to review their justification. This resulted in a revision to their response (Ref. 190) which was supported by the IVC (Ref. 191). I intend to review this response under Action 2 of GDA Issue **GI-AP1000-SI-01;** however it is clear that this review has concentrated on the detectability of surface breaking and near-surface, embedded defects. The capability of detecting deeply embedded defect appears not to have been considered and I will consider whether this is acceptable during my review.

286 The table above also shows that the QEDS derived from the fracture mechanics assessment for the SG Lower Shell Barrel A to Tubesheet is significantly smaller than assumed when the Weld Ranking Report (Ref. 163) was prepared. This potentially challenged the judgement not to prepare an IP for that weld since when the judgement was made it was believed to be easier to inspect than the shell welds on the RPV and the PZR. In their revised response to TQ-AP1000-1255 (Ref. 190) Westinghouse presented arguments for being able to detect defects only 5.6 mm through-wall if they are surface breaking or embedded and near either surface. These arguments were endorsed by the IVC (Ref. 191). Again the arguments presented make no claims about the capability for detecting deeply embedded defects. This response will be reviewed under GDA Issue Action **GI-AP1000-SI-01.A2).**

4.2.5.5 Conclusions, Issues and Findings Relating to Qualified Manufacturing Inspections

287 The two Inspection Plans which I reviewed were thorough and provided a reasonable argument the required inspection capability could be achieved and I share the IVCs view that an acceptable Technical Justification could be prepared for these inspections. The argument that these can be extended to the smaller QEDS required, in some cases, following the completion of the fracture mechanics assessments are less well developed but nevertheless reasonable. My high level review and the endorsement by the IVC of the IPs which I have not been able to assess within Step 4 has provided me with sufficient confidence that TJs could be developed for these inspections to enable me to support an IDAC.

288 I have raised Action 2 of **GI-AP1000-SI-01** to enable me to complete my assessment of the remaining IPs and the arguments presented in Ref. 190 to support the smaller QEDS.

> ***GI-AP1000-SI-01.A2:*** *Westinghouse shall demonstrate that there are qualifiable inspection techniques capable of detecting the limiting defects with adequate margin in a representative range of components for which the likelihood of gross failure is claimed to be so low it can be discounted. The main activity shall involve making adequate responses to questions arising from ND assessment of documents submitted during GDA Step 4 or as a result of this Action.*

289 The complete GDA Issue and associated actions(s) are formally defined in Annex 2.

290 The following Assessment Findings have been raised.

> ***AF-AP1000-SI-07:*** *The Licensee shall prepare Technical Justifications for the qualified manufacturing Inspections of all the HSS and HI welds.*

> ***AF-AP1000-SI-08:*** *The Licensee shall justify the selection of defects used to qualify each manufacturing inspection.*

> ***AF-AP1000-SI-09:*** *The Licensee shall set up a robust and independent Inspection Qualification Body.*

291 All three findings must be completed before the qualified manufacturing inspections are performed and I have therefore linked them to the generic milestone Install RPV. This is because it would be extremely difficult to make substantive changes once the components start to be installed which could then lead to substantial delays and additional costs.

4.2.6 Forging Inspections

292 Westinghouse commits within the CSRs (Refs 17, 21, 22, 26 and 27) to carryout a full defect tolerance assessment of all HSS and HI locations including selected locations within the parent material however the vast majority of the forgings will be inspected in accordance with the requirements of ASME III (Ref. 30) only. This code requires that forgings are inspected using both a surface inspection technique and also ultrasonics to detect embedded defects. Radiography is not generally required. The two allowed surface inspection techniques are magnetic particle inspection and dye penetrant inspection and I am confident that both have good capability for detecting small surface defects in forgings. The capability of the ultrasonic inspections is less easy to quantify and so I arranged a small contract with Serco to review this aspect.

293 Serco reviewed what types of defects were most likely to be found in modern forgings and assessed the capability of an ultrasonic inspection which complied with the minimum requirements of ASME III to detect and characterise such defects. They considered both a forging with a simple geometry and also, as a sample, the more challenging geometries in the steam generator channel head. Their findings are reported in Ref. 181 and summarised below.

294 There is relatively little literature available describing flaws found in heavy metal forgings used in the nuclear industry but by increasing the search to include other industries it was possible to identify three classes of defect:

- Planar defects that arise from discontinuities formed when the steel is folded over itself in the forging process. These comprise laps, centre burst cracking and hydrogen flaking. Typically these defects are planar (although they can be volumetric) and roughly parallel to the component surface;
- Planar defects formed primarily under the action of stresses generated in the component such as surface cracking, micro-cracking and hot tears. Typically these would be roughly perpendicular to the surface and are likely to depend on the working directions in the forgings;
- Volumetric defects such as voids, segregations and possibly centre bursts.

Where defects of the three types listed above have occurred they tended to be small, at the most a few mm in extent, and therefore not of structural significance.

295 ASME requires repair of any linear surface-breaking defects longer than 5 mm in thick section (> 50mm) forgings. Westinghouse has imposed more stringent requirements than ASME for defects detected with ultrasonic angle beams. They require (Ref. 159) that any defect which is crack-like or is near the surface and gives a signal with amplitude greater 20% of the signal from the calibration defect is reported to the Purchaser for evaluation and acceptance. The calibration defect is a 60° V notch of height 3% of wall thickness or 6 mm, whichever is the smaller. So a forging should not go into service if it contains a crack-like defect provided that this defect has been detected and appropriately characterised.

296 The full procedures for inspection of the forgings were not available within GDA but ASME specifies the use, as a minimum, of a 2.25 MHz 0° longitudinal wave probe and a 1 MHz 45° shear wave probe, although other frequencies can be used if desirable.

297 Over the years there have been many privately funded reviews of the capability of ultrasonic inspections which remain confidential. However, based on their knowledge and experience, Serco conclude that:-

- Detection of volumetric defects is generally good.
- Small misoriented planar defects are easier to detect than large misoriented defects.
- Large smooth planar defects are readily detected if their misorientation relative to the ultrasonic beam is less than 20° of tilt and 5° of skew for 4-5MHz probes increasing to 10°of skew for 2 MHz probes.
- Large, rough, planar defects are easier to detect, if misoriented, than smooth ones.
- Planar vertical defects are easier to detect if they break the surface.

- The minimum size of planar defect which can be reliably detected in a thick section ferritic component is 3 mm through wall.

298 From this I conclude that significant volumetric defects and planar defects parallel to the surface should be detected readily. However it is possible that smooth planar defects may not be detected simply because they are not well oriented to the beam. Careful design of the inspection is therefore required to minimise the risk of this.

299 As an example of a complex geometry Serco reviewed the capability which could be achieved in the steam generator channel head. They concluded that it would be possible to achieve full coverage but that optimised contoured probes would probably be required for the near surface regions under the nozzle inner and outer blend radii.

300 Ultrasonic inspections are only capable of detecting planar defects with high reliability if the defect is relatively well oriented to the beam. When designing weld inspections this is relatively easy to deal with as the cracks are likely to be either along or across the weld. This is not the case for forgings so either a very large number of probes and scans are required, which is impractical, or a more pragmatic approach is adopted such as that required by the ASME Code. The licensee will therefore need to justify the coverage and capability of these inspection even though it is not a requirement for these inspections to be qualified, See Assessment Finding **AF-AP1000-SI-10** below.

301 On the evidence presented above, an inspection meeting the minimum ASME III requirements of a 2.25MHz 0° longitudinal wave probe and a 1MHz 45° shear wave probe pointing in two perpendicular directions will have a reasonable chance of finding embedded planar defects but this cannot be guaranteed because of the range of conceivable orientations. Thus the safety case for the absence of significant defects in forgings relies less on inspection evidence than for welds and more on the confidence that the forgings are well made. As discussed in Section 4.2.4.2.2 I accept the argument that forgings are largely free from residual stress and likely to have a higher fracture toughness than welds so even if defects were present in forgings they are less likely to grow to the limiting defect size than similar defects in welds.

302 As the proposed inspection coverage will not detect defects of any arbitrary orientation the discovery of any planar defect more than a few mm in either direction should be seen as indicating poor control of the forging and thus the need for a review of the inspection strategy to confirm the extent of defectiveness. In this case simply arguing that the planar defect which was found is of acceptable size or has been removed would not be sufficient. The inspection procedures need to make this clear, see Assessment Finding **AF-AP1000-SI-10** below.

4.2.6.1 Conclusions and Findings Relating to Forgings.

303 The following Assessment Finding has been raised.

> ***AF-AP1000-SI-10:*** *The Licensee shall develop inspection procedures for the HI, and HSS forgings and justify their coverage and capability These procedures should specify the actions to be taken if defects are detected*

304 These procedures need to be in place at the time the forgings are inspected and since the inspections are part of the forging manufacturing process I have linked this finding to the generic milestone Install RPV. This is because it would be extremely difficult to make substantive changes once the components start to be installed which could then lead to

substantial delays and additional costs. In practice the findings will need to be addressed earlier to match the forging manufacturing process.

4.2.7 Derivation of Materials Properties Especially Toughness

4.2.7.1 Initiation Toughness Data

305 Lower bound materials toughness properties are required for the fracture assessments on the HSS Components and these toughness properties need to be underpinned by additional fracture toughness test testing on parent material and representative welds.

306 The toughness data used by Westinghouse in their fracture assessments for GDA is presented in the Methodology report (Ref. 52), and proposals for additional fracture toughness testing to support the values used are contained in Ref. 64.

307 The fracture toughness values are taken from a number of recognised sources, and I am generally satisfied with the values that have been assumed for the purposes of the GDA fracture assessments.

308 It is notable that an upper shelf initiation toughness of 220MPa√m is assumed for the low alloy steels. No differentiation is made between forging material and weld material so it is assumed that this value applies to both parent forgings and welds. I understand that this value is consistent with general practice in the United States, and is referenced back to Ref. 65 which provides the technical background to the ASME XI Flaw Evaluation Procedures, Ref. 66. Whist this value is in general use in the United States, it 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, Ref. 67, quotes an upper shelf toughness for low alloy steel weld of 170MPa√m above 200^{o}C.

309 It is outside the scope of my assessment to review the reasons behind the differences. However, I am content for the Westinghouse to use the 220MPa√m for their fracture assessments on the basis that the value will be underpinned by the additional fracture toughness testing on parent material and representative welds.

310 My concern about using this relatively high value is if the additional fracture toughness testing on the parent material and representative welds failed to support the value assumed, as this would lead to a difficult situation during the manufacturing stage. This is essentially a commercial risk for Westinghouse as it will be necessary for a Licensee to provide a fully defensible justification should any such difficulties arise.

4.2.7.2 Stable Tearing

311 As noted previously, Westinghouse has chosen not to invoke ductile tearing in their fracture assessments. As a consequence Ref. 52 quotes all toughness values in terms of initiation toughness.

4.2.7.3 Additional Fracture Toughness Testing Proposals

312 The proposals for additional fracture toughness testing to support the values used in the fracture assessment are contained in Ref. 64.

313 Ref. 64 proposes to undertake fracture toughness testing beyond that required by the ASME III code on the beltline forgings and a further group of forgings with limiting Charpy

properties to confirm the fracture toughness assumptions made for the HSS forgings and weld/HAZ regions.

314 Whilst Westinghouse recognised the need to propose additional testing over and above that required by the ASME III (Ref. 30) requirements in their response plan to RO-AP1000-19 on the Avoidance of Fracture (Ref. 68), it was only recognised late in GDA that this would need to be addressed within the GDA timeframe. As a result the actual additional testing proposals in Ref. 64 did not arrive till very late in Step 4 GDA process and it has not been possible to discuss and clarify the proposals with Westinghouse in advance of writing this report.

315 I acknowledge that the proposals in Ref. 64 are positive proposals for additional fracture toughness testing on the material used for the HSS components, but the proposals as currently stated do not yet adequately satisfy my expectations. In particular I do not fully understand the scope of the testing on the forged material, what proposals are being made for the testing of weld/HAZ material, nor what proposals are being made to account for batch-to-batch variability in the welding consumables. There is also a need to explain how the proposals will cover the testing of weld/HAZ on all the different types of HSS welds including the dissimilar metal welds, MSL welds and potentially the HI weld on the MCL.

316 Whilst the details of the additional testing are a matter for the Licensee, it is necessary to establish the principles of the additional testing within GDA. The proposals as currently stated are not adequate for this purpose. However, they are sufficient to give me confidence Westinghouse should be able to meet my expectations following a period of further discussion and clarification.

317 As a consequence I am content to support an IDAC at this point in time, but there is a need for further discussion of the proposals beyond Step 4 of GDA in order to develop the principles to a point that will meet my expectations and allow me to support a DAC. I have therefore raised GDA Issue Action **GI-AP1000-SI-01.A3** as part of the overall GDA Issue on Avoidance of Fracture to allow for the ongoing assessment activity in this area.

4.2.7.4 Remaining Material Data

318 The materials property data in Ref. 52 also includes upper bound crack growth rates and other more generic materials data needed for the fracture assessments such as temperature specific minimum yield strength values and material stress strain curves (the latter used to derive material specific failure assessment diagram for the fracture assessment to the R6 Procedure, Ref. 50.

319 The crack growth data is taken from the ASME XI, Ref. 66, and I am satisfied with the use of this data.

320 The more generic materials data comes from a variety of sources. The minimum yield stress values are taken from recognised sources, but the origins and provenance of some of the other data is not always clear. For example, material stress strain curves are referred back to an internal Westinghouse Calculation Note (Reference 14 of Ref. 52) and this gives no indication of the origin and provenance of the data.

321 Whilst there is a shortfall in the referencing and justification of some of the data within Ref. 52, I do not believe there is any reason to dispute the values quoted. Thus for the purposes of GDA I am prepared to accept that the remaining materials data in Ref. 52 provides an adequate basis for the fracture assessments.

322 The material data supplied in Ref. 52 is adequate to support the fracture assessments 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 (Section 4.2.7.3). 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 and will need to be updated through life following developments in the field and in the light of through life testing of materials subject degradation mechanisms. This is taken forward as Assessment Finding **AF-AP1000-SI-11**.

4.2.7.5 Conclusions and Findings relating to Derivation of Materials Properties

323 Lower bound toughness data and other materials data used by Westinghouse in their fracture assessments for GDA is presented in Ref. 52, and proposals for additional fracture toughness testing to support the fracture toughness values assumed in the analyses are made in Ref. 64.

324 There is a shortfall in the referencing and justification of some of the data presented in Ref. 52 but I do not believe there is any reason to dispute the values quoted and the toughness properties will be underpinned by the additional fracture toughness testing programme on parent material and representative weld mock ups. The upper shelf initiation toughness assumed for the low alloy steels is consistent with general practice in the United States, but is higher than that quoted in more recent European Nuclear Pressure Vessel design codes. I am content for Westinghouse to use this value in their justification as it will be underpinned by the additional fracture toughness testing undertaken on parent material and representative welds, and it is essentially a commercial risk if a shortfall was discovered through this testing.

325 I therefore accept that Ref. 52 provides adequate lower bound toughness data and other materials data for use in the GDA the fracture assessments. However, post GDA the Licensee will need to produce a comprehensive material data set for use during the design and assessment process, and also to support through life operation. This is taken forward as Assessment Finding **AF-AP1000-SI-11**.

> ***AF-AP1000-SI-11:*** *The Licensee shall produce a comprehensive material data set for the HSS, HI, Standard 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.*

326 The basic design data and its pedigree should be available to support the programme to demonstrate avoidance of fracture, which should be completed before the components are installed because it would be extremely difficult to make any substantive changes once the components start to be installed. However, the overall finding will be linked to the generic milestone of Hot Operations as the confirmatory test data may not be available until after the components have been installed in some cases. The timescale for the through life updating of the data is outside the scope of this milestone.

327 The additional fracture toughness testing proposals are not yet adequate to fully establish the principles of the additional testing within GDA. The proposals are, however, sufficient to give me confidence that Westinghouse should be able to meet my expectations following a period of further discussion and clarification, and as a consequence I am content to support an IDAC. Further discussion of the proposal will be required beyond Step 4 of GDA in order to develop the principles and GDA Issue Action **AP1000-SI-01.A3** includes support for this ongoing assessment work on the additional fracture toughness testing proposals.

*GDA Issue Action **GI-AP1000-SI-01.A3:** Westinghouse to provide formalised proposals for additional materials testing to underpin the fracture toughness values used in the fracture analyses.*

Activities should comprise:

- Provision of formalised proposals for additional materials testing to underpin the fracture toughness values used in the fracture analyses.

- Adequate responses to questions arising from ND assessment of documents submitted during GDA Step 4 or as a result of this Action.

4.2.8 Conclusions and Findings Relating to Avoidance of Fracture

328 I am broadly satisfied that the strategy set down by Westinghouse for demonstrating avoidance of fracture for the HSS and HI is satisfactory. In particular, I welcome the commitment that the fracture analyses will cover limiting loading conditions using lower bound materials properties supported by fracture toughness measurements. Similarly the commitment to undertake manufacturing inspections qualified according to ENIQ principles is welcomed.

329 Because of the late delivery of reports I was not able to complete my full assessment during GDA Step 4. Based on a high level review, I have come to the following conclusions:

- The limiting defect sizes calculated are adequate to support an IDAC but I will need to undertake a more detailed assessment of the fracture assessment reports before I could support a DAC.
- The two Inspection Plans which I reviewed within GDA were thorough and provided a reasonable argument the required inspection capability could be achieved and that an acceptable Technical Justification could be prepared. My high level review, and the endorsement by the IVC of the IPs which I have not been able to assess within Step 4 has provided me with confidence that TJs could be developed for these inspections as well but will require a more detailed assessment before I could support a DAC. The arguments that TJs could also be prepared for the smaller QEDS which are required, in some cases, following the fracture mechanics assessments are less well developed but nevertheless reasonable and will also require a more detailed assessment before I could support a DAC .
- The additional fracture toughness testing proposals are adequate to support an IDAC but are not yet sufficient to establish the principles for a DAC.

330 On balance, I have sufficient confidence in the overall approach to conclude that it should be possible to provide a suitable demonstration for the safety case and thereby to support an IDAC. However a more detailed assessment post GDA Step 4 will be required to

confirm that an adequate justification has been made before I am confident to support a DAC. A GDA Issue has been raised to support this ongoing assessment work post Step 4 and is the subject of **GI-AP1000-SI-01**.

331 The key activities which will need to be completed by Westinghouse under this GDA Issue are:

- Support the assessment by ND of the fracture mechanics analyses across a range of relevant components, locations within components and loading conditions in order to determine limiting defect sizes.
- Support the assessment by ND of the remaining Inspection Plans and the smaller QEDS values.
- Support the assessment by ND of the additional materials testing needed to underpin the fracture toughness values used in the fracture analyses.

332 These activities are described under GDA Issue Actions SI-01 Actions 1 to 3 in Annex 2.

4.3 Materials Specifications and Selection of Material Grade - Reactor Pressure Vessel, Pressuriser, Steam Generator Shells

4.3.1 ND Assessment of Materials Specifications and Forging Processes for Main Forgings

4.3.1.1 Background and Key Issues from Step 3 assessment

333 This activity continues the assessment which followed from Step 3 Regulatory Observation RO-AP1000-21 and captured in RO Action RO-AP1000-21.A03.

334 Materials specifications and forging processes were extensively discussed in the Step 3 report (Ref. 7 Paras 138-164) and included a specialist review by Prof. Knott (Ref. 100).

335 The Executive Summary of Ref. 7 commented “Aspects of the chemical composition of the low alloy ferritic steels for the main vessels (Reactor Pressure Vessel, Steam Generators and Pressuriser) remain to be resolved. This topic will also carry into GDA Step 4, but it is an item that needs to be resolved sooner rather than later. Largely based on authoritative advice received under a support contract, there may be detailed aspects to discuss with Westinghouse relating to several matters of material specification. However I do not see these aspects as fundamental impediments to progress and resolution.”

336 Ref. 7 noted that although the fracture toughness properties of SA-508 Grade 3 Class 1 (specified for the RPV) have been studied extensively worldwide for many years and reasonably extensive databases exist, the extent of such data available for SA-508 Grade 3 Class 2 (specified for the steam generators and pressuriser) will need to be established.

337 The specialist assessment by Prof. Knott raised a number of issues which needed to be taken forward during Step 4 especially:

- Confirmation that the steels will be fully-killed and the process for de-oxidation.
- Ingot casting practice and how much material is discarded (to remove segregation or inclusions) needs to be specified and available for review.
- The degree of forging reduction achieved in manufacture of the various forged parts should be confirmed.

- The question of limits on arsenic, antimony, tin and hydrogen should be taken further.
- With modest forging reduction ratios, fully-killed steel should be used, at least for the RPV, and so low silicon levels are appropriate. It should be confirmed whether fully-killed steel is intended to be used for the steam generators and pressuriser.
- Sulphur maximum limits of 0.01% in the 'belt-line' region and 0.025% elsewhere are not ambitious and 0.005% in the 'belt-line' region should be easily achievable, without any major alteration to steelmaking practice.
- Information on tensile properties at operating temperature and scatter in tensile properties within and between forgings should be considered.
- The difference in predicted RT_{NDT} shift between belt-line forgings and welds should be explained.
- For welding processes, details of weld and cladding qualification tests should be reviewed and the corrosion properties of cladding confirmed.

338 More detailed evidence has been sought during the Step 4 Assessment, not just to resolve the questions arising from the Step 3 report but to explore the detailed understanding of the safety case for these very important components.

4.3.1.2 Step 4 Assessment Process

339 As part of my Step 4 assessment, I asked Westinghouse to respond to the comments made by Prof. Knott in his materials review for Step 3 (Ref.100). I used this approach to examine how Westinghouse control the specification and procurement of the forgings for the components of HSS. Following an initial teleconference Westinghouse provided a set of written responses by letter on 16 June 2010 (Ref. 101). I asked Prof. Knott to review these responses and subsequently I asked Westinghouse some further questions of clarification in TQ-AP1000-771. In reply Westinghouse provided six further documents (Refs 105-110) on which I also sought Prof. Knott's expert opinion.

340 I asked Prof. Knott to edit his original review (Ref. 100) to include a commentary on the two rounds of Westinghouse responses. This edited version, Revision 1 (Ref. 102, 14 Sept 2010), was sent by ND to Westinghouse for their information.

341 Subsequently in November 2010, after further discussions between Westinghouse and ND, a significant third set of comments was received from Westinghouse (Ref. 103). In the time available it was not possible to take account of these by a complete revision of Prof. Knott's review and instead some comments were added to the Executive Summary and a second Appendix was included (Ref. 104).

4.3.2 Assessment of Generic Materials Specifications for Forgings

4.3.2.1 Overview of Westinghouse Approach to Control of Materials Specifications

342 This topic has been difficult to assess because Westinghouse appears to specify materials compositions and forging processes at a relatively high level (consistent with the ASME Code) and refer more detailed refinements back to the suppliers and fabricators of the vessels.

343 The essence of the Westinghouse approach is provided on Page 6 of their final response (Ref. 103):

"In summary, the Westinghouse approach is to define the chemical, mechanical property, microstructural and NDT requirements, as well as minimum tempering temperature requirements and PWHT requirements that will insure that the forgings will meet or exceed the requirements of the applicable ASME Code, and allow the material suppliers to determine the specific processes and procedures that will meet these requirements."

344 A particular difficulty in my assessment arises from the fact that the main requirement for Westinghouse vendors is to meet property specifications, and the tone is that how vendors choose to modify their processes to meet these specifications is very much their own proprietary information. Since I regard Westinghouse as acting as a surrogate Licensee during the generic design assessment, I would expect a greater level of involvement in specifying and controlling the procedures for production of the main forgings. Even if meeting property specifications is considered adequate for start-of-life properties, a more detailed involvement is likely to be needed to ensure that through-life effects are fully understood and controlled.

345 In a number of instances, it is made clear that vendors are required to submit details of their procedures to Westinghouse for approval, but Westinghouse has not been able to clearly explain the criteria that they apply to decide whether or not a submitted procedure is, or is not, acceptable (e.g. Westinghouse responses to questions 3 and 4 of TQ-AP1000-771).

346 I recognise that the procurement arrangements for long lead items have been categorised within the MSQA topic area as out of scope for GDA. Nevertheless my assessment has concentrated on establishing how the technical adequacy of the specifications for manufacturing the forgings for HSS components is achieved.

347 In my view, the high level specifications provided by Westinghouse may have adequate detail for a generic design but they are not sufficiently detailed to control the manufacture and composition of these forgings whose initial and through-life properties are so important to the safety case. In the following sections I have raised a number of assessment findings which will require any Licensee to develop more detailed specifications.

4.3.2.2 Assessment of Forging Processes

348 The Step 3 review by Prof. Knott (Ref. 100) suggested that the steel-making processes should be clarified, especially the processes for killing and de-oxidation.

349 Westinghouse has confirmed (Ref. 101) that the steel will be aluminium-killed and the supplementary specifications for the RPV forgings (Refs 105 and 106) are to be revised to specify this (Ref. 102 and Response 2 to TQ-AP1000-771). The use of aluminium-killed steel is welcomed.

350 Westinghouse has also confirmed (Ref. 101) that the melting procedures involve ladle refinement and vacuum treatment prior to and during the ingot casting. However, based on the evidence we obtained, Westinghouse leaves manufacturers the freedom to submit detailed melting procedures to Westinghouse for approval without having clear criteria for acceptance. The response to Question 3 of TQ-AP1000-771 states that Westinghouse "does not have documentation describing what is and what is not acceptable for the various features of the steel-making and casting process for our vendors". It is difficult for me to establish the adequacy of the procedures approved by Westinghouse if there are

no criteria to decide what is, or is not, acceptable. This is an example of the difficulties mentioned in Section 4.3.2.1 above.

351 Similarly, in the case of the forging procedures, the Westinghouse specifications are not specific about the ingot casting practice or how much material is to be discarded. The detailed forging processes are the responsibility of the individual vendors. Westinghouse states that "The vendor's processes must be capable of meeting the AP1000 design specifications. However, the specific processes are typically the proprietary information of the individual vendors." (Ref. 102 and Response 4 to TQ-AP1000-771).

352 This approach will place additional requirements on any Licensee for a UK AP1000. I would expect the Licensee for a UK AP1000 to demonstrate a tighter control of the specification of the melting, forging and heat treatment operations to ensure that they provide full coverage of the parameters which need to be controlled and are consistent with modern good practice. This is addressed by Assessment Finding **AF-AP1000-SI-12**.

353 Westinghouse has confirmed (Ref. 101) that the current design of the AP1000 RPV applies a set-in arrangement for the main inlet and outlet nozzles utilizing full penetration welds across the thickness of the nozzle ring.

4.3.2.2.1 Conclusions and Findings on Forging Processes

354 The Licensee needs to ensure that the document envelope for the main pressure vessel forgings is more comprehensive and that supplementary requirements documents provide full coverage of the parameters which need to be controlled and are consistent with modern good practice. The following Assessment Finding has been raised:

> ***AF-AP1000-SI-12:*** *For the casting and forging manufacturing processes, the Licensee shall explain how the details of suppliers' procedures are assessed and provide the criteria used for deciding on whether they are acceptable. Examples of the aspects to be fully documented are the details of the casting process, control of segregated regions and material discarded, forging processes and forging ratios and heat treatment details.*

355 This activity must be complete before procuring the long lead items and is linked to the generic milestone of long lead item and SSC procurement specifications. This is because it would be extremely difficult to make changes to the forgings once that have been manufactured, which could then lead to substantial delays and additional costs.

4.3.2.3 Assessment of the Chemical Compositions and Mechanical Properties of Forgings

356 The RPV forgings are SA 508 Grade 3 Class 1. Details of the chemical composition are provided in Table 3 for the forgings and in Table 4 for the core region welds.

357 The main Steam Generator and Pressuriser forgings are SA 508 Grade 3 Class 2 (although some small secondary side nozzles on the Steam generator are SA508 Class 1A). SA-508 Grade 3 Class 2 has the same chemical composition as Grade 3 Class 1 but has a higher specified strength (about 12% higher). The chemical compositions for these forgings are listed in Table 5.

358 The selection of SA508 Grade 3 Class 1 for the RPV and SA 508 Grade 3 Class 2 for the steam generators and pressuriser is appropriate, with the proviso that additional restrictions on composition are likely to be required and I note that Westinghouse uses supplementary material specifications for this purpose.

359 Some supplementary materials specifications have been provided (e.g. Refs 105 and 106), but these documents are stated to be under revision (Westinghouse's response to Questions 2 and 11 of TQ-AP1000-771).

360 Some of the information provided has been unclear, for example Westinghouse's response to Question13 of TQ-AP1000-771 claimed that the sulphur level in the core region has been reduced to 0.005% which is consistent with our expectations. However the most recent Westinghouse information (Page 16 of Ref. 103) states that the sulphur limit remains unchanged in the core region at 0.01%. I am encouraged to note that the maximum sulphur level in the other RPV forgings has been reduced to 0.015% (Ref. 105) from the ASME III value of 0.025%. However this latest information on the belt-line forgings is disappointing since we understand (Ref. 104) that 0.005% S should not be difficult to achieve and would be beneficial, for example in reducing propensity to stress relief cracking (Page 24 of Ref. 104). I believe that a future Licensee should aim to reduce the level of sulphur in the belt-line forgings and consider whether the levels for other forgings can also be improved. This is addressed by Assessment Finding **AF-AP1000-SI-15**.

361 Table 3 lists the chemical composition of materials selected for the RPV whilst Table 5 applies to the steam generator (SG) and pressuriser (PRZ) shells. For comparison purposes these tables also include the compositions which were applied for Sizewell B and are referred to as UK usage.

362 I judge that the use of ASME III materials is adequate as a starting point for the generic design of the main ferritic forgings. However, additional restrictions may be required to comply with ALARP considerations for start-of-life materials properties and through-life changes and any Licensee will need to justify the actual composition limits specified. This is addressed by Assessment Finding **AF-AP1000-SI-13**.

363 The permitted maximum carbon levels (0.25%C) are higher than for UK usage which is 0.2% in the RPV and SG and 0.22% in the PZR. Although a certain level of carbon is important for achieving mechanical properties and hardenability, restricting the upper limit on carbon content is beneficial in reducing susceptibility to HAZ cracking and in contributing to the achievement of a lower transition temperature.

364 Westinghouse relies on vendors (Ref. 101, Page 11) to "control carbon equivalence values to limit the influences these values would have on their manufacturing, welding, and heat treatment of the material, but Westinghouse does not specify a required carbon equivalence." Although there is a claim (TQ-AP1000-771 response to question 7) that "Westinghouse reviews and approves prior to production all manufacturing, welding, and heat treatment procedures", I have generally not been able to establish what criteria form the basis of this review and approval. As this will be part of the overall duties of a Licensee I have raised an assessment finding for all the main forgings expecting any Licensee to specify reasonably practicable controls on the composition and variability (e.g. in segregated areas) of carbon and other elements which affect the likelihood of defects from welding. This is addressed by Assessment Finding **AF-AP1000-SI-14**.

365 The supplementary material specification for the RPV beltline forgings (Ref. 106) imposes restrictions on copper (Cu), phosphorous (P), vanadium (V), sulphur (S), nickel (Ni) and chromium as reproduced in Table 3. These restrictions are generally appropriate, although the sulphur limit could be more restrictive as discussed above.

366 Westinghouse requires suppliers to measure the composition of a wide range of elements as listed in ASTM E350-95, but limits are not specified on all of these elements. In

particular there are no limits for the residual elements arsenic, antimony, cobalt and tin. The issue with trace impurity elements is whether they can segregate during PWHT and service to prior austenite grain boundaries, to produce embrittlement in addition to any caused by phosphorus. Certainly, there is evidence that they can give rise to reheat cracking during PWHT. These sort of effects may reduce the toughness of CGHAZ regions near End of Life (EoL). Based on specialist advice, I judge that it is desirable to specify limits for such residual elements, and I have raised an assessment finding on this which requires the precedent set by previous UK usage to be taken into account. This is addressed by Assessment Finding **AF-AP1000-SI-16**.

367 The tendency to justify specifications solely on the basis of compliance with ASME III or NRC Regulatory Guides is demonstrated by Westinghouse Response 16 (Page 33 of Ref. 104) relating to the significantly larger Chemistry Factor applied to welds than to forgings as quoted below.

368 "Westinghouse *Response 16. The chemistry factors for both the welds and the forgings are determined using Tables 1 and 2 of Regulatory Guide 1.99, Revision 2. These tables were based on empirical data from commercial power reactors that indicated the shifts in* RT_{NDT} *were overall higher for the welds than for those associated with base metals (forgings and plates)."*

369 This response does not demonstrate an understanding of the reasons for the differences between Chemistry Factors of welds and forgings, nor does it justify that the approach is ALARP.

370 Independently, I have recently received specialist advice from NNL (See Section 4.4) which suggests that the dose-damage relationship in Reg Guide 1.99 was derived from materials with distinctly higher levels of copper and phosphorus and it may not be the most appropriate relationship to describe modern materials with lower levels of these elements. An assessment finding on this matter is derived later in this report (Section 4.4) under irradiation embrittlement, **AF-AP1000-SI-22**.

4.3.2.3.1 Mechanical Properties

371 Tensile test properties are given in Table 2 of SA-508/SA-508M. The minimum yield strength for Grade 3, Class1 (RPV steel) at room temperature is 345 MPa, with UTS 550-725MPa and elongation, A% > 18%. For Grade 3, Class 2 (PZR, SG), the minimum yield strength is 450 MPa, UTS 620-795 MPa, A% > 16%.

372 Prof. Knott recommended (Ref. 100) that information on the scatter in tensile properties should be provided, including data at operating temperatures. Westinghouse has not supplied data on scatter at room temperature. They also replied that they have no plans to conduct tensile tests at operating temperatures and claimed that there is no significant decrease in tensile properties for SA508 at the current operating temperatures.

373 The level of scatter in tensile properties is an indicator of the level of inhomogeneity in the material, and I would expect any Licensee to supply evidence of the capability of the forgemaster's processes and procedures to achieve a satisfactory level of uniformity in mechanical properties. This is addressed by Assessment Finding **AF-AP1000-SI-17**.

374 Table 3 of SA-508/SA-508M gives specified Charpy Impact values as follows:

Grade 3, Class 1 (RPV): at +4.4°C min. average. (of 3) 41J; min. individual. 34J

Grade 3, Class 2 (PZR, SG) +21°C min. average. (of 3) 48J; min. individual. 41J.

The minimum upper shelf energies at start of life are 102J for both belt-line forging and weld.

It is likely that the actual values of Charpy and upper shelf energies achievable with modern materials and forging processes may be better than those specified in ASME III. For example another internationally recognised nuclear code (RCC-M: M2111 and M2112: Ref. 112) specifies values of Charpy Impact at 0^0C of 80J and 60J for minimum average and individual respectively and a minimum upper shelf energy of 130J for all main RPV forgings.

375 The values for start-of-life RT_{NDT} are -23^{o}C for the belt-line forging, -29^{o}C for the “belt-line” weld and +12^{o}C elsewhere. The figures for UK 508 RT_{NDT} are -12^{o}C for all forgings except nozzle forgings and -22^{o}C for nozzle forgings. The RCC-M Code (Ref. 112) specifies ≤-20^{0}C for all RPV main forgings.

I regard the start-of-life RT_{NDT} values for the belt-line forging and weld to be good. However the minimum value (+12^{o}C) specified for other forgings might be improved. I believe that a future Licensee should review whether improved values for the mechanical properties of the forgings could be adopted in the light of other relevant good practice. This is addressed by Assessment Finding **AF-AP1000-SI-18.**

376 As mentioned in the Step 3 report (Ref. 7), an adequate level of fracture toughness data must be available for SA-508 Grade 3 Class 2 to provide evidence in support of the lower bound properties assumed in the generic design. This has been discussed earlier in Section 4.2.6.

4.3.2.3.2 Conclusions and Findings Relating to Chemical Compositions and Mechanical Properties

377 Westinghouse relies on the expertise of their suppliers to achieve the required materials compositions. In my opinion relying primarily on mechanical properties to decide whether products are acceptable does not provide confidence that all important parameters have been adequately controlled. It is therefore necessary for the Licensee to demonstrate that the chemical compositions have been adequately controlled to ensure that the risk of problems occurring with the vessels whether during manufacture or in-service is as low as reasonably practicable.

378 I regard the fundamental choice of materials specifications based on SA508 Grade 3 Class 1 for the RPV and SA 508 Grade 3 Class 2 for the steam generators and pressuriser to be appropriate. However I expect any Licensee to develop more detailed, and where necessary more restrictive, specifications for chemical composition. It is on the basis that such tighter controls are feasible that I am prepared to support a DAC, but have raised the following Assessment Findings:

> ***AF-AP1000-SI-13:*** *The Licensee shall define and justify the chemical compositions of the main forgings regardless of whether the composition is based on ASME III compositions or on more restrictive limits. The justification shall take into account start-of-life materials properties and through-life changes.*
>
> ***AF-AP1000-SI-14:*** *The Licensee shall specify reasonably practicable controls on the composition and variability (e.g. in segregated areas) of carbon and other elements which affect the likelihood of defects from welding.*
>
> ***AF-AP1000-SI-15:*** *The Licensee shall ensure that the maximum value of sulphur content in the belt-line forgings is restricted, either by setting an upper limit not*

exceeding 0.005% or by setting a target value with a rigorous process for reviewing the acceptability of the sulphur content should the actual value be above 0.005%. For the other main forgings (outside the belt-line), the licensee shall also consider whether it is reasonably practicable to reduce the sulphur levels specified.

AF-AP1000-SI-16: *The Licensee shall specify and justify limits on residual elements arsenic, antimony, cobalt and tin which take account of the precedent set by UK usage as listed in Tables 3 and 5.*

AF-AP1000-SI-17: *The Licensee shall supply evidence of the capability of the forgemaster's processes and procedures to achieve a satisfactory level of uniformity in mechanical properties.*

AF-AP1000-SI-18: *The Licensee shall review whether, in the light of other relevant good practice, it is reasonably practicable to improve the mechanical property values in the specification for the forgings.*

379 All these findings must be complete before procuring the long lead items and is linked to the generic milestone of long lead item and SSC procurement specifications. This is because it would be extremely difficult to make changes to the forgings once that have been manufactured, which could then lead to substantial delays and additional costs.

4.3.2.4 Welding and Cladding Issues

380 Table 4 includes the information on weld metal composition recently provided in the filler metal specification (Ref. 107). It is encouraging that the values listed are consistent with or in some cases slightly more restrictive than those quoted for previous UK usage. The limit on Ni content of 0.85% is particularly helpful in controlling irradiation damage for the RPV beltline welds.

381 The additional limits specified for Ni, P, Cr, Cu and V for the RPV beltline welds and listed in Table 4 are considered appropriate. The limit on Cr of 0.15% is new information in GDA Step 4 and is confirmed in the supplementary materials specification (Ref. 106).

382 When I asked about arrangements for preventing underclad cracking, APP-GW-Z0-609 (Ref. 110) was obtained but this describes inspection techniques, which add little to the understanding of how to minimize the incidence of under-clad cracks by paying attention to preheat, inter-clad, and bake-out temperatures and times. No distinction appears to be made between hydrogen-induced "cold" cracks, and cracks which may form during stress relief ("reheat cracks"). Again, there is the impression that avoidance of under-clad cracks is left to the fabricator's "know-how" rather than something for which Westinghouse assumes direct responsibility. Finally, Westinghouse conceded that they do impose a minimum preheat temperature of 121^0C on vendors: this contradicts an earlier statement and in any event is a relatively low limit to set. The values for post-heat temperature may also benefit from being increased. I believe the preheat and post-heat temperatures should be reviewed and evidence provided to ensure that they are consistent with modern good practice. Clarification of controls on grain size is also necessary. This is addressed by Assessment Finding **AF-AP1000-SI-19**. The welding procedures for the highest integrity components are reviewed in Section 4.12 below.

383 As discussed earlier in relation to casting and forging, I have similar reservations about the approach adopted by Westinghouse relating to the main construction welds and cladding operations. Vendors have to submit details of their procedures to Westinghouse for approval, but it is not clear what criteria are applied to decide between acceptance

and rejection. Examples of the approach are given by Westinghouse responses to TQ-AP1000-771; for example Question 7 on welding and carbon equivalent, Question 9 on pre-heat and inter-pass temperatures, and Question 12 on grain size requirements. The absence of acceptance criteria for vendor procedures will place additional responsibilities on any Licensee to demonstrate that all the important parameters of the welding and cladding operations are adequately controlled.

384 Confidence that cladding procedures avoid underclad cracking seems to rely mainly on non-destructive and destructive methods used to examine clad blocks. This is illustrated by Ref. 110 supplied in response to Question 17 of TQ-AP1000-771. Ref. 103 discusses the Westinghouse experience with underclad (hydrogen) cracking and provides some evidence that the cladding processes will be controlled so as to minimise the risk of underclad cracking occurring. Nevertheless, I consider it necessary to check that underclad cracking has actually been avoided by undertaking sample non-destructive inspections on a representative number of the clad forgings. This is addressed by Assessment Finding **AF-AP1000-SI-20**.

385 The internal surfaces ferritic steel components in the primary circuit are clad with austenitic stainless steel cladding to protect them from the corrosive effects of the boric acid. It is essential that this barrier is intact; this is confirmed during and that the end of fabrication by inspection.

386 Ref. 204 defines the inspection requirements. It requires that after any extended period of preheat or PWHT during the cladding process the clad surface in inspected using dye pentrants, in accordance with ASME (Ref. 66). This inspection would reject crack like defects, large rounded indications, four close, aligned point indications and more than ten point indications in a an area of 4000 mm^2 . Once the cladding is complete it will also be inspected using a 70° longitudinal wave ultrasonic probe scanned in two perpendicular directions to detect cracking. I consider that this level of inspection provides the necessary assurance that the cladding is free from defects and that it will protect to the ferritic steel from corrosion.

4.3.2.4.1 Conclusions and Findings on Welding and Cladding

387 The following Assessment Findings have been raised:

> ***AF-AP1000-SI-19:*** *The Licensee shall ensure that welding and cladding procedures are demonstrated to be consistent with modern good practice, that they include appropriate limits for the preheat and post-heat temperatures and that evidence is provided to ensure that grain size is adequately controlled.*
>
> ***AF-AP1000-SI-20:*** *The Licensee shall ensure that sample ultrasonic inspections for underclad cracking are performed during manufacture of the RPV, SGs and PZR.*

388 Both these findings must be complete before the generic milestone for installation of the RPV. This is because it would be extremely difficult to make substantive changes once the components start to be installed which could then lead to substantial delays and additional costs. In practice the findings will need to be addressed earlier to match the manufacturing programme requirements.

4.3.2.5 Responses to Specific Queries from Step 3

389 This section summarises how the specific queries raised in the Step 3 report have been addressed in this report.

390 *1. It should be confirmed that the steels will be fully-killed and what process is used for de-oxidation.* Westinghouse has confirmed that the forgings will be fully-killed using aluminium and ladle refinement and vacuum degassing will be used. This is appropriate. (See Section 4.3.2.2 above).

391 *2. Ingot casting practice and how much material is discarded (to remove segregation or inclusions) needs to be specified and available for review.* Westinghouse has been unable to supply detailed evidence of this topic because Westinghouse relies on vendors submitting procedures which Westinghouse assesses, but the criteria for this Westinghouse assessment have not been provided. (See Section 4.3.2.2 above).

392 *3. The degree of forging reduction achieved in manufacture of the various forged parts should be confirmed.* (See Section 4.3.2.2 above).

393 *4. The question of limits on Arsenic, Antimony, Tin and Hydrogen should be taken further*.

394 No significant new evidence has been provided during Step 4 on the control of residual elements. This has not yet been satisfactorily resolved. (See Section 4.3.2.3 above).

395 *5. With modest forging reduction ratios, fully-killed steel should be used, at least for the RPV, and so low silicon levels are appropriate. It should be confirmed whether fully-killed steel is intended to be used for the Steam Generators and Pressuriser.* This has been satisfactorily clarified for all the main forgings. (See Section 4.3.2.2 above).

396 *6. Sulphur maximum limits of 0.01% in the 'belt-line' region and 0.025% elsewhere are not ambitious and 0.005% in the 'belt-line' region should be easily achievable, without any major alteration to steelmaking practice.* Sulphur can play a role on stress relief cracking. Westinghouse has not provided any evidence to explain why a sulphur level of 0.005% is difficult or not desirable to achieve for the belt-line forgings. I note that the limit has been reduced to 0.015% for the other main forgings. Based on specialist advice, my view is that a limit of 0.005% sulphur should be imposed for the belt-line forgings and a Licensee should consider whether the levels for other forgings can be improved. (See Section 4.3.2.3 above).

397 *7. Information on tensile properties at operating temperature and scatter in tensile properties within and between forgings should be considered*. Westinghouse has made clear that they do not require tensile testing at operating temperatures and consequently have not provided any data. (See Section 4.3.2.3.1 above).

398 8. The difference in RT_{NDT} shift between belt-line forging and weld should be explained. A more satisfactory explanation is still required, as discussed in Section 4.3.2.3.1 above and in Section 4.4.3 below.

399 *9. For welding processes, details of weld and cladding qualification tests should be reviewed and the corrosion properties of cladding confirmed.* Two assessment findings have been raised to address this. (See Section 4.3.2.4 above).

4.3.2.6 Conclusions Relating to Forging Production, Chemical Composition, Welding and Cladding

400 I have reservations about the approach adopted by Westinghouse relating to casting, forging, heat treatment, welding and cladding of the forgings for the main vessels whereby the emphasis is mostly on checking that products have satisfactory measured properties. In a number of areas Westinghouse has not provided details of the criteria used to ensure the adequacy of vendors' processes and procedures. This approach will place additional responsibilities on any Licensee to demonstrate that all the important parameters of the manufacturing operations are adequately controlled.

401 Although Westinghouse has a number of requirements documents which supplement ASME Code requirements, I have identified a number of areas where certain parameters are either not specified or where the specified values have not been demonstrated to be take proper account of relevant good practice.

402 The Licensee for an AP1000 plant will need to demonstrate a detailed understanding of the safety case for the RPV, Steam Generators and Pressuriser. This will require the ability to specify and agree procedures at a detailed level with suppliers to ensure that, in addition to meeting the requirements of the ASME Code, additional controls on composition and manufacturing processes are incorporated where they are desirable and reasonably practicable.

4.4 Effects of Irradiation on Reactor Pressure Vessel Body Cylindrical Shell Forging Material and Associated Circumferential Welds

4.4.1 Background and Summary of Step 3 Assessment

403 This is activity AR09058-5 on the Step 4 Action Plan (Ref. 1). It continues the assessment which followed from Step 3 Regulatory Observation RO-AP1000-22.

404 Neutron irradiation embrittlement of base materials and welds in PWR RPVs have historically been a significant issue and remains so for older reactors. The consequences of this effect were explored in some detail in Step 3 and are recorded in the paragraphs 165-232 in that report (Ref. 7).

405 The lower bound, low temperature fracture toughness of steels is normally defined by a generic curve with a single parameter which sets the nominal "nil ductility" reference temperature, RT_{NDT}. The effect of aging mechanisms, including irradiation, is to increase this temperature and thus cause the steel, at a particular temperature to have a fracture toughness, equal to that of an unaged steel at a lower temperature.

406 The shift in RT_{NDT} with irradiation has been extensively studied and correlations developed to predict how it depends on neutron fluence, various trace elements in the steel principally copper, phosphorous and nickel, and temperature. During Step 3, nine databases were considered (Ref. 153) with the review effort concentrating on US Regulatory guide 1.99 Revision 2, EONY and RM-9. These are respectively, the current US correlation and two correlations under consideration for replacing this in the US.

407 Reducing the level of the elements copper, phosphorus and nickel reduces the shift in RT_{NDT} with irradiation and thus Westinghouse has specified maximum levels for these elements for all for the RPV beltline region, which includes the upper shell, lower shell, transition ring and the associated girth welds. These specifications on chemical composition were considered more fully in Section 4.3.2.3 above and are listed in Table 3.

408 Within this section I am considering two specific issues.

- Is it practical to use a low leakage core to reduce the total neutron fluence to the RPV?
- Is there sufficient confidence in the methods used to predict the fracture toughness at the end of life?

4.4.2 Assessment of the Practicality of Using a Low Leakage Core

409 In RO-AP-1000-22.A2 I asked Westinghouse to investigate the potential for use of a' lower leakage' core, the purpose being to reduce the neutron dose to the adjacent wall of the RPV, to ALARP.

410 Westinghouse responded to this RO in their letter WEC000388 (Ref. 151) in which they explained:-

"The design analysis demonstrates that the core internal design is acceptable assuming the most limiting fuel design. The calculated fluence on the AP1000 RPV vessel over the lifetime of the plant has been demonstrated to be below the widely accepted limits proscribed in the ASME code. The utilization of low leakage fuel designs will further reduce the vessel fluence significantly from an already acceptable point."

411 Thus in their view, even with the most limiting allowed core design, it will still comply with the ASME requirements throughout its design life.

412 There are three areas of concern when considering irradiation embrittlement of the RPV these are the beltline forging and the upper and lower circumferential welds. The other welds and forgings in the RPV are bounded by these. In Ref. 156 Westinghouse has calculated the fluence at the end 60 years operation (taken as 54 full power years bated on a 90% utilisation factor) and used this to determine the end of life RT_{NDT} using the methodology required by the US NRC in the Ref. 157 based on the expected material properties. The results are tabulated below.

Reactor Vessel material	Surface Fluence	End of Life RT_{NDT}
Beltline Forging	8.9×10^{19} n/cm^2	34°C
Upper Circumferential Weld	1.25×10^{19} n/cm^2	60°C
Lower Circumferential Weld	2.5×10^{19} n/cm^2	64°C

413 In their guidance 10 CFR 50.61 (Ref. 192) the NRC staff concluded that RT_{NDT} less than 270°F (132.2°C) for plate material and axial welds, and less than 300°F (148.9°C) for circumferential welds, present an acceptably low risk of vessel failure from pressurized thermal shock events. It can be seen from the table above that based on the expected properties this will be achieved by a considerable margin.

414 Nevertheless Westinghouse concedes that further reductions in the neutron flux could be achieved by the Licensee adopting a lower leakage fuel design than currently proposed which makes use of enrichment differences between various sub regions and burnable poisons.

415 I have consulted my colleagues in the Fuel Team and I am advised that the initial fuel load which has been proposed for GDA has a fuel elements with the highest enrichment at the periphery of the core (often called the out-in design) which will result in a relatively high neutron flux to the RPV in the first cycle. This has been proposed since it gives a relatively uniform radial flux distribution which, amongst other things, makes it more tolerant to faults. Subsequent core designs have not been defined and will be the responsibility of the Licensee.

416 It is clear from the correspondence and discussions with Westinghouse that they believe that within GDA they have demonstrated that even with their proposed core design the end of life RT_{NDT} is readily compliant with the expectation of the USNRC and that it is up to the Licensee to decide the core design for the initial and subsequent fuel load. The expectation in the UK is that each core design is justified in a safety case and that this is demonstrated to be ALARP. This is addressed by Assessment Finding **AF-AP1000-SI-23**. I would expect this ALARP assessment to weigh the safety and commercial benefits of a uniform radial flux distribution with the benefits of reducing the neutron flux to the RPV.

417 Westinghouse was also asked to consider the addition of a radial reflector to reduce the flux. In the letter they advise that they do not believe such a design change at this stage to be ALARP given the significant amount of redesign which would be required and the relatively small benefit that would be gained. I accept this judgement.

4.4.3 Calculation of Shift in RT_{NDT}

418 Westinghouse has designed the AP1000 based on the assumption that the end of life RT_{NDT} can be predicted using the methodology set down in the US NRC Regulatory Guide 1.99 Rev 2 (Ref. 152). This predicts end of life shifts in RT_{NDT} in the beltline due to irradiation ranging from 31°C in the forging to 57°C in the lower girth weld.

419 During Step 3 NNL calculated that shifts in RT_{NDT} which would be predicted with other correlations (Ref. 153) which as explained typically differed by up to 20°C for the most relevant correlations.

420 Within Step 4 I consulted NNL further as to which correlation would be most appropriate for AP 1000. They advised (Ref. 154) that their understanding was that the correlation used in the Regulatory Guide 1.99 Revision 2 was appropriate for the steels with relatively high copper content common in reactors in the 1980s but that this was probably not suitable for modern low copper steels as would be used for the AP1000 RPV. For these steels they believed that EONY database is likely to be the most suitable. This is addressed by Assessment Finding **AF-AP1000-SI-22**. The predictions from these two databases are shown in the table below.

Component	Fluence (10^{19} n/cm^2)	RT_{NDT} shift (°C) according to	
		Reg Guide 1.99*	EONY*
Lower girth weld	2.5	57	42
Upper girth Weld	1.25	48	32
Beltline forging	8.98	31	55

*These predictions are based on an irradiation temperature of 280°C. With the EONY database a lower temperature would predict a lower shift.

421 If this database is used the end of life shifts in RT_{NDT} range from 55°C in the belt line forging to 32°C in the upper girth weld. So using this database the shift in the welds is predicted to be less by around 15°C but the shift in the forging is predicted to be greater by 24°C. Therefore the maximum shift for the beltline is essentially the same at 55°C compared to 57°C. However the PT limit curve is determined by the most onerous ¼ wall flaw which with the Regulatory Guide 1.99 Revision 2 correlation was an axial flaw in the forging; hoop stress is about twice the axial stress which more than offsets the higher end of life RT_{NDT} in the welds. So adopting this database would result in the RT_{NDT} curve being moved to the right by 24°C

422 The reasons for different databases giving different correlations are not fully understood but seem most likely to be related to differences in carbon or nitrogen between the data sets or perhaps the methods of production. Within that context of GDA this is largely academic for two reasons. Firstly all the correlations show that the end of life shift in RT_{NDT} is unlikely to be greater than an historically low value of about 60°C. Secondly surveillance specimens will be used to confirm that the irradiation ageing is in line with predictions. Nevertheless it is important that the Licensee decide which database he intends to use to predict end of life RT_{NDT} and justify its use. This is addressed by Assessment Finding **AF-AP1000-SI-22**.

423 A materials surveillance scheme is proposed using representative samples of parent material, weld and heat affected zone material inserted in baskets attached to the low dose regions around the circumference of the core barrel. The basic principles are outlined in the RPV CSR (Ref. 17). I have assessed the principles of the surveillance scheme within GDA but the detailed implementation will be part of Phase 2 (NSL) since there may be differences of detail depending on the Licensee and/or plant. The detailed implementation of the surveillance scheme has been confirmed as an out-of-scope item in Section 2.3.6.

4.4.3.1 Update on Dose-Damage Relationships

424 I asked NNL for an expert opinion on the extent to which the hardening effects due to irradiation, thermal ageing and strain ageing should be additive.

425 NNL explained (Ref. 154, answer to question 4), that thermal ageing is likely to be very slow for temperatures typical of the RPV beltline (~300^{0}C and below). Long-term ageing tests have shown no significant ageing-induced changes in the mechanical properties of base or weld metal at 300^{0}C. Similarly, strain-ageing effects for the beltline region are expected to be small.

426 Consequently I judge that not regarding the separate mechanisms as cumulative is pragmatic and reasonable for the RPV belt-line forgings and welds. Irradiation hardening is likely to dominate the observed rate of embrittlement.

427 Although thermal ageing is not likely to be very significant for the RPV belt-line forgings, it could be significant at higher temperatures such as those experienced by the RPV outlet nozzles and the PZR. I would expect a future Licensee to have access to an adequate database on thermal ageing effects and this may require a thermal ageing surveillance programme for materials operating at higher temperatures. This is addressed by Assessment Finding **AF-AP1000-SI-23**.

4.4.4 Conclusions and Findings Relating to Effect of Irradiation

428 Westinghouse has proposed an initial core build and a database to predict the shift in RT_{NDT} which is fully compliant with the expectations of the US NRC however whilst this is valuable it not sufficient, by itself, to support these proposals for a UK AP1000. It is remains necessary to justify that it would not be ALARP to have an initial core build with a lower leakage. I also believe that the proposed database for predicting the irradiation shift in RT_{NDT} may not be representative of the steels which will be used for the RPV in the UK so the Licensee will need to identify an appropriate database and justify its use. The following Assessment Findings have been raised:

> ***AF-AP1000-SI-21:*** *The Licensee shall prepare an ALARP justification to support the proposed initial core design which takes appropriate account to the benefits of reducing the flux to the RPV. Safety cases will also be required to support subsequent core designs and these will also need to consider the benefit of reducing the RPV flux.*

> ***AF-AP1000-SI-22:*** *The Licensee shall demonstrate that the damage correlation used to determine the shift in RT_{NDT} is suitable for the RPV materials. This needs to reflect on the current understanding of dose damage correlations and should be kept under review over the life of the plant as new data becomes available from surveillance specimens and from worldwide data.*

429 Both these findings need to be complete before the generic milestone of Fuel Load is reached as it is necessary to ensure that the irradiation damage being accumulated is controlled from the start of operations to ensure that the integrity of the RPV is not compromised towards the end of life.

430 The approach of not regarding the separate hardening mechanisms as cumulative is pragmatic and reasonable for the RPV belt-line forgings and welds. Irradiation hardening is likely to dominate the observed rate of embrittlement. However thermal ageing may be significant for forgings operating at higher temperatures, and the following Assessment Finding has been raised:.

> ***AF-AP1000-SI-23:*** *A Licensee shall have access to an adequate database so that thermal ageing effects can be reliably predicted and, if necessary, a thermal ageing surveillance programme should be established for materials operating at temperatures experienced by the RPV outlet nozzles and the pressuriser.*

431 This finding needs to be complete before the generic milestone of Install RPV. This is because it would be extremely difficult to make substantive changes once the components start to be installed which could then lead to substantial delays and additional costs.

4.5 Pressure – Temperature Limit Diagrams and Low Temperature Overpressure Protection

4.5.1 Background and Key Issues from Step 3 Assessment

432 This activity AR09058-7 on the Step 4 Action Plan (Ref. 1) and continues the assessment which followed from Step 3 Regulatory Observation RO-AP1000-29.

433 The RPV Pressure – Temperature limit curve is used by the reactor operator to ensure that the at all temperatures the reactor pressure is sufficiently low to ensure that vessel will not fail due to the fast fracture; this is particularly important during start-up and

shutdown when the material toughness is reduced due to the lower temperatures. In the case of the AP1000 it is proposed that this will be calculated in accordance with US NRC Regulatory Guide 1.99 Revision 2 (Ref. 152).

434 As described in the Step 3 Report (Ref. 7) the methodology for generating this curve was relaxed in the ASME code introduced in stages between 1998 and 2000. The justification for these changes was not clear when the Step 3 Report (Ref. 7) was prepared and this was proposed for review in Step 4.

435 The importance of the Pressure –Temperature limit curves and the margins they imply on fracture toughness have were recognised in “ND Statement on the Operation of Ferritic Steel Nuclear RPVs” (Ref. 100) this implies that it is important that the margins are as large a reasonably practical. The feasibility of increasing the margins was explored through RO-AP1000-29 A4.

4.5.2 Key Points from the Assessment During Step 4

4.5.2.1 The Change to Using K_{1C} Instead of K_{1A}

436 ASME XI Appendix G calculates a PT limit curve using static initiation fracture toughness K_{1C} rather than the arrest fracture toughness K_{1a} which was the measure of toughness that has been used for Sizewell B. Whilst it this change has been accepted both by ASME in the US and RCC-M in France the justification for this relaxation was unclear and I therefore asked SERCO (Ref. 158) to review this. I also asked them to consider the implications on the potential for failure of the vessel.

437 This change was initiated in the US in the early 1990s and resulted in ASME Code Case N-640 which was incorporated into the ASME Code in 1999. The justification for this change was based of work carried out by Electric Power Research Institute (EPRI) (Ref. 162) and at Oak Ridge Nuclear Laboratory (ORNL) (Ref. 176). The latter being funded by the US NRC.

438 Oak Ridge Nuclear Laboratory (Ref. 176) presented six reasons to support the change. In summary these are:

- Because the temperature change is at a slow constant rate it is technically correct to use the static lower bound fracture toughness K_{1C}.
- K_{1A} was used in 1974 when the rules were first defined to cover uncertainties, since that time a significant amount of work has been undertaken to understand these.
- The calculation assumes the presence of a large surface-breaking flaw, no such flaws have been found.
- By 1999 there was about ten times as much K_{1C} data as there had been in 1974.
- Part of the argument for using K_{1A} was to account for local brittle zones which could result in cracks popping through that zone. Subsequent work has found these are not significant.
- There are benefits in opening up the operational window which could, on balance, improve plant safety.

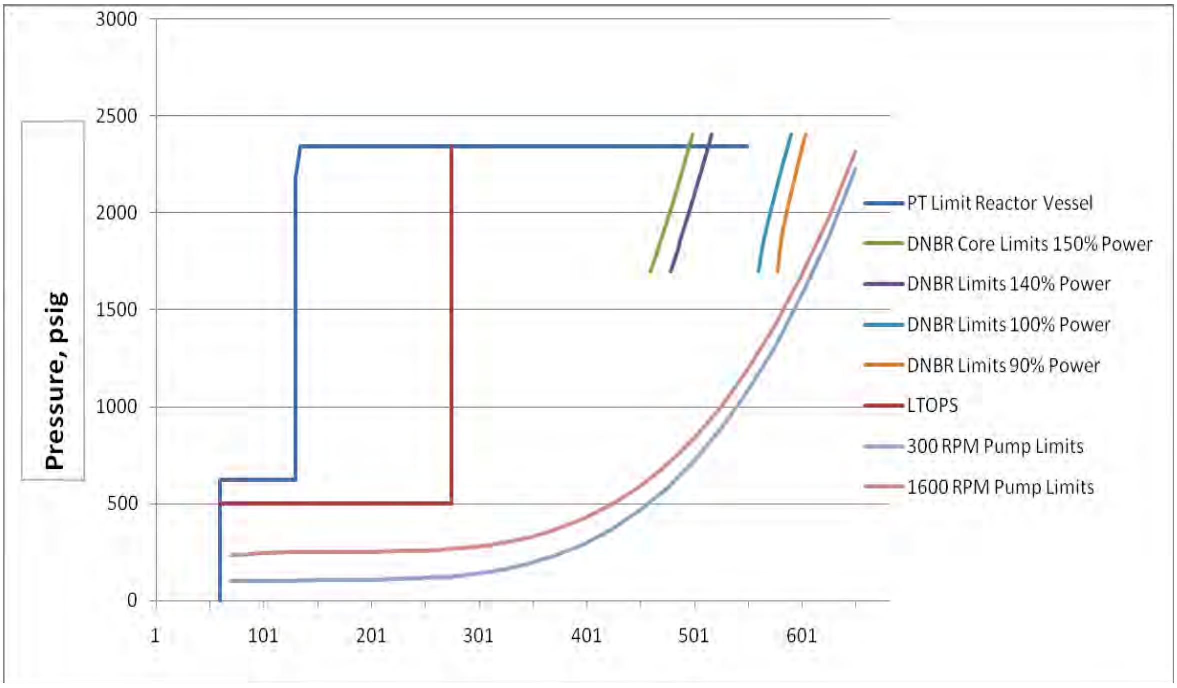
439 It is worth noting that EPRI (Ref. 162) suggested seven further areas for work which might have further relaxed the setting of P-T limits, including changing the assumption that the pressure vessel contains a surface breaking ¼ wall defect. None of these further suggestions were adopted.

440 SERCO (Ref. 158) noted that using the K_{1C} based curve may not be demonstrably conservative if the copper content of the steel is greater than about 0.07%. Since the maximum copper content of the AP1000 RPV beltline forgings and welds is specified to be less than 0.06% this will not be an issue for a UK AP1000.

441 SERCO (Ref. 158) also identified that the ASME K_{1C} based curve may not be conservative if T-RT_{NDT} is less than about -93°C for shallow defects and in general for temperatures of less than -130°C because the curve is not fully bounding. In practice these low temperature concerns are not important here since the expected value or RTNDT at end of life is 64.4°C for the belt line welds and only 34.4 °C for the forgings (Ref. 13) so the vessel would need to be at temperature of well below 0°C before these effects are relevant.

4.5.3 ALARP Basis for the P-T Limit Curve

442 SAP EPS.4 Overpressure Protection requires that '*Overpressure protection should be consistent with any pressure-temperature limits of operation*' (Ref. 4). Westinghouse explained how this would be achieved for a AP1000 in their response to RO-AP1000-29.A4 in letter WEC00247N (Ref. 155) and clarified this further in response to TQ939 which included a figure which is reproduced below.



443 The reactor will be operated with pressures and temperatures above and to the left of the relevant pump limit line and DNBR (Departure from Nucleate Boiling Ratio) line. This protects the pumps and to ensures effective heat transfer in the steam generators. Note the x axis in the figure above is °F. The operators are required by Limiting Condition of Operation LCO 3.4.3 to ensure that the reactor pressure and temperature complies with the curve above. (Chapter 16 of Ref.15). This is an administrative control which requires confirmation every 30 minutes during start-up and cool down.

444 In addition to the administrative control physical control against overpressure is provided at low temperatures by the Low temperature operation protection system (LTOPS). This system automatically opens the relief valve in the normal residual heat removal system at pressures above 500 psi when the cold leg temperature is less than 130°C (275°F) thereby ensuring that the pressure and temperature are always to the right and below that line shown in the figure above. The LTOPS system is controlled through LCO 3.4.14 (Ref. 15). Also shown on the graph is the limit which would be required to simply achieve compliance with the P-T Limit curve.

445 It is clear from the graph that the LTOPS restriction is significantly more onerous than a simple requirement not to challenge the P-T Limit curve so it is instructive to understand how these two are calculated and in particular what margins have be added.

446 The PT limit curve will calculated using the methodology required by US NRC Regulatory Guide 1.99 Rev 2 (Ref. 152) and Westinghouse's calculation is set down in APP-RXS-ZOR-001 (Ref. 156).

- Below 54.4°C (130°F) the pressure is restricted to a maximum of 4.28MPa (621 PSI) and above this temperature the pressure is determined assuming that a ¼ wall axial flaw is present in the forging and that it is subject to an internal pressure load and a bending load due to heat up or cool down of the vessel at the maximum allowed at a rate of 37.8°C/hr. A factor of two is applied to the primary, pressure stress to ensure that the calculation is conservative.
- Linear-elastic fracture mechanics are used to calculate the temperature dependant failure pressure based on a K_{1C} fracture toughness derived from lower bound properties as described above.
- For the PCSR (Ref. 12) the start of life RT_{NDT} has been determined from generic properties of the steel and the end of life RT_{NDT} calculated by adding a shift to RT_{NDT} (ΔRT_{NDT}) dependant on the total estimated fluence at the crack tip (1/4 wall) and the nickel and copper content of the steel. In operation the actual start of life RT_{NDT} will be used.

447 Both the start of life RT_{NDT} and the shift in RT_{NDT} are best estimate values however USNRC 1.99 (Ref. 152) takes account of this by adding a margin to the end of life RT_{NDT} to take account of measurement errors. These are either the measured standard deviation or generic values defined in Ref. 152. In the for GDA the latter are used; the generic standard deviations are 9.5°C for the start of life RT_{NDT} and of 9.5°C for the forging and 15.5°C for the weld for ΔRT_{NDT}. The standard deviations are added together in quadrature and doubled to obtain a margin (M) of 36.4° C. So the

End of Life RT_{NDT} = Start of Life RT_{NDT} + Δ RT_{NDT}+ M .

In addition when the reactor is critical a further margin of 22.2°C is added, resulting in a total margin of 58.6°C.

448 So to calculate the P-T Limit curve lower bound data is used to determine the fracture toughness (K_{1C}) based on a best estimate value of the end of life RT_{NDT} to which a margin of two standard deviations of the measurement errors is added with a further margin added when the reactor is at power. It could be argued that by adding in the margin to account for measurement errors an upper bound (worst case) RT_{NDT} is also being used, however this is not strictly true as it is the measurement error not the scatter in the value of RT_{NDT} with fluence which is being accounted for here. Nevertheless by inspection of the data presented in (Ref. 153) it appears that a margin of 36.4°C readily accounts for

this scatter. Thus providing a realistic best estimate relationship is used to determine ΔRT_{NDT} the P-T Limit curve will effectively be generated from an upper bound (worst case) RT_{NDT} value and lower bound data to establish the fracture toughness and thus should be sufficiently conservative. In Section 4.4 above I discuss the need to ensure that an appropriate database is used to calculate ΔRT_{NDT}.

449 As described above the LTOPS further restricts low temperature operation by opening the relief valve in the normal residual heat removal system at pressures above 500 psi when the cold leg temperature is less than 130°C (275°F). This provides a further margin of around 17°C to the critical P-T Limit curve at maximum reactor pressure and a significantly greater margin of around 100°C at 130°C.

450 In RO-AP1000-29.A4 I asked if it would be reasonably practical to move the P-T Limit curve to the right by 10°C, 20°C or 30°C as I wished to understand it this additional margin could be obtained at little cost. In their response (Ref. 155) Westinghouse conceded that a shift of up to 14°C might be possible but believed that the limits were already very conservative and that this may have a negative impact on safety elsewhere in the system.

451 The figure above shows that operation is quite heavily constrained at low temperature principally because of the crude operation of LTOPS which effectively switches on when the temperature drops below 130°C. Were this system more sophisticated, it might more closely set an operational restriction which ensures that the RPV pressure at specific temperatures is kept away form the P-T Limit curve with a conservative constant margin. However, it is a simple system and therefore likely to be reliable it also has the added benefit of offering the greatest margin to the P-T Limit curve at low temperatures when the vessel is least ductile.

452 From the evidence presented it is clear that if a future Licensee adopts the settings for LTOPS proposed by Westinghouse the RPV will be kept away for the P-T Limit curve by a considerable margin. However, it is the responsibility of a future Licensee to set the LTOPS limits and to justify these, see Assessment Finding **AF-AP1000-SI-24** below.

453 It could be argued that one non-conservatism in this approach is the use of the fracture toughness at the crack tip rather than the inside wall of the vessel since the crack could grow in length rather than through wall. However, if RT_{NDT} were calculated at the wall the additional shift would be about 2°C which I judge to be negligible considering the margins already in place.

4.5.4 Conclusions and Findings Relating to P-T Limit Diagrams

454 Overall I judge that the adoption of the use of the static initiation fracture toughness K_{1C} instead of the arrest fracture toughness K_{1A} to have been adequately justified.

455 Westinghouse has proposed setting the P-T and LTOPS Limits for a UK AP1000 in the same way as they would be require to in the USA. I have concluded that this is conservative and reasonable and therefore does not stop me supporting a DAC however since these are operational limits it will be necessary for the Licensee to incorporate these in his arrangements and justify their adequacy. This is addressed by Assessment Finding **AF-AP1000-SI-24**, which is linked to **AF-AP1000-SI-22**.

> ***AF-AP1000-SI-24:*** *The Licensee shall propose P-T and LTOPS limits for a UK AP1000 and justify these.*

456 This needs to be complete before the generic milestone Hot Operations, because the suitable margins must be maintained at all times once the reactor pressure vessel is taken to operational temperatures.

4.6 Fatigue Usage Factor Analysis Results for the Pressuriser Surge Line

457 This activity continues the assessment which followed from Step 3 Regulatory Observation RO-AP1000-26.

4.6.1 Background and Summary of Step 3 Position

458 The Step 3 Structural Integrity Report (Ref. 7) identified that, as a result of the work reported in Ref. 69, there was a potential non-conservatism in the existing ASME III (Ref. 30) fatigue design analyses for stainless steel components due to the environmental effects of the LWR environments. As a starting point to determine the significance of this matter a request was made for a list of stainless steel component locations whose fatigue usage factor was predicted to exceed 0.75.

459 Westinghouse’s response identified a small number of reactor internal components which came into this category, but these were not considered significant in the Step 3 report. However, Westinghouse was unable to report on the fatigue usage factor for the pressuriser surge line as the detailed analysis work was still ongoing. This matter was not seen as critical to the GDA Step 3 conclusions, and the matter was therefore carried forward to the GDA Step 4 assessment.

460 In addition to this specific piece of work on the pressuriser surge line fatigue usage factor carried forward from the Step 3 assessment, I have undertaken a wider review of the position on the Environmental Effects on Fatigue Design Curves in Section 4.13.

4.6.2 Step 4 Position on the Fatigue Usage Factors Results for the Pressuriser Surge Line

461 Regulatory Observation Action RO-AP1000-AP26.2 (Ref. 10) requested specific information on the fatigue analysis of the pressuriser surge line, but Westinghouse chose not provide any such information during Step 4 of GDA as the fatigue analysis for the surge line is due to be revised during 2011.

462 I questioned whether this revision to the fatigue analysis was specific to the surge line or whether it was part of a wider revision, and why the revision was necessary, TQ-AP-1137 (Ref. 9). The response indicated that the revision was part of a wider package of work on design finalisation to complete the final as-designed ASME code fatigue analysis for Class 1 piping. The response indicated that the bounding fatigue analyses that had been undertaken as part of the initial design work showed cumulative fatigue usage factors in excess of 1.0, but expressed confidence that as successive layers of conservatism were removed, that they would achieve acceptable results. I was not provided with the results from the bounding analyses, but Section 20E.6.1 of the Rev A PCSR (Ref. 13) notes that the MCL design assessment remains an open issue with regard to the surge line, and that it remains to be confirmed that the ASME fatigue limits are met.

463 Thus Westinghouse has yet to demonstrate that the pressuriser surge line design meets the ASME III fatigue limits, but they have confidence that this will be achieved as part of the design finalisation process.

464 I further questioned the extent to which the plant as a whole had not yet been shown to be ASME III code compliant in terms of sizing stress and fatigue limits, TQ-AP-1258 (Ref. 9). The response identifies the current ASME design reports for the equipment considered within Chapter 20 of PCSR. It notes that the reports may be updated as part of the design finalisation activities, and that the March 2011 submittal of the PCSR will reflect the current status of the ASME design compliance. I will review this aspect in due course, but my review of Revision A of the PCSR (Ref. 13) indicates that the pressuriser surge line is probably the most significant area where Westinghouse has yet to show compliance with the ASME fatigue limits.

4.6.3 Conclusions and Findings Relating to Fatigue Usage Factor Results for the Pressuriser Surge Line

465 Westinghouse has yet to demonstrate that the pressuriser surge line design meets the ASME III fatigue limits, but they have confidence that this will be achieved as part of the design finalisation process.

466 I am not prepared to support a DAC until I have confidence that the design has been shown to be compliant with the ASME III fatigue limits for the 60 year life of the plant. I am, however, prepared to accept Westinghouse's assurance that they will be able to demonstrate compliance in order to support an IDAC.

467 I have therefore raised GDA Issue **GI-AP1000-SI-02** for Westinghouse to show that the ASME III Class 1 pipework has an adequate fatigue life for the 60 year design life of the reactor.

> ***GI-AP1000-SI-02:*** *Westinghouse shall provide sufficient evidence to show that ASME III Class 1 pipework has an adequate fatigue life for the 60 year design life of the reactor.*

468 The complete GDA Issue and associated action(s) are formally defined in Annex 2.

4.7 Containment Pressure Vessel Shell

4.7.1 Background and Key Issues from Step 3 Assessment

469 This activity AR09058-8 on the Step 4 Action Plan (Ref. 1) continues the assessment which followed from Step 3 Regulatory Observation RO-AP1000-30.

470 Appendix K of the Containment Vessel CSR (Ref. 19) describes the containment vessel (CV) and its safety function. It is a large, vertical steel cylindrical vessel 39.6m in diameter with ellipsoidal steel closure heads giving it a total height, crest to crest of 65.6m. The lower dome is embedded in concrete and concrete is also poured into the lower dome to provide the support for the structures within containment. The vessel is housed within the containment building which provides protection against aircraft impact and some protection against the weather and also supports a Passive Containment Cooling Water Storage Tank (PCCWST.)

471 The containment vessel fulfils a number of safety functions the most significant of which are :

- It contains any airborne release of radioactive material following design basis accidents (DBA).

- It provided a heat transfer surface to remove heat in normal operation and following steam line breaks and primary circuit loss of coolant accidents. Following major failures the water from the PCCWST would be sprayed on to the containment vessel to aid heat removal.
- It provides structural support for the polar crane.
- It supports the containment air baffle which is designed to encourage a cooling airflow over the containment vessel.

472 This vessel has been designed in accordance with ASME Section III, Article NE-2000 and has been designated within the UK AP1000 safety classification as Class 1. It was judged not to warrant a higher classification because it is not normally pressurised and only provides a pressure boundary function following failure of another component. I am content with this classification.

473 The internal pressures reached in the event of the most severe design basis accidents, double-ended cold leg failure and main steam line failure is slightly below 400kPa compared to the design pressure of 407kPa. The internal air temperatures are also calculated and reach 213°C for the hot leg failure and 190°C for main steam line failure. These temperatures are in excess of the design temperature of the vessel of 149°C however the European Design Control Document (Ref. 15) shows that air temperatures only exceed this for a few seconds for the hot leg failure and for about 10 minutes for the main steam line failure. The PCSR asserts that the metal temperature in this case does not exceed the design temperature and whilst I do not believe this has been proven I judge this to be reasonable however it is necessary for the Licensee to confirm this. This is addressed by Assessment Finding **AF-AP1000-SI-25**.

474 In each of these major failure scenarios the pressure and temperature rapidly drops towards 160kPa and 110°C. These long term pressures and temperature are not particularly demanding for example a domestic pressure cooker on “high“ pressure is designed to work at 100kPa and 120°C.

475 The only UK PWR (Sizewell B) has a steel lined reinforced concrete containment but there are some steel CVs in the United States and most German nuclear power plants use a steel CV and thus the use of a structural steel CV is not unusual. In the AP1000 design it gives a safety benefit by providing an efficient passive heat transfer surface which can be enhanced by spraying water onto it, however the design did require some amendments to ASME III (Ref. 30) to allow Westinghouse to use their preferred material SA-738 Grade B. Principal amongst these were the inclusion of this material in the nuclear code and the exemption of this material from post weld heat treatment up to a maximum thickness of 44mm. The evolution of these changes was described in detail in paras 287 to 303 of the Structural Integrity Step 3 Report (Ref. 7) and in more detail in Ref. 166.

4.7.2 Step 4 Assessment

476 Although the use of a steel containment vessel is not unusual in the nuclear context, three areas were identified as potentially challenging and these were the areas where I concentrated my Step 4 review. They were very small corrosion allowance on the plate thickness, no requirement for post weld heat treatment and loading on the vessel particularly from the polar crane.

477 To assist me in my assessment TWI undertook a review (Ref. 167) of the fabrication procedures for the containment vessel, the acceptability of no corrosion allowance on the thickness of the vessel and protective coating being proposed.

4.7.2.1 Fabrication of the Containment vessel

478 The CV will be constructed from ASME SA-738 Grade B plate. The first cylindrical course is 47.6 mm thick and the axial weld will be post weld heat treated. The rest of the cylindrical wall of the vessel is 44.4 mm thick and in accordance with ASME Section III 2001 edition with 2002 addenda it is not necessary to post weld heat treat (PWHT) these welds. The relaxation to allow this was only agreed in ASME Code Case N-655 which was agreed on 25 February 2002. The heads are 41.3 mm thick and again will not be post weld heat treated.

479 Since I had some concern about the acceptability of not requiring PWHT on welds of this thickness I sought guidance from UK codes. There is no applicable UK nuclear code and the nearest equivalent is the non-nuclear code PD5500 (Ref. 168) which would be used for designing pressure vessels. If there were a UK nuclear code I would expect it to be at least as rigorous and thus this is a suitable benchmark to guide my judgement. If PD 5500 were used the post weld heat treatment requirement would normally be defined in Annex D. However, in this case the steel would not fit into the material banding used there and thus Annex U would be used instead. This Annex requires a fracture mechanics assessment to determine the need for PWHT based on an agreed reference defect.

480 It was not practical to agree reference defect however TWI carried out a basic evaluation of the limiting defect size which, following a review after the report was issued, was very conservatively assessed as 3.6 x 100mm. This calculation assumed that the vessel temperature was -28°C (the minimum design temperature for the vessel), the residual stress in the weld was the yield stress and the applied pressure stress was that achieved in a pneumatic test, this takes the CV to 10% above design pressure. In a LOCA following the gross failure of either the MSL or the RCL the pressure will not exceed the design pressure and even if the CV temperature started at -28°C it is reasonable to assume that the release will heat it up before the maximum pressure is reached, I anticipate that if these two effects were quantified they would result in a significantly larger limiting defect size.

481 I invited Westinghouse to comment on this in TQ-AP1000-1248 (Ref. 9) Westinghouse stated that their design was based on the ASME code and that they thought the fracture mechanic assessment simply provided an alternative approach which they judge is not necessary.

482 In support of their argument they provide Westinghouse report (Docket Number 52-006) (Ref. 169) which was provided to the US NRC to justify that the fracture toughness levels were in compliance with ASME without post weld heat treatment. However since the main benefit from post weld heat treatment is the reduction in residual welding stress which was the dominant stress in the weld and led to the small critical defect size in the fracture assessment I do not currently believe that this addresses my concern. Nevertheless, as discussed above, I have identified a number of conservatisms in the assessment provided and currently I judge that it will be possible to justify the containment vessel has adequate fracture tolerance even without the post weld heat

treatment. I am therefore able to support an IDAC but need to see the arguments and evidence which will be provided in response to Action 2 of **GI-AP1000-SI-04.**

4.7.2.2 Corrosion Allowance on Plate Thickness

483 The 2001 Edition of the ASME Boiler and Pressure Vessel Code with the 2002 addenda are used to determine the overall thickness of the CV and this code does not require a corrosion allowance. With the exception of the first cylindrical course which is 3.2 mm thicker to allow for the potential for gross overall corrosion in the concrete embedment transition region (Ref. 19) and the lower dome where an equivalent margin apparently exists (Section 3.8.2.6 of Ref. 15), the CV has been designed without a corrosion allowance.

484 Given that there is generally no designed allowance for corrosion it is important that steps are taken to minimise the risk of corrosion and therefore Westinghouse has specified (Ref. 171) that:-

'The CV interior surface is coated with an IOZ primer, except for those portions fully embedded in concrete. Table A-2 identifies elevations for application of coating systems in the CV. The interior IOZ coating shall extend downward one foot below the lowest elevation of concrete. Above the operating floor, the IOZ primer is top coated with epoxy. The epoxy topcoat extends above the operating floor to a wainscot height of 7 feet (2134 mm.)

The CV exterior surface is coated with an IOZ primer, except for those portions fully embedded in concrete. Table A-2 identifies elevations for application of coating systems in the CV.'

485 The CV CSR (Ref. 28) provides further evidence to support the acceptability of negligible corrosion allowance on the majority of the CV. The most relevant of these are

- The National Association of Corrosion Engineers (NACE) handbook provides a single face corrosion rate of 0.1 – 0.3mm/year for carbon steel in a "quiet seawater" environment. Thus at the fastest rate it will take 148 years to corrode through wall.
- The air baffle, which is attached to the outside of the containment vessel, is specifically designed to aid evaporation of any moisture which is necessary for corrosion to take place.
- The internal atmosphere of the CV is humidity controlled.
- If corrosion were to occur it is likely to be localised and even with a localised thinned area the vessel will still be able to withstand the internal pressure and maintain containment.
- The exposed internal and external surfaces of the CV will be visually inspected approximately every 40 months to confirm that there is no evidence of corrosion or damage to the protective coating.

486 The lower part of the vessel is embedded in concrete and clearly cannot be inspected. The important arguments for the adequacy of this design are the following.

- The concrete will be inspected and monitored to ensure it is free of penetrating cracks.

- The moisture barrier at the junction where the shell becomes embedded will be subject to ageing management activities in accordance with ASME Section XI Sub-section IWE requirements and will be inspected every 18 months and repaired as necessary.

487 In response to TQ–AP1000-732 I was provided further information about the moisture barrier. Either side of the CV wall there is a 50mm deep, rectangular section channel in the concrete which is 100mm wide on the inside surface and 50mm wide on the outside. The surfaces of the channel, both steel and concrete, are epoxy coated and the channel is filled with a silicone based seal which is pre-compressed and inserted into the channel. I am satisfied that, if inserted properly, this should provide a good moisture barrier nevertheless the inspection every 18 months is an important part of ensuring the barrier continues to be effective over the life of the plant. This is addressed by Assessment Finding **AF-AP1000-SI-26**.

488 In response to TQ-AP1000-1041(Ref. 9) I was provided with report APP-GW-T2R-005 (Ref. 172) which gives extensive and very detailed information on the specific coating selected (PPG Amercoat) and the testing of this against standard including ASTM D3911 and D3912. The tolerance of this coating to a number of chemicals found on power stations was tested and no change was found after 5 days exposure. These chemicals included 1.03% hydrogen peroxide which could potentially drip onto the vessel from the PCCWSR and 5% boric acid which could be present following leakage from the primary circuit. Whilst it is encouraging that no evidence of damage was seen in the 5 day trial it will be necessary to demonstrate that the coating provides a robust barrier for a longer period of time. This is addressed by Assessment Finding **AF-AP1000-SI-27**.

489 This report also gives very detailed information on coating application and repair of coating defects and guidance on the qualification of staff for application and inspection of coatings. It is important that this information and guidance is included in the Licensee's procedures and arrangements. This is addressed by Assessment Finding **AF-AP1000-SI-28**.

490 It would have been good engineering practice to design the CV with some corrosion allowance to provide added assurance of its integrity over the full 60 years of operation. However, it has to be recognised that had this been done the vessel would have been even heavier and it would have been necessary, under the ASME code, to undertake post weld heat treatment of all the welds which would have been difficult and time consuming. Westinghouse has therefore opted to argue that:

a) corrosion should not occur because of the protective coating;

b) if it were to occur it would be slow, and localised and therefore tolerable; and

c) it would also be detected and repaired.

491 There is sufficient evidence to support the claim the zinc oxide coatings will protect the vessel and if this were to fail that the corrosion rate would be relatively slow - <0.3mm/year. I also accept the argument that corrosion, if it were to occur, is likely to be localised. Material corroding at the fastest predicted rate of 0.3mm/y would have pits of a depth of only 1mm at the end of the 40 month inspection interval. Given that this is also likely to be localised I accept the judgement that localised corrosion of this depth will have negligible structural impact for a vessel with a minimum thickness of 41.3 mm.

492 Concern has been expressed that the CV could become perforated due to deep, undetected, corrosion as was observed in the steel lining of the Beaver Valley concrete

pressure vessel. From my understanding of this event there is a significant difference between the Beaver Valley containment liner and the AP1000 Containment Vessel. The Beaver Valley liner is a pressure vessel liner rather than a pressure vessel, with no structural function. It is therefore much thinner; 9.5mm compared to a minimum of 41.3 mm in the AP1000 CV. Thus it would be much less likely that corrosion could progress undetected to the point of perforation in the AP1000 CV.

493 I am therefore satisfied that the CV is unlikely to suffer from significant corrosion throughout the life of the plant provided the Licensee:-

- Carries out regular and effective inspections of the coating. This is addressed by Assessment Finding **AF-AP1000-SI-26**.
- Demonstrates that the protective coating is capable of protecting the CV for much longer than the 5 days so far demonstrated. This is addressed by Assessment Finding **AF-AP1000-SI-27**.
- Properly applies and repairs the coating**.** This is addressed by Assessment Finding **AF-AP1000-SI-28**.

4.7.2.3 Loadings on the Containment Vessel

494 As stated above the design pressure for the containment vessel is 407kPa which is marginally above the maximum pressure that it would see following a double-ended cold leg guillotine failure or a main steam line failure at 30% power. The fault studies team have reviewed the calculations used to predict these pressures and are satisfied that they are adequately conservative.

495 Westinghouse has also calculated the ultimate capacity of the containment vessel, which is the pressure it could sustain following yield, and determined that at 38°C this would be a factor 2.63 greater at 1069kPa, at 204°C this is only 17% lower at 889kPa. This is clearly a significant additional margin which allows me to be content with the small margin between the predicted maximum pressure and the design pressure.

496 The CV is a sealed vessel with a maximum allowed leakage of 0.1% of the air per day therefore if the pressure were to equalise to atmospheric pressure at the maximum operating temperature of 49°C and then the temperature were to drop to the minimum operating temperature of 10°C this would result in a calculated negative differential pressure of about 20kPa. This is greater than the design negative pressure of 11.7kPa. In order to protect the vessel against this scenario automated vacuum relief valves will be introduced under a category 2 change (Ref. 173) to ensure that this pressure is not exceeded. The design of these valves has been reviewed by the Mechanical Engineering team and found to be in line with expectations.

497 At only 44.4mm thick and nearly 40m in diameter the containment vessel is a thin wall vessel for which the possibility of buckling needs to be considered. Particular concerns were negative pressure as described above and the weight of the polar crane especially during critical lifts.

498 To assist me with my assessment Westinghouse provided me with two reports describing their stability analysis. The first of these (Ref. 174) used the eigenvalue technique in accordance with ASME Code Case N284-1 to assess the stability of the CV under a range of loads and combinations of load including seismic loads, negative pressure as

well as parking and critical lift loads on the polar crane. In all cases these were comfortably within the allowed limits.

499 Westinghouse also considered the benefit of adding a further stiffener below the bottom of the polar crane girder (Ref. 175). In this report they concluded that even for a negative pressure of about 20kPa, compared with the design negative pressure of 11.7kPa and with the polar crane either parked or undertaking a critical lift this was not necessary. From these results I conclude that the containment vessel is adequately secure against buckling.

500 The weight of the polar crane is supported through the girder which is attached to the containment vessel. This will therefore subject the containment vessel wall to significant loads in the through wall direction and thus I considered that there could be a risk that the vessel wall will be subject to lamellar tearing. If this were to occur it could result in the failure of the polar crane support. I therefore asked Westinghouse in TQ-AP1000-1213 (Ref. 9) to explain what steps they had taken to reduce this risk to an acceptable level.

501 Westinghouse provided a full response to this TQ which identified a series of measures they had undertaken to minimise the risk. These included vacuum degassing of the steel when molten which ensures that the sulphur levels are low. A sulphur level of 0.001% weight is required which is well below 0.005% weight recommended by both the American Welding Society and TWI to reduce the risk of lamellar tearing.

502 Double sided weld welds are also used since balanced welding reduces the stress and therefore the risk of tearing. Overall I am satisfied that Westinghouse has taken sufficient steps to reduce the risk of lamellar tearing.

503 In TQ-AP1000-732 (Ref. 9) I asked Westinghouse about the thermal stresses generated in the containment vessel when the water of the PCCWSR flows over it to provide cooling following a large LOCA. Westinghouse confirmed in their response to this TQ and to follow up TQ-AP1000-1040 that it was being investigated and that they expected to have this work complete by December 2010. Unfortunately this work, which forms part of the normal design finalisation process, was not available at the end of December 2010 when the assessment phase of GDA Step 4 finished.

504 During subsequent discussions Westinghouse set out the intended scope of the analysis which I understand will initially consider a worst case transient with the CV at it hottest design temperature 149°C and the water at the minimum temperature of 5°C with stresses calculated for the worst case flaw size location and orientation and conservative fracture toughness values. This appears to be an appropriate initial calculation but Westinghouse recognises that the results of this evaluation may not be acceptable and thus they may need to use more realistic bounding cases. Again this is a reasonable approach.

505 Westinghouse does not anticipate any problems reaching a successful conclusion with this analysis however this is an essential analysis to confirm that the CV will perform as an adequate barrier in the event of a design basis accident. Furthermore Westinghouse has not been able to provide me with evidence why they have high confidence that this analysis will be successful and thus it has resulted in Action 1 of GDA issue **GI-AP1000-SI-04**. Nevertheless I am satisfied that it is likely that Westinghouse will achieve an acceptable result therefore it would not be appropriate to withhold an IDAC. The complete GDA Issue and associated action(s) are formally defined in Annex 2.

4.7.3 Conclusions and Findings Relating to Containment Vessel Shell

506 The majority of the containment vessel (CV) is fabricated from plate which is just thin enough not to require post weld heat treatment under the recently revised relevant ASME code. In the UK there would be a requirement to demonstrate that this was not necessary through a fracture analysis. Currently I have not received adequate arguments to support the claim that post weld heat treatment is not required and I will pursue this under GDA Issue Action **GI-AP1000-SI-04.A2**.

> ***GI-AP1000-SI.04.A2:*** *Provide sufficient evidence to show that the containment vessel has adequate tolerance to small defects in the absence of post weld heat treatment.*

507 The CV will be subject to thermal loads when water flows over it from the PCS. These stresses had not been calculated during the Step 4 assessment period and will be assessed under GDA Issue Action **GI-AP1000-SI-04.A1**.

> ***GI-AP1000-SI.04.A1:*** *Provide sufficient evidence to show that the containment vessel has adequate tolerance to the thermal shock due to the flow of PCS water onto the top head .*

508 The complete GDA Issue and associated action(s) are formally defined in Annex 2.

509 I judge that the design conditions for the containment vessel appear reasonable but have not been proven, and the following Assessment Finding has been raised:

> ***AF-AP1000-SI-25:*** *The Licensee shall confirm that the containment vessel wall temperature does not rise above the design temperature in the event of a reactor coolant loop or main steam line failure or if it does justify that this is acceptable.*

510 Since this could result in a change to the equipment specification this should be completed prior to the procurement of the containment vessel. I have therefore linked this Finding to the generic milestone of long lead item and SSC procurement specifications.

511 I judge that the containment vessel should not suffer significant corrosion during operation provided that the proposed coating is adequately applied and inspected, and have raised the following Assessment Findings.

> ***AF-AP1000-SI-26:*** *The Licensee shall include planned periodic visual inspection of the CV, its protective coatings and the moisture barrier in its arrangements for periodic inspections. Particular attention should be given to the concrete embedment transition.*

> ***AF-AP1000-SI-27:*** *The licensee shall demonstrate the protective coating applied to the containment vessel is capable of protecting it against extended exposure to the potentially corrosive chemicals to which it may be exposed.*

> ***AF-AP1000-SI-28:*** *The Licensee shall include the guidance on coating application, repair of coating defects and the qualification of staff for application and inspection of coatings in its procedures and arrangements.*

512 **AF-AP1000-SI-26** will be required once the protective coatings and moisture barrier have been installed. I have therefore linked this Finding to the generic containment pressure test milestone as the protective coating and moisture barrier will have been installed by this stage.

513 **AF-AP1000-SI-27 and -28** will be required before the coating is applied as it would be difficult to make substantive changes once the coating had been applied, and that could

lead to substantial delays and additional costs. The coating will be applied after the CV has been fabricated but prior to its pressure test. I have linked the Findings to the generic containment pressure test milestone, but in practice they should be undertaken prior to applying the coating.

4.8 Documentary Envelope for Specific Components

514 This activity sets out to explore the hierarchy of documents that defines and justifies the construction of safety-critical components. The scope of assessment is outlined in AR09058-9 and by RO-AP1000-65 (Ref. 10).

515 RO-AP1000-65 explained that I intended to focus on those components for which gross failure is claimed to be so unlikely that it may be discounted and specifically the RPV, steam generators and pressuriser.

4.8.1 Generic and Site Specific Safety Related Documents for Primary Circuit Pressure Boundary components

516 RO-AP1000-65.A1 and RO-AP1000-65.A2 asked for a definition of the document hierarchy and access to a sample of the documents as requested. In response Westinghouse provided an overview of their document hierarchy, Ref. 78, a list of documents relevant to the RPV, and an offer of access to sample the information on request.

517 The document hierarchy is defined by the plant design criteria, ASME code requirements, and applicable regulatory requirements. This leads into various detail specifications to support the design, for instance the design specification, functional specification, manufacturing specification, material specification and testing specification. The component's design is then further detailed via design drawings, and various analyses and assessments. This includes transient analysis, finite element analysis, sizing calculations, failure modes and effect analyses etc and an ASME Code Report to demonstrate compliance with the applicable code requirements.

518 The overall documentation structure is maintained for each component as a 'released document' list to provide a complete list of the latest documents available for fabrication and identifies the latest revision, title and release date for each document.

519 Ref. 78 provided a document list for the RPV design to illustrate the type of supporting document available, and the list includes design and functional specifications, material and fabrication specifications, load and transient specification and code compliance reports.

520 I was satisfied that the documentation structure described in Ref. 78 should be a suitable basis for defining and justifying the construction of safety critical components, and the next stage was to request documentation lists for other components and sample the documents themselves.

521 However, the publication of the component safety reports Refs 17 to 27 essentially negated the need to request this additional information. The component safety reports (CSR) provide a summary of the safety case being submitted for each of main components being considered in the structural integrity assessment. As such the CSRs are written from the documentation structure available for each component, and provide a full index of the technical reports underpinning each component. (The CSRs are

incorporated into the PCSR as the Appendices to Chapter 20 of the draft PCSR, Ref. 13 and issued PCSR, Ref. 28).

522 The CSRs also provide an overview of the important design information contained in the supporting documentation, such as material specifications, design transient cycles, allowable stress margins, ASME code limits, and summary of fatigue usage factors.

523 I therefore reviewed the documentation structure and design information provided in the CSRs. My review provided me with a good understanding of the documentation structure and the important design information and I did not identify any significant anomalies. I did not therefore consider it necessary to undertake additional general sampling of the individual documents identified against the RPV steam generator and pressuriser in the context of the documentary envelope, and was content to move through to a deeper sample of the design documents discussed in the next section. However, I noted the following:

i) Westinghouse intends that the main design documents, for example the functional specification, the transient definitions, the stress analyses, the ASME code checks, should be generic for the AP1000. This is reflected in the documentation structure where there is no reference to site specific safety related documents. This is an important matter and something that the Licensee will need to manage carefully to ensure that the safety case for the individual site reflects the build and operation on that site.

ii) Several of the important design documents referenced in the CRS have open items associated with them and are subject to revision. This will be managed by Westinghouse through the 'released document' list which is maintained for each component. I chose not to sample these 'released document' lists, but it is clearly a matter will have to be managed carefully by any Licensee during the design, manufacture and construction phases.

4.8.1.1 Overview on Document Envelope

524 I judge that an adequate documentation structure is in place, and that a systematic process exists to define and justify the construction of the most important vessels. However, I have noted that main documents in the documentary envelope will be generic to the AP1000 rather than site specific, and that several are still subject to open items and revision. These matters will have to be managed carefully by the Licensee. I have therefore raised an Assessment Finding for the Licensee to ensure that the safety case for the individual site reflects the actual build and operation on that site and for the Licensee to show that the hierarchy of documents relevant to that each stage of the design and construction phase is in place before the work commences. This is addressed by Assessment Findings **AF-AP1000-SI-29** and **AF-AP1000-SI-30**.

525 A deeper assessment of a sample of design documents is discussed in the next Section.

4.8.2 Review of Design Reports

4.8.2.1 Background and Key Issues from Step 3 Assessment

526 This activity is AR09058-9.2 on the Step 4 Action Plan (Ref. 1) and was not the subject of a Regulatory Observation.

527 It is important that nuclear power plant is not only designed to appropriate codes but also that the code is correctly and accurately interpreted by the designers. It would not be appropriate for a regulator to systematically check every calculation that is made rather it is the expectation that the designer has suitably qualified and experienced staff and appropriate procedures to ensure that the design complies with the chosen design codes. Nevertheless given the importance of “getting the design right” I decided to check a sample of the design calculations for the most safety significant steel components to ensure that the design of the AP1000 complies with the claimed version of the ASME, generally 1998 with 2000 addenda (Ref. 30).

528 Much of the design of large pressure vessels uses standard rules, which are specified in the Code, to calculate the wall thickness of the main vessels and also to determine the degree of reinforcement required around nozzles. I therefore asked a specialist TSC (EASL) to a check on the accuracy of these calculations for the vessel wall and for selected nozzles in the RPV and the pressuriser.

529 Where the structure is more complex, such as the main inlet and outlet nozzles in the RPV, this “Design by Rule” approach is not appropriate and it is therefore necessary for Westinghouse to use a finite element model to predict the stresses. Setting up and running such models is very labour intensive and typically requires many man-months of work and given that these models are not novel I decided that it was not necessary to repeat these calculations. Nevertheless it is important that appropriate models and input data are used. I therefore asked EASL to review the approach Westinghouse had used for the for finite element analysis of the RPV inlet and outlet nozzles.

4.8.2.2 Step 4 Assessment

530 The results of the EASL review are reported in (Ref. 179) and are discussed in more detail below. In general these reports were difficult to follow and this resulted in a large number of comments which ranged from identifying areas where clarity could be improved, through the report not being clear enough to confirm that the conclusions were correct to significant errors being identified in the “Design by Rule” calculations. I therefore raised TQ-AP1000-1290 (Ref. 9) which asked Westinghouse to provide a response to these comments.

4.8.2.2.1 AP1000 Reactor Vessel Sizing

531 Westinghouse report “AP1000 Reactor Vessel Sizing Calculation” (Ref. 196) systematically reviews the wall thickness and reinforcement areas of the RPV and confirms that they are larger than required by the ASME code. This is a substantial report which summarises results from supporting documents and I therefore decided to ask EASL to review a sample of the calculations presented in the report; these included the calculations for the closure head, lower head, main shell, the head and vessel flanges and the nozzles..

532 EASL also found calculations presented to confirm the thickness of the flange impossible to follow and therefore was not able to confirm that these were correct although he found no evidence to suggest that they were incorrect. Because this review was not completed until after the end of the assessment phase of GDA it has not be possible to engage in detailed discussions with Westinghouse to allow them to explain their approach.

533 The inlet and outlet nozzles on the RPV are relatively close together considering their size and it appears that the claimed reinforcement areas for adjacent nozzles may

overlap. The ASME code does not allow this therefore it is important to demonstrate that this has not occurred, and this was not evident in Ref. 196. In response to TQ-AP1000-SI-1290 Westinghouse advised that they will check this and include this evaluation in the next revision of the RPV sizing calculation.

4.8.2.2.2 Detailed and Transient Analysis of RPV Inlet Nozzle

534 Westinghouse report "Detailed and Transient Analysis for AP1000 RPV Inlet Nozzle," (Ref. 197) is a calculation note which describes the analysis of the RPV inlet nozzle against the ASME "design by analysis" requirements. Hand calculations are used to determine acceptability for primary stresses from external piping reactions and internal pressure and the ANSYS general purpose finite element program is used to determine nozzle acceptability for primary plus secondary stresses and fatigue from the combination of transient thermal loading, external loads applied to the RV, external piping reactions, and RV support loads (including deadweight).

535 As explained above it was not considered necessary to repeat the finite element calculations performed by Westinghouse so the review concentrated on whether the approach used was appropriate.

536 This report takes as an input loads those defined in the RPV Design Specification (Ref. 160) and does not justify these. In a number of cases approximations were made which were not obviously conservative and these were questioned. In their response to TQ-AP100-SI-1290 (Ref. 9) Westinghouse argued that these approximations had a negligible effect on the final result. This appears to be reasonable but will be subject to further review through GDA Issue Action **GI-AP1000-SI-05.A1**.

537 Hand calculations were used to evaluate the nozzle for primary stress code compliance by considering the combination of internal pressure, external loads from the attached piping, and the support pad loads. The report identifies two sections where these calculations were performed; one in the reinforced area and a second in the unreinforced area of the nozzle. For the latter the section chosen is in the stainless steel safe end rather than the slightly thinner ferritic section which also has a much longer lever arm. This stainless steel section was chosen because it has a much lower strength however it was not clear from the report that this necessarily out weighed the greater stress and therefore it was recommended both sections should have been assessed. In response Westinghouse advised that they had indeed performed calculations on both sections but only the bounding section was included in the report.

538 Stresses and fatigue loads are calculated and reported at a set of distinct sections through the reinforced region of the nozzle and also the safe end region. These regions (20 in total) were selected because they were judged to have the highest stresses. Our assessment noted that these did not include any section at the root of the nozzle where the reinforced section intersects the vessel wall; an area where in the experience of EASL the highest nozzle stresses normally occur. Westinghouse advised that in a later version of this report sections in this area had been added.

539 The fatigue assessment was performed using the ANSYS fatigue module which gives very limited visibility and control of the calculations which are performed and thus can lead to the actual analysis being different from that assumed to have been carried out. In our view unexpectedly high utilisation factors have been calculated at some locations and Ref.179 provides a number of suggestions as to how this may have occurred. It is clear from the response provided by Westinghouse that they were aware of the potential

problems with this module and have taken steps to avoid them. In my view it is much more likely that, if there are errors in these calculations, they will have resulted in over predicting than under predicting the fatigue utilisation factors and since these are much less than unity they are not a concern.

540 The EASL report (Ref. 179) contains a number of detailed comments on the approximations used in the analytical and finite element model. In response Westinghouse advised that the consequences of these approximations were negligible.

4.8.2.2.3 Detailed and Transient Analysis of RPV Outlet Nozzle

541 Westinghouse report "Detailed and Transient Analysis for AP1000 RPV Outlet Nozzle" (Ref. 198) is a calculation note which describes the analysis of the RPV outlet nozzle against the ASME "design by analysis" requirements. This report is basically the same as the inlet nozzle report reviewed above and resulted in similar comments.

4.8.2.2.4 Pressuriser Sizing Calculation

542 To ensure that my sample extended beyond the RPV I also selected the AP1000 Pressurizer Sizing Calculation Report (Ref. 199) for review by EASL. I originally identified revision D of this report since this was referenced from the AP1000 Pressuriser ASME Code Stress Report (Ref. 200) which is in turn a first tier reference to the pressuriser CSR (Ref. 21). However I was advised that the latest available version was revision B so I chose to review this revision.

543 The main purpose of this report was to determine whether the thickness of the main parts of the pressuriser and the reinforcement around the nozzles complies with the "Design by Rule" requirements of ASME Section III. The calculations were carried out using the MathCad package and are set out in the report.

544 In our review of the calculations a number of significant errors were identified related to not properly understanding a number of key features of the MathCad package. The most significant consequence of these errors was that the calculated available reinforcement area for the safety relief nozzles was 30% greater than was actually available and this led to the conclusion that there was sufficient reinforcement available when in fact there was not.

545 In their response to TQ-AP1000-1290 (Ref. 9) Westinghouse advised that during the period in which Ref. 199 was being reviewed they had revised the report and issued it at Revision 0 (Ref. 201). This report is, in practice, not a revision to correct errors and improve clarity rather it is in fact a repeat of the calculations starting from scratch and using Excel for the calculation rather MathCad and a complete rewrite of the report.

546 The required and available reinforcement area for each of the nozzles calculated in Ref. 201 differ significantly from those calculated in Ref. 199 and correctly identifies that there is insufficient reinforcement for the safety relief nozzle according to the ASME "Design by Rule" requirements.

547 Due to Ref. 201 not being available during the assessment phase of GDA I have not been able to subject it to a full assessment however, I have undertaken a high level review and checked some of the calculations contained within it. The report sets down very clearly the calculations performed in a manner which systematically follows the

ASME methodology and I therefore have confidence that the correct approach has now been followed. I have also confirmed that accuracy of a number of the calculations.

548 The ASME code recognises, in clause NB 3331(c) that the “design by rule” approach to determine the reinforcement around nozzles is conservative and that it is acceptable to demonstrate the adequacy of a proposed design by a more sophisticated analysis of the stresses around the nozzle. Westinghouse has confirmed the design using this approach thereby justifying the designs compliance with the ASME code.

4.8.2.3 Conclusions from Step 4 Assessments of Design Reports

549 EASL found the RPV reports difficult to follow and this resulted in a significant number of comments because they were unable to confirm that “Design by Rule” compliance had been demonstrated. It is clear from the responses provided to specific comments and discussions with Westinghouse that some of the statements in the reports which appear to be assertions are actually supported by calculations not evident from the report.

550 Westinghouse has provided a full and thorough response to the comments raised in Ref. 179 and based on my high level review of these responses I judge that it is likely that Westinghouse will be able to provide adequate responses to all the comments.

551 Although this review has identified specific concerns about the “Design by Rule” calculations these vessels are also subject to a “design by analysis” calculation which provides a diverse confirmation of the adequacy of the design. Of course in the rare cases that “design by rule” compliance is not demonstrated the adequacy relies solely on “design by analysis”.

552 The pressuriser sizing calculation report that was subject to EASL’s review was only at Revision B which, because this is a letter rather than a number, means that the report did not have the status of a finalised design report nevertheless it was a fully verified report and thus the presence of significant errors is a concern which will be pursued through GDA Issue Action **GI-AP1000-SI-05.A2**.

553 Based on my high level review the main concerns about the RPV design reports appear to have been adequately answered or a least capable of being adequately answered and the latest revision of the pressuriser sizing calculation report appears to be both complete and accurate. Therefore I judge that Westinghouse will be able to justify that their design complies with ASME and thus my residual concerns are not sufficient to stop me supporting an IDAC. Nevertheless it will be necessary for me to fully assess the response to TQ-AP1000-SI-1290 and Ref. 201 before I am able to support a DAC. I have therefore raised GDA Issue Action **GI-AP1000-SI-05.A1** to enable me to do this.

4.8.3 Findings Relating to Documentary Envelope

554 On the basis of the evidence from the reports sampled I am satisfied that an adequate documentation structure exists and there is a systematic process to define and justify the construction of the most important vessels, but I have raised two Assessment Findings against the overall document envelope:

> ***AF-AP1000-SI-29:*** *The Licensee shall ensure that the safety case for the structural integrity components on the individual site reflects the actual build and operation on that site*.

555 This Assessment Finding shall be completed before the generic milestone of hot operations as the structural integrity case should be in place prior to hot operations.

> ***AF-AP1000-SI-30:*** *The Licensee shall demonstrate that, for each stage of the procurement, manufacturing and construction process, the hierarchy of documents for the structural integrity components relevant to that stage is in place before the work commences.*

556 This Assessment Finding shall be completed before the generic milestone of RPV installation. This is because it would be extremely difficult to make substantive changes once the components start to be installed which could then lead to substantial delays and additional costs. In practice the finding will need to be completed at various times to suit the construction programme.

557 The review of the Westinghouse Design reports raised concerns about the adequacy for the documentation to demonstrate that the design complies with the ASME code. The response to the concerns raised is sufficient to allow me to support an IDAC however it is still necessary to thoroughly assess the responses provided and the new report before I am able to support a DAC, I also need to have evidence to confirm that Westinghouse's report preparation and review processes are adequately robust. I have therefore raised GDA Issue **GI-AP1000-SI-05** which has two actions.

> ***GI-AP1000-SI-05.A1:*** *Provide Response to Findings raised in "Review of Stress Analysis in the AP1000" Report.*

> ***GI-AP1000-SI-05.A2:*** *Provide evidence that there will not be similar errors elsewhere in the design support documentation.*

558 The complete GDA Issue and associated action(s) are formally defined in Annex 2 of this report.

4.9 Generic Categorisation and Classification

4.9.1 Background

559 Allocating an appropriate classification to pressure equipment and tanks is an important part of the structural integrity safety case as it defines the design, construction, inspection and through life maintenance of the component. Section 4.1 of this report discusses Categorisation and Classification in terms of identifying components whose likelihood of failure is so low that it can be discounted, i.e. identifying the HSS components, whereas this section considers classification in generic terms of how design codes and standards are allocated to a wider range of nuclear safety significant pressure equipment and tanks.

560 ND's SAP ECS.3 (Ref. 4) states that systems, structures and components that are important to safety should be designed, manufactured, installed, commissioned, quality assured, maintained, tested and inspected to the appropriate standards, and SAPs EMC.1 to EMC.34 identify the SAPs underpinning the structural integrity of metal components which implicitly require an appropriate classification.

561 Westinghouse has adopted a UK specific classification scheme, the UK AP1000 classification scheme and the methodology is described in Ref. 81. The scheme is a development of their global approach to Categorisation and Classification described in the European Design Control Document, Ref. 15, but has been modified to make it compliant with ND's SAPs. The application of the methodology down to an individual system/component level is shown in Ref. 77.

4.9.2 UK AP1000 Classification Scheme – Design Codes and Standards

562 Ref. 81 describes the methodology for applying Categories and Classes to systems. Structures, Systems or Components (SSC) are classified according to their contribution to a safety function in line with the SAP ECS.2, and are allocated to Class 1, 2 or 3. Ref. 81 then indentifies which codes and standards would be applied to each class of component.

563 My initial assessment of Ref. 81 led me to believe that the approach met my requirements in terms of the design codes and standards applied to pressure vessels and tanks, and indeed Ref. 81 is considered to be compliant with the SAPs in other technical topic areas. However, very late in the assessment process I became aware that the application of the methodology, Ref. 77, differed from how I had anticipated the methodology would be applied. Nuclear specific codes have been applied to Class 1 pressure equipment and storage tanks but non-nuclear standards have been applied for Class 2 pressure equipment and storage tanks.

564 The supporting paragraphs to SAP ECS.3, paras 157-161, clarify that codes and standards should reflect the functional reliability requirements of the structures, systems and components and be commensurate with their safety classification. Codes and standards should preferably be nuclear specific codes and standards, but Class 3 components may use appropriate non-nuclear specific codes. In the case of pressure equipment and storage tanks there are nuclear design and construction codes available in the form of a number of nuclear specific codes, for example ASME III. Thus applying non-nuclear codes for the design and construction of Class 2 pressure equipment and tanks does not meet ONR's normal expectations for pressure equipment and tanks.

565 Whilst this does not meet our normal expectation, I believe it may be possible for Westinghouse to provide appropriate claims, arguments and evidence to justify the use of non-nuclear codes. Thus where non-nuclear pressure equipment and storage tank design and construction codes are used in the design of Class 2 components Westinghouse will need to fully justify each case to show the arguments and evidence which support the use on non-nuclear codes. The arguments and evidence should take account of the safety significance of the component; the demands that are placed on the system in terms of loadings, fatigue, temperature etc and the consequences of failure of pressure boundary in terms of both the loss of system function and on the Internal Hazards safety case.

566 As I only became aware of this problem very late in the assessment process, Westinghouse was unable to provide the necessary supporting evidence for this stage of the assessment process. I have therefore taken the matter forward as Action 1 of GDA Issue **GI-AP1000-SI-06** for Westinghouse to provide evidence to show that the principal design and construction codes adopted for Class 2 Pressure Equipment and Storage Tanks are consistent with ND's expectations as detailed within the SAPs. It is important to note that WEC's approach is considered to be consistent with the SAPs in other technical topic areas, and that this Issue relates to pressure vessels and tanks only.

4.9.3 Allocation of ASME III Classes

567 The AP1000 standard classification scheme describe in the AP1000 European Design Control Document (Ref. 15) allocates Equipment Classes according to a set of defined criteria and is guided by standards such as ANS-51.1-1983 (Ref. 82). These Equipment

Classes link to the three ASME III classes used for the design and construction of pressure equipment and tanks, Class 1, Class 2 and Class 3.

568 I reviewed this document and noticed that the Accumulator Tanks in the Passive Core Cooling System are assigned an Equipment Class C and are therefore designed and constructed to ASME III Class 3 requirements. However the ANS classification for pressurised water reactors defined in ANS-51.1-1983 (Ref. 82) states that Accumulator Tanks are ANS Safety Class 2 and should therefore be designed to ASME III Class 2 requirements.

569 I recognise that the AP1000 has required an adaptation of the previous safety classification standards because of AP1000's passive approach to safety. It is this adaptation that has lead to the Accumulator Tanks in the Passive Core Cooling System being allocated an AP1000 Equipment Class C, and therefore that they will to be designed and constructed to ASME III Class 3. Unfortunately there is no explanation behind the adaptation or justification of why the adaptation can downgrade the core cooling system compared with the classification applied in previous PWR designs.

570 Again I only recognised this problem very late in the assessment process and Westinghouse was unable to provide the necessary supporting evidence for this stage of the assessment process. I have therefore taken the matter forward as Action 2 of GDA Issue **GI-AP1000-SI-06** for Westinghouse to provide evidence to show that components in AP1000 Equipment Class C have been assigned a class that is consistent with their intended duty and implied reliability.

4.9.4 Conclusions on Generic Categorisation and Classification

571 In other topic areas ND is satisfied that the UK AP1000 classification scheme is compliant with the SAPs, but late in the assessment process I identified two areas where the classification of pressure equipment and tanks needed further justification; the use of non-nuclear design and construction codes for Class 2 pressure equipment and tanks; and the use of ASME III Class 3 for the design and construction of the Accumulator Tanks in the Passive Core Cooling System.

572 As these problems were only recognised very late in the assessment process Westinghouse was unable to provide the necessary supporting evidence for this stage of the assessment process. I have therefore taken the matter forward as Action 1 and 2 of GDA Issue **GI-AP1000-SI-06** for Westinghouse to provide the necessary evidence to support their arguments.

GI-AP1000-SI-06: *Structural Integrity Categorisation and Classification. The key activities which will need to be completed under Action 1 and 2 of this GDA Issue are to:*

- *Provide evidence to show that the principal design and construction codes adopted for Class 2 Pressure Equipment and Storage Tanks are consistent with ND's expectations.*
- *Provide evidence to show that components in AP1000 Equipment Class C have been assigned a class that is consistent with their intended duty and implied reliability*.

573 In addition a problem was found with the evidence to support the allocation of a standard Class 1 structural integrity classification for the reactor coolant pump bowl in Section

4.1.2 of this report, and this is being taken forward through Action 3 of GDA Issue **GI-AP1000-SI-06** and the key activity against this action is to:

- *Provide arguments and evidence to show that catastrophic failure of a reactor coolant pump bowl would not challenge the effectiveness of the vertical support for the Steam Generator*

574 The complete GDA Issue and associated action(s) are formally defined in Annex 2.

4.10 Review of Access for In-Service Inspection

4.10.1 Background and Key Issues from Step 3 Assessment

575 This activity AR09058-10 on the Step 4 Action Plan (Ref. 1). This activity continues the assessment which followed from Step 3 Regulatory Observation RO-AP1000-66.

576 The Safety Assessment Principles (Ref. 4) recognise the important role in-service inspection (ISI) plays in confirming that a structure is free from significant defects. For example EMC 27 states:-

"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"

577 ISI, by its nature, will not be carried out until the plant enters service and thus it is the responsibility of the Licensee rather than the Requesting Party to ensure that it is fit for purpose. For the most important components I expect this to be achieved through an ENIQ qualification of the planned inspection. ND will therefore consider the details of the ISI during the site licensing phase and not during GDA. Nevertheless the design of the plant can have a significant effect on the capability of future ISI so within GDA I need to be satisfied that with the proposed plant design adequate ISI is likely to be possible. Additionally I wish to ensure that there are no ALARP modifications to the design which could be made to ensure that ISI had sufficient capability.

4.10.2 Step 4 Assessment

578 Westinghouse systematically reviewed the design requirements for the ASME Class 1 components in the AP1000 in Ref. 177. In this report there is a clear statement that:

'*Owners of commercial power facilities have regulatory commitment to perform periodic inservice inspection (ISI) of their plants. IWA-1400(b) and IWA-1500 of the ASME Code Section XI define the owner's responsibility to ensure that adequate design and access provisions are incorporated in order to meet this commitment. For the AP1000, Westinghouse provide the design and access implementation of inservice inspection.'*

Strictly compliance with ASME Section XI is not sufficient within the UK to confirm that the proposed in-service inspection is adequate since there is a need to demonstrate its adequacy in the safety case. This is recognised in the CSRs for the HI and HSS components (Refs 17, 21, 22 and 27) which contain a forewarning of failure leg which requires a qualified in-service inspection.

579 Ref. 177 also sets down the concepts, design philosophy and goals for ensuring that the design takes good account of the need for inspectability which appear to be reasonable. They are defined as follows:-

'CONCEPTS AND DESIGN PHILOSOPHY

The basic concepts and design philosophy related to ISI of AP 1000 components are as follows:

- *The primary goal is to maximize the inspectability. That is, inspection equipment/ personnel access and the component design conform, with current inspection technology and strategies.*
- *The design will reflect the capability to employ the latest proven examination techniques regardless of ASNII; Code-endorsed techniques in anticipation that the Code will ultimately incorporate such advances.*
- *The design will accommodate as much as reasonably possible, newly emergent technologies consistent with the finalization of the design.*
- *The design will comply with requirements considered necessary to achieve API000 objectives.*
- *ISI design requirements will not result in a loss of the inherent reliability of a component design merely to satisfy an inspectability requirement. When a conflict between these two desirable attributes is identified. A decision will be made as to which criterion is to prevail. Plant safety is the ultimate judgment criterion.*
- *ISI design requirements and their implementation will be coordinated with other important design considerations.*
- *A basic tenet of the AP 1000 inspectability process design is to assure that ISI design decisions are made in-process and are not an after-the-fact reconciliation of the design to the requirements.*

GOALS

The emphasis is on how ISI is affected by the design and how the design can be formulated to make these inspections more reliable. Factors such as examination requirements, examination techniques. Accessibility, component geometry, and material selection are used in evaluating the component designs-and are referred to as the ISI design evaluation factors. The goals of applying ISI design evaluation factors are as follows:

- *Eliminate non-inspectable components to the maximum extent reasonable.*
- *provide adequate accessibility.*
- *Reduce personnel radiation exposure following ALARA principles.*
- *Reduce inspection times and costs.*
- *Allow the use of state-of-the-art inspection systems and methodologies.*
- *Provide for enhanced flaw detection and characterization reliabilities.*
- *Provide for added margins of safety.*
- *Minimize plant downtime for inspection.'*

580 In this report the ASME code requirements, the likely inspection techniques required and any accessibility issues are considered for each of the main welds. Westinghouse confirmed, in response to TQ-AP1000-941 (Ref. 9), that this accessibility review was not simply against mandatory ASME access requirements rather they '*…applied the essential parameters (beam angles, probe sizes, required access, etc.) of existing qualified*

inspection systems (in accordance with the PDI Appendix VIII program).' They noted that for strict compliance with this program they would need to extend the qualification to the component sizes found in the AP1000. Nevertheless it is reasonable to assume that this review is likely to identify any potential access problems.

581 This work leads onto a second phase which was to carry out a more thorough review of each component identifying in more detail which inspection techniques are likely to be used, what their access requirements would be and any design modifications or constraint are required. Each report identified a set of "Design/Fabrication Actions" and a set of "Pre- PSI/ISI Implementation Actions"

582 In total six reports were prepared for the ASME Class 1 components and as a sample I decided to carry out an "in depth" review of the report prepared for the RPV (Ref. 178). The report prepared for the RPV was both systematic and thorough and identified eight actions to be carried out during design or fabrication. These all appeared to be relevant and valuable, typically they required extra restrictions on the design drawings to ensure adequate surface flatness or finish or sufficient clearance for inspection equipment.

583 A further eleven actions were identified to be carried out pre PSI/ISI. Again these appeared to be relevant and valuable and typically they were reminder to future Licensees to a) invoke ASME Code cases; b) ensure appropriate specimens were prepared; and c) develop special equipment for constrained inspections.

584 During my review I raised a few questions of clarification under TQ-AP1000- 940 and received satisfactory responses to each. The most significant response was confirmation that where the need to develop special equipment was identified Westinghouse's judgement was that this equipment could be developed and in fact had been for their Chinese customer.

4.10.3 Conclusions and Findings for Access for In-Service Inspection

585 Based on the sample of the reports I reviewed I judged that Westinghouse had an adequate process in place to confirm that the HSS and HI welds could be adequately inspected during PSI and ISI, and I did not need to extend my sample. A key outcome of this in-house review are the actions identified in the ISI Inspectability Reports it is therefore essential that the Licensee ensures that these actions are either completed or the issue addressed in an alternative way. This requirement has been captured in two Assessment findings **AF-AP1000-SI-31** and **AF-AP1000-SI-32** which are findings relating to access for in-service inspection.

> ***AF-AP1000-SI-31**. The Licensee shall ensure that all the Design/Fabrication Actions in the ISI Inspectability Reports are either completed, or the issue addressed in an alternative way.*

586 This Assessment Finding shall be completed before the generic milestone of RPV Installation. This is because it would be extremely difficult to make substantive changes once the components start to be installed which could then lead to substantial delays and additional costs..

> ***AF-AP1000-SI-32**. The Licensee shall ensure that all the Pre PSI/ISI Actions in the ISI Inspectability Reports are either completed, or the issue addressed in an alternative way.*

587 This must be complete prior to PSI and I have linked this to the Cold Ops generic milestone since this it first milestone which is clearly after PSI.

588 In-Service Inspection is a significant leg of the safety case for each of the HSS and HI welds and it is therefore necessary for the Licensee to develop inspection procedures for each of these after GDA. These will need to be technically justified. This is addressed by Assessment Finding **AF-AP1000-SI-33**:

> ***AF-AP1000-SI-33**. The Licensee shall prepare justified PSI/ISI inspection procedures for each of the HSS and HI welds*

589 This must be complete prior to PSI and I have linked this to the Cold Ops generic milestone since this it first milestone which is clearly after PSI.

4.11 Operation of Plant within Safe Limits

4.11.1 Background

590 This is a new topic for Step 4 which was taken forward under a cross-discipline Regulatory Observation RO-AP1000-094 (Ref. 10). In this RO Westinghouse was asked provide clearer visibility on how the key limits and conditions embedded within the safety case will translated into the operating and maintenance documentation.

591 Westinghouse provided a response to this RO in their letter WEC00446N Ref. 193) however, this response was provided very late in the assessment period and was driven mainly by US Regulatory criteria and Standards and not derived from safety analyses. For this reason it did not meet ND's expectations and therefore GDA Issue **GI-AP1000-CC-01** has been raised. This is described in more detail in the AP1000 Cross-Cutting Topics Step 4 Report (Ref. 194)

4.11.2 Operating Limits Set by Structural Integrity Considerations

592 The Structural Integrity CSRs make explicit claims on the operational arrangements in the following areas.

- In-service Inspection on the main steel components.
- Inspection of the protective coating on the containment vessel.
- Monitoring of the RPV P-T limits.
- Inspection of irradiation embrittlement specimens.
- Leak detection.

These are reviewed in more detail below.

593 Accessibility for in-service inspect was within the scope of GDA and is covered in Section 4.9 above. However the proposals for the scope and frequency of in-service inspection are outside the scope of GDA and therefore have not been provided within GDA. Nevertheless in-service inspection is a major claim in the safety cases for all the HSS and HI welds and it is expected that that the Licensee will develop adequate proposals. This is addressed by Assessment Finding **AF-AP1000-SI-33.**

594 The integrity of the protective coating on the on the containment vessel is important to ensure that it does not degrade in service. My review of its safety case as described in Section 4.7 above and Assessment Findings **AF-AP1000-SI-26** to **AF-AP1000-SI-28** resulted from this review.

595 The importance the RPV PT Limit curve was discussed in some detail in Section 4.6 above. It is essential the Licensee ensures that this limit is complied with during heat up and cool down of the RPV and Westinghouse has proposed LCO 3.4.3 in the EDCD (Ref. 15). I share the judgement that a LCO is required and expect the Licensee to propose one similar to that in Ref. 15.

596 Section 3.4.3 of the RPV CSR (Ref. 17) describes a Materials Surveillance Programme to confirm the irradiation shift in the properties of the RPV. As described in Section 4.4.3 above I have agreed the principle of the surveillance scheme set out in this CSR but the details of the scheme have not been reviewed since they are more appropriately defined by the Licensee and are therefore not within the scope of GDA.

597 A future Licensee of an AP1000 will need to put arrangements in place for detecting leaks from the plant and managing these effectively. This is important for two main reasons. Firstly leaks can result in component being exposed to fluids which they are not compatible and can cause degradation. For example the water in the reactor coolant loop is acidic and therefore has the potential to dissolve ferritic components particularly if it is concentrated through boiling. Secondly, as noted in the CSRs (Refs 17-27), leaks can, in certain circumstances, provide early warning of minor failures which could progress to a more significant failure. It is recognised that the safety case for a UK AP1000 does not rely on a formal “leak before break” argument nevertheless the detection of leaks does provide a valuable defence in depth which I would expect a prudent operator to embrace. I have raised a finding to capture this need. This is addressed by Assessment Finding **AF-AP1000-SI-34**.

In addition to the explicit safety case claims on operational arrangements discussed above there are also a number areas were there are implicit claims in the safety case. These include ensuring that the temperature, pressure, chemical environment and loadings (both transient and fatigue) are as assumed in the design.

I expect the Licensee’s arrangements to capture these and GDA Issue Action **GI-AP1000-CC-01.A2** captures the need for the RP to provide guidance on how these should be set.

4.11.3 Findings Relating to Operation of Plant within Safe Limits

598 The following Assessment Finding has been raised:

> ***AF-AP1000-SI-34:*** *The Licensee shall set up suitable arrangements to ensure that leaks are reliably and promptly detected and subsequently managed.*

599 This needs to be complete before the generic milestone of Hot Operations. This is because these arrangements should be in place before the plant enters the operational phase.

4.12 Review of the Welding procedures

4.12.1 Background and Key Issues from Step 3 Assessment

600 This is activity AR09058-12.1 on the Step 4 action Plan (Ref. 1) and is a new activity for Step 4.

601 The AP1000 has been designed and will be built in accordance with the requirements of the ASME Code (Ref. 30). Within the UK ND have generally been comfortable with the requirements placed on welding set down in ASME Section III.

602 It is important that nuclear power plant is not only designed to appropriate codes but also that the code is correctly and accurately interpreted by the designers and that the fabricators are supplied with sufficiently detailed procedures to enable them to fabricate the plant to meet the designers and the codes intent. I therefore asked a contractor to undertake a systematic review of a few equipment specifications and weld procedures.

4.12.2 Step 4 Assessment

603 On reviewing the available documentation it became clear that the equipment specifications were at a relatively high level and that considerable reliance was placed on the fabricators to prepare detailed procedures compliant with ASME which would need to be approved by Westinghouse prior to their use.

604 This interpretation was confirmed in response to TQ977 (Ref. 9) but it was explained that Westinghouse did provide supplementary guidance (Ref. 159). I therefore asked TWI to review this document and the welding requirements in the equipment specification for the RPV (Ref. 160) which I judged to be the most detailed of the available equipment specifications.

605 TWI (Ref. 161) did not identify any areas where they judged these two documents to be contrary to UK practice however there were a number of areas where they judged both documents could have been more precise and these are discussed below.

4.12.2.1 Review of Supplemental Fabrication and Inspection Requirements

606 The following paragraphs set down the key points from the review of Ref. 159.

607 The need for hardness testing of austenitic stainless steel parent materials is recognised at a few places within this document and although the method of testing is not specified the "Definitions and Acronyms" section only defines Rockwell Hardness B. So by implication this method would be used and would be acceptable.

608 There is no mention of hardness testing of welds and HAZ; this is surprising since it provides a good indicator of susceptibility to hydrogen cracking. It is not clear from this document whether the expectation is that such requirements will be captured in the lower tier documents provided by the fabricators. Nevertheless it is important that the Licensee recognises that this is not required by the high level documentation. I have captured the need to consider hardness testing of the welds and HAZ as a finding **AF-AP1000-SI-35**.

609 Whilst the Rockwell Hardness B technique is appropriate for measuring the hardness of the parent material the size of the indenter makes it less accurate than alternative methods such a Rockwell C or Vickers hardness for measuring the hardness of welds and HAZ so the Licensee will also need to consider which method is most appropriate.

610 Ref. 159 permits the use of air carbon arc gouging, a technique which introduces the risk of carbon pick-up from the HAZ into the weld which would reduce the ferrite number and consequently increase the risk of solidification cracking. A second concern is that the gouging process, by its nature, introduces uncontrolled heat which may leave coarse grained material with low fracture toughness, adjacent to the cut surface. It is normal to remove this material using non-thermal means and this document requires that at least 1/32 inch (0.75 mm) material is removed is the material is subsequently welded and twice this if it is not to be welded. TWI (Ref. 161) judge that it would be prudent to remove more material; they suggest 3 mm.

611 I note the values in Ref. 159 are minimum values which might be increased by more stringent requirements in the fabrication specifications. Unfortunately fabrication specifications are site specific and were not available for GDA so I am unable to tell whether or not it normal practice to introduce more stringent requirements through this route. The Licensee should therefore confirm the requirements placed on non-thermal removal material after gouging are sufficiently conservative to ensure that the risk of cracking is acceptably low. This is addressed by Assessment Finding **AF-AP1000-SI-36**.

612 Ref. 159 sets the expectation that "extra low hydrogen" welding consumables are used for welding non-austenitic base metal with a specified tensile strength greater than 485 MPa; however it does allow higher hydrogen levels to be present in the consumables if they are qualified on test plates. High levels of hydrogen can result in hydrogen cracking and establishing the acceptability on test pieces cannot be 100% reliable. I therefore judge that this is not a desirable route to take on any HSS, HI and Class 1 weld. The Licensee should therefore ensure that either the fabrication procedure specifies "extra low hydrogen" welding consumables or, if this is not possible, a robust qualification process is specified. This is addressed by Assessment Finding **AF-AP1000-SI-37**.

4.12.2.2 Review of RPV Design Specification

613 Both Ref. 160 and also Ref. 159 require that a minimum pre-heat of 121°C is established for the P-3 group of material, this includes SA508, Grade 3 Class 1 which is the material used for the main forgings on the RPV. However neither these documents nor ASME III (Ref. 30) provide guidance on the method of preheat or the location for preheat measurement thermocouples which I anticipate will be provided in the fabrication procedure. For thick section forgings the appropriate placing of the thermocouples can be important to ensure adequate pre-heat so the Licensee will need to ensure that the measurement equipment, the thermocouple position and measurement time are specified. The international standard EN ISO 13916 (Ref. 195) provides a suitable model for this. This is addressed by Assessment Finding **AF-AP1000-SI-38**.

614 Ref. 160 requires the fabricator to either maintain the pre-heat after welding prior to going straight into post weld heat treatment (PWHT) or to increase the temperature to between 232°C and 289°C and maintain it at this temperature for at least 4 hrs before allowing the temperature to reduce. Post-heat reduces the weld metal hydrogen content and thereby reduces the risk of hydrogen cracking. Evidence presented in Ref. 203 shows that maintaining a 50 mm thick weld at about 300°C for four hours after welding will reduce the weld hydrogen content by 20% whereas if the weld were 100% thick this would only reduce the content by 5%. So the minimum time of four hours for post-heat is not applicable to all weld thicknesses and may not be sufficient for very thick welds. Therefore if the fabrication procedure allows a delayed PWHT it should specify a post heat temperature and dwell time which are adequate to ensure that the resulting weld metal hydrogen content is sufficiently low to avoid the risk of hydrogen cracking. I have not raised a specific assessment finding to capture this concern as it is adequately covered in the earlier, more generic finding. This is addressed by Assessment Finding **AF-AP1000-SI-19.**

4.12.3 Conclusions and Findings on Welding Procedures

615 Because the equipment specification and even the supplemental guidance are set at a relatively high level it has not been possible to confirm within GDA that the procedures

which will be used to control the welding will be suitable to assure weld quality and I have identified a number of Findings below which will need to be addressed prior to the generic milestone of RPV installation. This is because it would be extremely difficult to make substantive changes once the components start to be installed which could then lead to substantial delays and additional costs. Since these Findings require a tightening of the welding specifications rather than wholesale changes it is appropriate that these are addressed by a future Licensee. These findings do not preclude my support of a DAC.

AF-AP1000-SI-35: *The Licensee shall review the need to carry out hardness testing on welds and HAZ and indentify which measurement technique will be used.*

AF-AP1000-SI-36: *The Licensee shall ensure that where a thermal method is used to remove metal sufficient additional material is removed by a non-thermal method to ensure that the risk of cracking is acceptably low.*

AF-AP1000-SI-37: *The Licensee shall ensure that either the fabrication procedure for the welding of non-austenitic materials specifies "extra low hydrogen" welding consumables or, if this is not possible, a robust qualification process is specified.*

AF-AP1000-SI-38: *The Licensee shall ensure that the measurement of the weld pre-heat is specified in sufficient detail to ensure that it is adequate.*

4.13 Reactor Coolant Pump Design

4.13.1 Background

616 Towards the end of the Step 3 assessment process ND became aware that the design of the reactor coolant pump for the UK AP1000 was to be changed to a KSB design of pump using a forged martensitic stainless steel pump bowl.

617 The Step 4 structural integrity assessment plan took this design change into account, and the main areas of interest from a structural integrity perspective were the pump bowl integrity case and the flywheel disintegration case.

618 The decision to look in detail at the pump bowl was based on an assumption that the bowl would be classified as a HSS component. As it turned out the bowl has not been classified as an HSS component, and is a standard ASME III Class 1 vessel. Thus the additional integrity requirements associated with an HSS component do not need to be considered for the design, but the as the bowl is a relatively complex forging, and the design was still evolving during the Step 4 assessment period, I decided to continue with a more detailed consideration of the integrity case.

619 The flywheel is also not a HSS component, and the flywheel disintegration case is required to show that the consequences of catastrophic failure are acceptable.

4.13.2 Assessment

620 Various documents were supplied by Westinghouse following initial meetings in January 2010, including an analysis of the effects of flywheel disintegration (KSB Report H23 07 P033, Ref. 75), but important information on the design and material specification for the pump bowl forging were not provided.

621 It became apparent during the course of 2010 that the delay in supplying this information was due to the pump design having not been finalised, and that that there was uncertainty in the material choice for the pump bowl.

622 In October 2010 Westinghouse declared that the material for the pump bowl would be changed to a low alloy ferritic forging and that the final design review would not take place till early December 2010. Westinghouse was unable to provide information on the design and material specification until after that final design review for the pump, and as a consequence this information was not provided till February 2011 and even then only in a draft from. This draft information was subsequently included in the March 2011 version of the PCSR, Ref. 28, but that report arrived too late for meaningful assessment.

4.13.3 Pump Bowl Case

623 I have been unable to undertake an assessment of the pump bowl integrity case during Step 4 of GDA as Westinghouse was unable to provide the necessary information on the design and material specification.

624 I consider this to be a less than satisfactory position as the reactor coolant pump bowl is an important component in the primary circuit, and the necessary information should have been available for the Step 4 assessment process.

625 However, I recognise that the pump bowl is now to be manufactured using a clad low alloy steel and that this is a conventional technology. I also note that the pump bowl is not in the Highest Safety Significance category, and its design needs to be considered in that context of a standard ASME III Class 1 component.

626 I therefore consider that it should be possible to provide an adequate structural integrity safety case for the new design, and on that basis I judge that it is reasonable to support an IDAC. However I will need to assess the case post GDA Step 4 to confirm that this is the case before I would be confident to support a DAC. In particular the draft information provided by Westinghouse indicates that they intend to use a nickel based alloy weld to join the pump bowl to the steam generator. This is confirmed in Section 20C.1.5.1.1 of the March 2011 PCSR, Ref. 28 and the use of a dissimilar metal weld to join two low alloy components will need to be considered further during the post GDA Step 4 assessment.

627 I have raised GDA Issue **GI-AP1000-SI-03** to support the ongoing assessment of the pump bowl case.

4.13.4 Flywheel Disintegration Case

628 Work to assess the flywheel disintegration case was progressed with EASL reviewing the analysis of the effects of flywheel disintegration.

629 The review has been completed, Ref. 76, but I did not progress the technical queries arsing from that work due to uncertainties in the design of the pump, and whether the analysis would be affected by any material changes. In the event Westinghouse claimed that the material changes would not affect the flywheel disintegration case, but by the time this was recognised it was too late in the Step 4 assessment process to progress the queries from EASL to a conclusion and confirm that this is the case.

630 Based on the review work undertaken to date it would appear likely that a case for the disintegration of the flywheel can be made and I judge that it is reasonable to support an IDAC on that basis. I will need to progress the remaining queries to a conclusion post GDA Step 4 to confirm that this is the case before I would be confident to support a DAC. GDA Issue **GI-AP1000-SI-03** includes support the ongoing assessment of the flywheel disintegration case.

4.13.5 Conclusions, Issues and Findings Relating to Coolant Pump Design

631 I have been unable to undertake an assessment of the pump bowl integrity case during Step 4 of GDA as Westinghouse was unable to provide the necessary information on the design and material specification. In addition I have not progressed the technical queries arsing from EASL's review of the effects of flywheel disintegration within the Step 4 timeframe.

632 I believe that it should be possible for Westinghouse to provide an adequate structural integrity case for the new design of Reactor Coolant Pump and judge that it is reasonable to support an IDAC on that basis. However, I will need to assess the case post GDA Step 4, and have raised GDA Issue AP1000-SI-03 to support the ongoing assessment of the pump bowl case and the flywheel disintegration case.

> ***GI-AP1000-SI-03:*** *Reactor Coolant Pump Bowl. The key activities which will need to be completed by Westinghouse under this GDA Issue are:*
>
> - *Supply a technical report addressing the structural integrity considerations related to a clad ferritic pump bowl casing and support the ongoing assessment of the Pump Bowl Integrity Case*
> - *Support the ongoing assessment of the Flywheel Disintegration Case*

633 The complete GDA Issue and associated action(s) are formally defined in Annex 2.

4.14 Environmental Effects on Fatigue Design Curves

4.14.1 Background

634 In the last few years questions have arisen about whether code fatigue design analysis methods and code fatigue design curves (S-N curves) adequately account for the effects of Light Water Reactor (LWR) water environments. Although a good deal of experimental and analytical work has been done in this area, there are still uncertainties and something of a lack of consensus across the international nuclear industry.

635 As a starting point for GDA Step 3 I requested a list of stainless steel component locations whose fatigue usage factor was predicted to exceed 0.75. Some residual matters from this request which were taken forward into Step 4 are addressed under Section 4.5, but I have expanded consideration of the environmental effects on fatigue life within GDA Step 4 by commissioning a review of the current position (Ref. 70) from a retired ND inspector.

636 The effects are most relevant to stainless steel components as they are in direct contact with the LWR environment, but it is also relevant to un-clad PWR ferritic steel components, for example the steam generator secondary shells. There do not appear to be any issues to address for Nickel-Chrome-Iron alloys.

4.14.2 Current Position and the Way Forward

637 Efforts have been made worldwide from at least the late 1990s to determine the effects of Light Water Reactor (LWR) water environments on the fatigue life of metal components, and important work conducted at Argonne National Laboratory (ANL), sponsored by the United States Nuclear Regulatory Commission (US NRC) culminated in 2007 with issue

of NUREG/CR-6909, Effect of LWR Coolant Environments on the Fatigue Life of Reactor Materials (Ref. 71).

638 The fatigue evaluation procedure proposed in NUREG/CR-6909 has been adopted into US NRC Regulatory Guide 1.207 (Ref. 72) without change, and the US NRC considers this to be applicable to new nuclear reactor designs. The ASME III Code has been revised (2009 Addenda) to include a fatigue design curve for stainless steel in air which is the same as that recommended in NUREG/CR-6909, but importantly the environmental enhancement factors, Fen, are the subject of code cases which are still under discussion and have not yet been included.

639 I raised this matter through TQ-AP1000-515 (Ref. 9) and Westinghouse provided a detailed statement on their position on the effect of this work on the AP1000 fatigue analysis under cover of Ref. 73.

640 The response states:

‘..*Regulatory Guide 1.207 is not part of the AP1000 licensing basis. This Reg Guide was issued after the AP1000 was certified by the NRC....’*

Within the GDA process, Westinghouse contends that the consideration of Regulatory Guide 1.207 for the AP1000 new plant should parallel the license renewal approach.”

641 However, the GDA process is a technical assessment of matters that could affect safety, based on the best information available at the time of the assessment. Thus the position being adopted by Westinghouse is not acceptable from a UK regulatory perspective.

642 For a plant still to be built in the UK it is reasonable to expect that fatigue design analyses are reviewed to determine the potential effects of environmental effects on predicted fatigue life. The review procedure adopted by the Licensee should take account of recent results of research and development in this area. Although the review would be most relevant to stainless steel components, a similar approach should also be adopted for any ferritic components that require a code fatigue design analysis and are in contact with the wetted environment, for example the SG secondary shells.

643 One way forward could be to establish a fatigue design evaluation review procedure that:

1. Takes account of the generally accepted revision to the (in air) fatigue design curve. For example the sort of curve now incorporated in the 2010 Edition of the ASME III Code for austenitic stainless steel.
2. Includes a basis for determining environmental enhancement factors for fatigue.
3. Includes some form of ‘screening criterion’ based on environmental enhancement factors. If a particular location was below the screening criterion then no further action would be needed. However if a location was above the screening criterion then it would require specific, further consideration.

644 There might be other review procedures that could adequately address the issue, but whatever review procedure the Licensee adopts, it is important that the basis of the procedure be clear and justified. Supporting documentation justifying the basis of the procedure should be available for scrutiny if ONR chose to assess the review work.

645 I do not believe that it is necessary to have undertaken this fatigue design evaluation within the timeframe of GDA, but it will need to be completed during the Licensing phase and before commercial operation can start. As a result the Licensee will have to undertake a fatigue design evaluation for locations in stainless steel and ferritic

components that are in contact with the wetted environment to ensure that the effects of environment have been properly accounted for in the fatigue design analysis, and I have taken this forward as Assessment Finding **AF-AP1000-SI-39.**

646 I also believe that a thorough review of fatigue design analysis should be undertaken for the first Periodic Safety Review (10 years after start of commercial operation) as there may be an internationally agreed way of dealing with fatigue design analysis with a PWR water environment by that stage. I have decided that it would be unreasonable to set out expectations for the Periodic Safety review as an Assessment Finding, and the comment is for information only. This is approach recognises the US NRC position that designs submitted before the new Regulatory Guide will not have to meet the guidance until licence extension after 40 years of operation.

4.14.3 Conclusions and Findings on Environmental Effects on Fatigue Design Curves

647 Westinghouse has not yet adequately addressed the emerging findings on the affect of environment on fatigue design curves in their fatigue analysis for the AP1000.

648 I accept that although the US NRC has clearly stated its position on these effects, there remain uncertainties and something of a lack of consensus across the international nuclear industry as to how this matter should be addressed.

649 I do not believe that it would be practicable for Westinghouse to have meaningfully addressed this matter within GDA, but that it is reasonable to expect the matter is addressed prior to commercial operation. I therefore believe that the Licensee will have to undertake a fatigue design evaluation for locations in austenitic stainless steel and ferritic components that are in contact with the wetted environment to ensure that the effects of environment have been properly accounted for in the fatigue design analysis, and I have taken this forward as Assessment Finding **AF-AP1000-SI-39**.

> ***AF-AP1000-SI-39:*** *The Licensee shall undertake a fatigue design evaluation for locations in stainless steel and ferritic components that are in contact with the wetted environment to ensure that the effects of environment have been properly accounted for in the fatigue design analysis.*

650 This needs to be complete before the Generic Milestone of Hot Operations is reached This is because the projected fatigue life of the plant should be confirmed as adequate before it enters the operational phase.

651 I also believe that a thorough review of fatigue design analysis should be undertaken for the first Periodic Safety review (10 years after start of commercial operation) as there may be an internationally agreed way of dealing with fatigue design analysis with a PWR water environment by that stage, but this comment is for information only and is not carried forward as an Assessment Finding.

4.15 Design and Assessment Codes

652 In general the AP1000 has been designed in accordance with the requirements of 1998 edition of Section III of the ASME Boiler and Pressure Vessel code including addenda up to 2000 although specific additional requirements are added in the individual component equipment specifications. There are two more significant exceptions to the use of this version of the code given below.

- The treatment of dynamic loads on piping (including seismic loads) follows the requirements of the 1989 version (Section 5.2.1.1 of Ref. 15).
- The containment vessel is designed and constructed according to the 2001 edition of the ASME Code, Section III, Sub-section NE, Metal Containment, including the 2002 Addenda. Stability of the containment vessel and appurtenances is evaluated using ASME Code, Case N-284-1, Metal Containment Shell Buckling Design Methods, Class MC, Section III, Division 1, as published in the 2001 Code Cases, 2001 Edition, July 1, 2001 (Section 3.8.2.2 of Ref. 15).

653 Westinghouse wishes, as far as possible, to have a fixed design for all the AP1000s which are built. Since these will be built over a period of time this means that design of the later reactors will not necessarily be compliant with the latest code. This approach clearly has commercial benefits and it also needs to be recognised that making small, bespoke changes to one reactor only may not improve the overall safety of an otherwise coherent design. Nevertheless code changes may be made to take account of significant new knowledge which, when implemented, will significantly improve the safety of the plant and thus they should not be completely ignored.

654 Westinghouse has reviewed of the impact on the design of the AP1000 from changes to the ASME code up to the 2007 Edition with the 2008 and 2009 Addenda (Ref. 202) and have concluded that generally these would not result in a significant impact to the AP1000 design. However, they have identified one area which could have had an impact, this is a code change which extends the design fatigue curve. Westinghouse judges that in this case it would only result in a reduction of margins.

655 This review is based on a large sample of the code changes which appeared to the reviewers to be most likely to have resulted in design changes to the AP1000 if they were they applied. This is a useful approach and it gives confidence that significant changes to the design are unlikely to be required to achieve compliance with the latest version of the code. In the context of GDA this is acceptable however before a design for a specific plant could be licensed it will necessary to carry out a thorough and systematic review the impact of changes to the code in the intervening years This is addressed by Assessment Finding **AF-AP1000-SI-40.** I anticipate that this will be largely based on Ref. 202.

656 As in the case with Periodic Safety Reviews on operating plant, where changes to the code would have required a change in design the Licensee will need to decide whether to make the changes or if not provide arguments why it would not be ALARP to do so.

4.15.1 Findings Relating to Design and Assessment Codes

657 The following Assessment Finding has been raised:

> ***AF-AP1000-SI-40:*** *The Licensee shall carry out a review the changes to the design which would be required if the current version of ASME III were used and either make these changes or justify why these changes are not practical.*

658 Since this could affect the design of the major components this finding should be completed prior to procuring the long lead items and is linked to the generic milestone of long lead item and SSC procurement specifications. This is because it would be extremely difficult to make substantive changes once the components start have started to be manufactured, which could then lead to substantial delays and additional costs.

4.16 Other Matters

4.16.1 Pressuriser Heater Design

659 In March 2010 The Sizewell B power station had a forced outage following a small leakage of primary coolant from a pressuriser heater well. The rate of leakage was well within make up capabilities on the plant and the failure of the pressure boundary did not pose a threat to the overall integrity of the pressuriser shell, but I took a decision to review the design of the AP1000 heaters in the pressuriser against this specific operational experience.

660 The investigation of the failure is discussed in ND's Project Assessment Report on the Justification for Return to Service, Ref. 74. The initial failure was due to stress corrosion cracking of the stainless steel heater element sheath where it passes through the heater support plate which allowed water to enter the heater sheath. The water caused the magnesium hydroxide electrical insulation in the heater to swell leading to high stress levels and ultimately axial cracking of the heater well, resulting in primary coolant leakage. The stress corrosion cracking of the heater sheath is attributed to a susceptible material, high residual stresses from the manufacturing process, the environment inside the pressuriser and localised rubbing between the sheath and heater support plate removing the protective chromium oxide film in that area.

661 Westinghouse is aware of the potential for stress corrosion cracking of the heater sheath. The AP1000 heater sheaths will still be manufactured from stainless steel, but steps will be taken during the manufacturing process to avoid the conditions that could lead to stress corrosion cracking in the sheaths. The sheath will be heat-treated following the cold forming operations (to 1040^{o}C followed by rapid cooling in an inert gas) to remove the effects of the cold work and hence sensitisation of the material, and it will then be mechanically treated (shot peened or roller burnished) to introduce a compressive residual stress layer (TQ-AP1000-1122).

662 I am satisfied that steps taken to de-sensitise the stainless steel and the introduction of a compressive residual surface stress should be sufficient to avoid stress corrosion cracking in the environment of the pressuriser. In any case it must be recognised that failure of a heater sheath outside the pressure boundary would be well within the make up capacity of the design.

4.16.2 Conclusions and Findings on Pressuriser Heaters

663 I have reviewed the design of the pressuriser heater sheaths in context of operational experience from a pressure boundary failure at Sizewell B in March 2010. Westinghouse is aware of the potential for stress corrosion cracking in the heater sheaths and has improved the design of the heaters for the AP1000 by taking steps during the manufacturing process to avoid the conditions that could lead to stress corrosion cracking.

664 I am satisfied that these steps should be effective in avoiding stress corrosion cracking and that the AP1000 heater sheath design is adequate from a safety perspective.

4.17 Overseas Regulatory Interface

665 HSE's Strategy for working with overseas regulators is set out in (Ref. 5). In accordance with this strategy, HSE collaborates with overseas regulators, both bilaterally and multinationally.

4.17.1 Bilateral Collaboration

666 HSE’s Nuclear Directorate (ND) has formal information exchange arrangements to facilitate greater international co-operation with the nuclear safety regulators in a number of key countries with civil nuclear power programmes. These include:

- The US Nuclear Regulatory Commission (US NRC).
- The French L’Autorité de sûreté nucléaire (ASN).
- The Finnish STUK.

No meetings were held during Step 4 relating to structural integrity of the AP1000.

4.17.2 Multilateral collaboration

667 ND collaborates through the work of the International Atomic Energy Agency (IAEA) and the OECD Nuclear Energy Agency (OECD-NEA). ND also represents the UK in the Multinational Design Evaluation Programme (MDEP) – a multinational initiative taken by national safety authorities to develop innovative approaches to leverage the resources and knowledge of the national regulatory authorities tasked with the review of new reactor power plant designs. This helps to promote consistent nuclear safety assessment standards among different countries.

4.18 Interface with Other Regulators

668 Joint workshops have been held with ND and Environment Agency assessors involved in the GDA process.

4.19 Other Health and Safety Legislation

669 No other health and safety legislation has been considered explicitly during my assessment.

5 CONCLUSIONS

670 This report presents the findings of the Step 4 Structural Integrity assessment of the Westinghouse AP1000 reactor.

671 I am broadly satisfied with the process for identifying the components of HSS whose likelihood of failure has to be demonstrated to be so low that it may be discounted. I have reservations about the intermediate category termed HI, but have primarily considered these as HSS welds in terms of the GDA demonstration.

672 To support an Avoidance of Fracture demonstration for the components whose gross failure has been discounted (HSS and HI) Westinghouse has undertaken a series of fracture mechanics analyses for the twelve welds identified as being representative of the limiting welds. Westinghouse has also prepared limited scope technical justifications, termed Inspection Plans (IPs), for the manufacturing inspections of seven of these. I am satisfied that a representative set of limiting weld locations have been defined for the purposes of GDA.

673 I was unable to complete my assessment of the Avoidance of Fracture demonstration as a number of important documents were supplied too late for assessment during Step 4 of GDA. However I was able to undertake a detailed review of some of the documents, and a high level review of the other documents in order to form a view on the demonstration in terms of an IDAC. A GDA Issue has been raised to support the need for ongoing assessment work post Step 4, GDA Issue **GI-AP1000-SI-01**.

674 The fracture mechanics analyses calculated limiting defect sizes and undertook through life fatigue crack growth calculations using approaches that are consistent with those previously adopted by Licensees in the UK and I judge them to be acceptable. My high level review identified a number of important matters that will need to be considered further through the ongoing assessment in GDA Issue **GI-AP1000-SI-01**, but overall I have confidence in the approach being taken by Westinghouse and judge that the limiting defect sizes calculated can be used as the basis for the overall Avoidance of Fracture demonstration in terms of an IDAC.

675 In terms of the manufacturing inspections I reviewed two IPs which provided a reasonable argument that the required inspection capability could be achieved. The arguments that these can be extended to the smaller QEDS required in some cases following the completion of the fracture mechanics assessments are less well developed but nevertheless reasonable. My high level review and the endorsement by the IVC of the remaining IPs which I have not been able to assess, provides me with sufficient confidence to support an IDAC. Ongoing assessment of the remaining IPs will be carried forward through GDA Issue **GI-AP1000-SI-01**.

676 Westinghouse presented proposals for additional fracture toughness testing on parent material and representative weld mock-ups to underpin the fracture toughness values assumed in the fracture justifications, but these are not yet sufficiently developed to fully establish the principles of the additional testing. However the proposals have given me sufficient confidence to support an IDAC, and ongoing assessment and discussion will be carried forward through GDA Issue **GI-AP1000-SI-01**.

677 Thus at this stage of my assessment I am broadly satisfied with the methodology and the indicative results generated from the Avoidance of Fracture demonstration. However, as discussed, a more detailed assessment post GDA Step 4 will be required to confirm that an adequate justification has been made before I am confident to support a DAC and

GDA Issue GI-AP1000-SI-01 has been raised to support the need for ongoing assessment work post Step 4.

678 I have reservations about the approach adopted by Westinghouse relating to casting, forging, heat treatment, welding and cladding of the forgings for the main vessels. As a consequence the Licensee for an AP1000 plant will require the ability to specify and agree procedures at a detailed level with suppliers to ensure that, in addition to meeting the requirements of the ASME Code, additional controls on composition and manufacturing processes are incorporated where they are desirable and reasonably practicable for the RPV, Steam Generators and Pressuriser. Thus the approach places additional responsibilities on any Licensee as exemplified by the significant number of Assessment Findings relating to materials which are detailed in Annex 1.

679 Westinghouse has yet to demonstrate that the pressuriser surge line and other Class 1 pipework meet the ASME III fatigue limits, but they have confidence that this will be achieved as part of the design finalisation process and on this basis I am prepared to support an IDAC. However, I am not able to support a DAC until I have confidence that the design has been shown to be compliant with the ASME III fatigue limits for the 60 year life of the plant. I have therefore raised GDA Issue **GI-AP1000-SI-02** for Westinghouse to complete this demonstration.

680 I have been unable to undertake an assessment of the reactor coolant pump bowl structural integrity case as Westinghouse was unable to provide the necessary information on the new pump bowl design and material specification in time for assessment during Step 4 of GDA. I consider this to be a less than satisfactory position as the reactor coolant pump bowl is an important component in the primary circuit. However, I recognise that the pump bowl is now to be manufactured using a clad low alloy steel and that this is a well understood technology. I also note that the pump bowl is not in the HSS category. I therefore consider that it should be possible to provide an adequate structural integrity safety case for the new design, and on that basis I judge that it is reasonable to support an IDAC. However I will need to assess the case post GDA Step 4 to confirm that this is the case before I would be confident to support a DAC. I have raised GDA Issue **GI-AP1000-SI-03** to support the ongoing assessment of the pump bowl case.

681 The majority of the Containment Vessel (CV) is fabricated from plate which is just thin enough not to require post weld heat treatment under the recently revised relevant ASME Code, but further evidence is required to show that the CV has adequate tolerance to small defects in the absence of post weld heat treatment. The CV will also be subject to thermal loads when water flows over it from the PCS and the thermal stresses have not yet been shown as tolerable. Since I have not yet received adequate evidence to support these aspects of the safety case I have raised a GDA Issue **GI-AP1000-SI-04**.

682 I judge that Westinghouse has an adequate process to confirm that the HSS and HI welds can be adequately inspected during PSI and ISI.

683 A review of the stress calculations carried out to confirm the compliance of the AP1000 design with ASME III identified a number of apparent shortfalls. Westinghouse has provided a response to the specific concerns I identified and based on my high level review of this response I believe that they will be able to demonstrate compliance and on this basis I am able to support an IDAC. However, I am not able to support a DAC until I have completed my assessment of the evidence provided in the Westinghouse response. I have raised GDA Issue **GI-AP1000-SI-05** to support my ongoing assessment work.

684 ND is generally satisfied that the categorisation and classification scheme developed for the UK AP1000 is compliant with the SAPs, but late in the assessment process I identified two areas where the classification scheme as applied to pressure equipment and tanks needed further justification; the use of non-nuclear design and construction codes for Class 2 pressure equipment and tanks; and the use of ASME III Class 3 for the design and construction of the Accumulator Tanks in the Passive Core Cooling System. As these problems were only recognised late in the assessment process Westinghouse was unable to provide the necessary supporting evidence for this stage of the assessment process and I have therefore taken the matter forward as part of GDA Issue **GI-AP1000-SI-06** on Structural integrity Classification.

685 In addition towards the end of the assessment process I became aware that further evidence was required on the indirect consequences of failure. This is needed in order to support the categorisation of the reactor coolant pump bowl, and this has also been taken forward as part of GDA Issue **GI-AP1000-SI-06** on Structural Integrity Classification.

686 The GDA Issues discussed above are of particular significance and will require resolution before HSE would agree to the commencement of nuclear safety related construction of an AP1000 reactor in the UK. The complete GDA Issues and associated action(s) are formally defined in Annex 2.

687 I have also identified several areas of a Licensee or site specific nature that do not need to be addressed as part of the GDA process but which will need to be followed up by any Licensee and these are listed in Annex 1 as Assessment Findings.

688 Overall, based on the sample undertaken in accordance with ND procedures, 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 AP1000 reactor design. The AP1000 reactor is therefore suitable for construction in the UK, subject to satisfactory progression and resolution of GDA Issues and assessment of additional information that becomes available as the GDA Design Reference is supplemented with additional details on a site-by-site basis.

5.1 Key Findings from the Step 4 Assessment

689 The design of the AP1000 is broadly in line with my expectations in relation to current national and international standards, guidance and relevant good practice.

690 I have made a number of observations during my assessment which should be taken forward as normal regulatory business.

691 However in six areas of my assessment listed below I am not yet in a position to make a secure judgement about the acceptability of the design:

- Demonstration that the components of highest integrity have a risk of failure which is so low that it may be discounted.
- Demonstration that the pressuriser surge line and other Class 1 pipework meet the ASME III fatigue limits.
- The structural integrity safety case for the pump casing of the new design of reactor coolant pump.
- The containment vessel and its ability to tolerate the residual and thermal stresses to which it might be subjected.

- The evidence that the design of the main pressure boundary components comply with ASME III design rules.
- Application of categorisation and classification in an appropriate manner.

These are each the subject of a GDA Issue listed in Section 5.1.2 below.

5.1.1 Assessment Findings

692 I conclude that the Assessment Findings listed in Annex 1 should be programmed during the forward programme of this reactor as normal regulatory business. Some examples of my Assessment Findings are:

- The Licensee shall review the upper shelf fracture toughness to be assumed in the fracture assessments of the low alloy steel forgings and their weldments to ensure that they have confidence that values can be reliably achieved during the manufacture of these components.
- For the casting and forging manufacturing processes, the Licensee shall explain how the details of suppliers' procedures are assessed and provide the criteria used for deciding on whether they are acceptable. Examples of the aspects to be fully documented are the details of the casting process, control of segregated regions and material discarded, forging processes and forging ratios and heat treatment details
- The Licensee shall define and justify the chemical compositions of the main forgings regardless of whether the composition is based on ASME III compositions or on more restrictive limits. The justification shall take into account start-of-life materials properties and through-life changes.
- The Licensee shall prepare an ALARP justification to support the proposed initial core design which take appropriate account of the benefits of reducing the flux to the RPV. Safety cases will also be required to support subsequent core designs and these will also need to consider the benefit of reducing the RPV flux.
- The Licensee shall demonstrate that the damage correlation used to determine the shift in RTNDT is suitable for the RPV materials. This needs to reflect on the current understanding of damage correlations and should be kept under review over the life of the plant as new data becomes available from surveillance specimens and from worldwide data.
- The licensee shall demonstrate the protective coating applied to the containment vessel is capable of protecting it against extended exposure to the potentially corrosive chemical to which it may be exposed.
- The Licensee shall demonstrate that, for each stage of the procurement, manufacturing and construction process, the hierarchy of documents relevant to that stage is in place before the work commences

5.1.2 GDA Issues

693 I conclude that the following GDA Issues must be satisfactorily addressed before Consent will be granted for the commencement of nuclear island safety related construction.

GI-AP1000-SI-01	Avoidance of Fracture – Margins Based on Size of Crack-Like Defects. Support assessment of fracture mechanics assessments and inspection plans.
GI-AP1000-SI-02	Provide sufficient evidence to show that ASME III Class 1 pipework has an adequate fatigue life for the 60 year design life of the reactor.
GI-AP1000-SI-03	Supply a technical report addressing the structural integrity considerations related to a clad ferritic pump bowl casing. And support the ongoing assessment of the Flywheel Disintegration Case.
GI-AP1000 SI-04	Provide sufficient evidence to show that the containment vessel has adequate tolerance to the thermal shock and to small defects in the absence of post weld heat treatment.
GI-AP1000-SI-05	Demonstrate compliance of AP1000 Main Structural Components with ASME III Design Rules
GI-AP1000-SI-06	Provide evidence to show that categorisation and classification has been applied in an appropriate manner to components with an important structural integrity claim.

694 The complete GDA Issues and associated actions are formally defined in Annex 2.

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Table 1

Relevant Safety Assessment Principles for Structural Integrity Considered During Step 4

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: the metal component or structure should be as defect-free as possible; 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.

Table 1

Relevant Safety Assessment Principles for Structural Integrity Considered During Step 4

SAP No.	SAP Title	Description
EMC.8	Integrity of metal components and structures: design. Requirements for examination	Geometry and access arrangements should have regard to the requirements for examination.
EMC.9	Integrity of metal components and structures: design. Product form	The choice of product form of metal components or their constituent parts should have regard to enabling examination and to minimising the number and length of welds in the component.
EMC.10	Integrity of metal components and structures: design. Weld positions	The positioning of welds should have regard to high-stress locations and adverse environments.
EMC.11	Integrity of metal components and structures: design. Failure modes	Failure modes should be gradual and predictable.
EMC.12	Integrity of metal components and structures: design. Brittle behaviour	Designs in which components of a metal pressure boundary could exhibit brittle behaviour should be avoided.
EMC.13	Integrity of metal components and structures: manufacture and installation. Materials	Materials employed in manufacture and installation should be shown to be suitable for the purpose of enabling an adequate design to be manufactured, operated, examined and maintained throughout the life of the facility.
EMC.17	Integrity of metal components and structures: manufacture and installation. Examination during manufacture	Provision should be made for examination during manufacture and installation to demonstrate the required standard of workmanship has been achieved.
EMC.21	Integrity of metal components and structures: operation. Safe operating envelope	Throughout their operating life, safety-related components and structures should be operated and controlled within defined limits consistent with the safe operating envelope defined in the safety case.

Table 1

Relevant Safety Assessment Principles for Structural Integrity Considered During Step 4

SAP No.	SAP Title	Description
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.

Table 1

Relevant Safety Assessment Principles for Structural Integrity Considered During Step 4

SAP No.	SAP Title	Description
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.

Table 1

Relevant Safety Assessment Principles for Structural Integrity Considered During Step 4

SAP No.	SAP Title	Description
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
EPS.1	Removable closures	The failure of a removable closure to a pressurised component or system that could lead to a major release of radioactivity should be prevented.
EPS.2	Flow limitation	Flow limiting devices should be provided to piping systems that are connected to or form branches from a main pressure circuit, to minimise the consequences of postulated breaches.
EPS.3	Pressure relief	Adequate pressure relief systems should be provided for pressurised systems and provision should be made for periodic testing.
EPS.4	Overpressure protection	Overpressure protection should be consistent with any pressure-temperature limits of operation.
EPS.5	Discharge routes	Pressure discharge routes should be provided with suitable means to ensure that any release of radioactivity from the facility to the environment is minimised.

Table 2

Areas for Further Assessment During Step 4 (derived from Step 4 Assessment Plan)

Inspection Plan Identifier	Description of Step 4 Assessment	Regulatory Observation	Report Section	TSC Report (if applicable)
AR09058-1	**Categorisation and Classification of Structures, Systems and Components** Agree categorisation of components and welds	RO-AP1000-18	4.1.2, 4.8.3	N/A
AR09058-2	**Avoidance of Fracture** • Agree methodology for determining limiting defect sizes, qualifying manufacturing inspections and deriving material properties. • Assess sample fracture mechanics analyses and manufacturing inspection technical justifications to gain confidence that a full set of analyses and technical justification can be developed after GDA.	RO-AP1000-19	4.2	Refs 58, 59, 60, 61, 181
AR09058-3	**Manufacturing Method for Reactor Coolant Pump Casings**	RO-AP1000-20	Planned but not required as the pump design was changed – see AR09058-13 below.	
AR09058-4	**Materials Specifications and Selection of Material Grade – Reactor Pressure Vessel, Pressuriser, Steam Generator Shells.** Review the chemical specification for ferritic forgings.	RO-AP1000-21	4.3	Refs 102,104
AR09058-5	**Effects of Irradiation on Reactor Pressure Vessel Body Cylindrical Shell Forging Material and Associated Circumferential Welds** • Review practicality of using low leakage core. • Review methodology for predicting irradiation embrittlement of RPV	RO-AP1000-22	4.4	Ref. 154
AR09058-6	**Fatigue Usage Factor Analysis Results for the Pressuriser Surge Line** Confirm that the Pressuriser Surge Line fatigue usage factor complies with ASME.	RO-AP1000-26	4.6	N/A

Table 2

Areas for Further Assessment During Step 4 (derived from Step 4 Assessment Plan)

Inspection Plan Identifier	Description of Step 4 Assessment	Regulatory Observation	Report Section	TSC Report (if applicable)
AR09058-7	**Reactor Pressure Vessel Pressure – Temperature Limit Diagrams and Low Temperature Overpressure Protection** Assess the acceptability of the new methodology for calculating the P-T limit curve and judge whether it is ALARP	RO-AP1000-29	4.5	Ref. 158
AR09058-8	**Containment Pressure Vessel Shell** Review design of the Containment Vessel giving particular consideration the acceptability of no post weld heat treatment of most welds and negligible corrosion allowance over most of the vessel.	RO-AP1000-30	4.7	Ref. 167
AR09058-9	**Documentary Envelope for Specific Components** Review the Design Specifications and a sample of the analyses of loading conditions contained within them.	RO-AP1000-65	4.8	Ref. 179
AR09058-10	**Review of Access for In-Service Inspection** Confirm that the design has given appropriate consideration to the needs of an adequate in-service inspection.	RO-AP1000-66	4.10	
AR09058-11	**Operation of Plant within Safe Limits** Review the demonstration that the constructed plant will be capable of being operated within safe limits, including the role of technical specification, maintenance schedule, procedures (especially normal operation) and operating limits giving particular emphasis on operating limits for components relevant to structural integrity.	RO-AP1000-94	4.11	
AR09058-12	**Review of the Welding procedures for the Highest Integrity Components** Review of welding procedures for a sample of the highest integrity components (e.g. RPV, Pressuriser, Steam generators, main pipework). Determine if standard Code requirements are used or if additional requirements are specified or should be		4.12	Ref. 161

Table 2

Areas for Further Assessment During Step 4 (derived from Step 4 Assessment Plan)

Inspection Plan Identifier	Description of Step 4 Assessment	Regulatory Observation	Report Section	TSC Report (if applicable)
AR09058-13	**Reactor Coolant Pump Design.** Assess structural integrity aspects of the new design of Reactor Coolant Pump. In particular the design of the welds and consequence of flywheel failure.		4.13	Ref. 76, 111
Further Assessment Identified During Step 4				
	Environmental Effects on Fatigue Design Curves Review the need to take account of the requirements of NUREG 1.207 (effect of environment on fatigue crack growth).		4.14	
	Design and Assessment Codes Consider the acceptability of using a design code which is more than ten years out of date.		4.15	
	Pressuriser Heater Design Consider the relevance of the recent Pressuriser heater leakage seen at Sizewell B to the AP1000		4.16	

Table 3

Reactor Pressure Vessel Materials (from RO-AP1000-21)

	ASME standard composition SA508 Grade 3 Class 1 2007 Edition (formerly SA508 Class 3) [1]	**UK Usage of SA508 Class 3 (now called SA508 Grade 3 Class 1) RPV Product Analysis**	**UK AP-1000 SSER Chapter 5 Section 5.3 Table 5.3-1 RPV Beltline Forging [4]**	**UK AP-1000 WEC supplementary material specification APP-VL51-Z0-004 Rev 3 2009 RPV Beltline Forging [4]**
Carbon	0.25% max	0.2% max		
Manganese	1.2 to 1.5%	1.2 to 1.5%		
Molybdenum	0.45 to 0.6%	0.45 to 0.6%		
Nickel	0.4 to 1.0%	0.4 to 0.85%	[redacted]	[redacted]
Sulphur	0.025% max	0.008% max	[redacted]	[redacted]
Phosphorus	0.025% max	0.008% max	[redacted]	[redacted]
Silicon [3]	0.4% max	0.3% max		
Chromium	0.25% max	0.15% max		[redacted]
Copper	0.2% max	0.08% max	[redacted]	[redacted]
Vanadium	0.05% max	0.01% max	[redacted]	[redacted]
Antimony	-	0.008% max		
Arsenic	-	0.015% max		
Cobalt	-	0.02% max		
Tin	-	0.01% max		
Aluminium	0.025% max [2]	0.045% max		

Table 3

Reactor Pressure Vessel Materials (from RO-AP1000-21)

	ASME standard composition SA508 Grade 3 Class 1 2007 Edition (formerly SA508 Class 3) [1]	UK Usage of SA508 Class 3 (now called SA508 Grade 3 Class 1) RPV Product Analysis	UK AP-1000 SSER Chapter 5 Section 5.3 Table 5.3-1 RPV Beltline Forging [4]	UK AP-1000 WEC supplementary material specification APP-VL51-Z0-004 Rev 3 2009 RPV Beltline Forging [4]
Hydrogen	-	1ppm (product) max		
Boron	0.003% max[2]			
Columbium *	0.01% max[2]			
Calcium	0.015% max[2]			
Titanium	0.015% max[2]			

*Columbium = Niobium

ASME specifies steel to be made using an electric furnace and vacuum-degassed.

Notes to Table 3

1 ASME A508 Specification – Supplementary Requirement S9 specifies:
S9.1.1 Phosphorus 0.015% max product, Copper 0.1% max product or
S9.1.2 Phosphorus 0.015% max product, Copper 0.15% max product
S9.2 Sulphur 0.015% max product.

2 Element limit added since ASME Code edition for UK usage.

3 ASME A508 Specification – Supplementary Requirement S11 sets limit on Silicon of 0.1% max. Supplementary Specification S16 sets range of Silicon content as 0.05 to 0.15%.

4 Element composition assumed the same as ASME standard composition for S508 Grade 3 Class 1 unless specific different limit stated.

Table 4

Weld Metal Chemical Composition

	UK Usage Weld Metal (as deposited)	UK AP-1000 SSER Chapter 5 Section 5.3 Table 5.3-1 Weld Metal (as deposited)[1]	UK AP-1000 WEC supplementary material specification APP-VW40-Z0-050 Rev 0 2009 AP1000 Filler Material to ASME III Sect II Part C SFA-5.5
Carbon	0.15% max		
Manganese	0.8 to 1.8%		
Molybdenum	0.35 to 0.65%		
Nickel	0.85% max	[redacted]	[redacted]
Sulphur	0.01% max	[redacted]	[redacted]
Phosphorus	0.01% max	[redacted]	[redacted]
Silicon	0.15 to 0.6%		
Chromium	0.15% max		
Copper	0.07% max	[redacted]	[redacted]
Vanadium	0.01% max	[redacted]	[redacted]
Antimony	0.008% max		
Arsenic	0.015% max		
Cobalt	0.02% max		
Tin	0.01% max		

Table 4

Weld Metal Chemical Composition

	UK Usage Weld Metal (as deposited)	UK AP-1000 SSER Chapter 5 Section 5.3 Table 5.3-1 Weld Metal (as deposited)[1]	UK AP-1000 WEC supplementary material specification APP-VW40-Z0-050 Rev 0 2009 AP1000 Filler Material to ASME III Sect II Part C SFA-5.5
Aluminium			
Hydrogen			

Notes to Table 4

1 Element composition assumed the same as ASME standard composition unless specific different limit stated.

2 Values for RPV core region.

3 All pressure boundary welds.

Table 5

Steam Generator and Pressuriser Materials

	ASME standard composition SA508 Grade 3 (Class 1 & 2) 2007 Edition (formerly A 508 Class 3) (¾Ni-½Mo-Cr-V)[3]	ASME standard composition SA 508 Class 1A 2007 Edition (Carbon steel)[3]	UK Usage of SA508 Class 3 (now called SA508 Grade 3 Class 1) and welds for Steam Generators – base materials and welds	UK Usage of SA508 Class 3 (now called SA508 Grade 3 Class 1) and welds for Pressuriser – base material and welds
Carbon	0.25% max	0.3% max	0.2% max	0.22% max
Manganese	1.2 – 1.5%	0.7 – 1.35%	1.2 – 1.5%	1.2 – 1.5%
Molybdenum	0.45 – 0.6%	0.1% max	0.45 – 0.6%	0.45 – 0.6%
Nickel	0.4 – 1.0%	0.4% max	0.4 – 0.85%	0.4 – 0.85%
Sulphur	0.025% max	0.025% max	0.01% max	0.01% max
Phosphorus	0.025% max	0.025% max	0.012% max	0.012% max
Silicon	0.4% max [2]	0.4% max [2]	0.3% max	0.3% max
Chromium	0.25% max	0.025% max	0.15% max[1]	0.15% max
Copper	0.2% max	0.2% max		
Vanadium	0.05% max	0.05% max	0.01% max	0.01% max
Antimony			0.01% max	0.01% max
Arsenic			0.02% max	0.02% max
Cobalt				

Table 5

Steam Generator and Pressuriser Materials

	ASME standard composition SA508 Grade 3 (Class 1 & 2) 2007 Edition (formerly A 508 Class 3) (¾Ni-½Mo-Cr-V)[3]	ASME standard composition SA 508 Class 1A 2007 Edition (Carbon steel)[3]	UK Usage of SA508 Class 3 (now called SA508 Grade 3 Class 1) and welds for Steam Generators – base materials and welds	UK Usage of SA508 Class 3 (now called SA508 Grade 3 Class 1) and welds for Pressuriser – base material and welds
Tin			0.015% max	0.015% max
Aluminium	0.025% max	0.025% max		
Hydrogen				

Notes to Table 5

UK precedent is to use SA508 Grade 3 Class 1 (formerly SA508 Class 3) for all major primary circuit pressure vessel forgings (including secondary shells of Steam Generators).

Notes to Table 3

1 Primary side shell only, no limit set on Chromium for secondary side shell.

2 Purchaser may specify minimum Silicon content of 0.15%.

3 Identification used in ASME II Part D stress tables.

Annex 1

Assessment Findings to Be Addressed During the Forward Programme as Normal Regulatory Business

Structural Integrity – AP1000

Finding No.	Assessment Finding	MILESTONE (by which this item should be addressed)
AF-AP1000-SI-01	The Licensee shall review the structural integrity classification scheme to determine whether the definition and use of an intermediate HI component/weld category is relevant and useful in terms of the overall safety case for the UK AP1000.	Install RPV
AF-AP1000-SI-02	The Licensee shall review the structural integrity classification scheme to remove the element of expert judgement in defining the HSS boundary by ensuring that the formalised assessments of the indirect consequences of failure of the Standard Class 1 and HI components/welds are fully reflected in the structural integrity classification scheme.	Install RPV
AF-AP1000-SI-03	The Licensee shall ensure that the case for categorising the main steam line and main feed line in the main steam isolation valve compartment as Standard Class 1 components includes explicit evidence that coincident failure of a main steam line and main feed line will not be limiting from a thermal hydraulic safety analysis perspective.	Install RPV
AF-AP1000-SI-04	The Licensee shall undertake fracture assessments on a wider range of weld locations on the HSS Components in order to demonstrate that the limiting locations have been assessed. The Licensee shall also undertake fracture assessments on the vulnerable areas of the parent forgings in order to demonstrate that the limiting locations have been assessed.	Install RPV
AF-AP1000-SI-05	The Licensee shall undertake fracture assessments to show that a postulated defect with a 10:1 aspect ratio defect would not lead to an unacceptably large reduction in the Defect Size Margin (DSM) in the overall demonstration of fracture i.e.the Licensee shall demonstrate that a 10:1 aspect ratio would not lead to a 'cliff edge' effect on the DSM.	Install RPV

Annex 1

Assessment Findings to Be Addressed During the Forward Programme as Normal Regulatory Business

Structural Integrity – AP1000

Finding No.	Assessment Finding	MILESTONE (by which this item should be addressed)
AF-AP1000-SI-06	The Licensee shall use a robust methodology for indentifying the limiting time steps for use in the more extensive fracture assessments that will be undertaken post GDA.	Install RPV
AF-AP1000-SI-07	The Licensee shall prepare Technical Justifications for the qualified manufacturing Inspections of all the HSS and HI welds	Install RPV
AF-AP1000-SI-08	The Licensee shall justify the selection of defects used to qualify each manufacturing inspection.	Install RPV
AF-AP1000-SI-09	The Licensee shall set up a robust and independent Inspection Qualification Body	Install RPV
AF-AP1000-SI-10	The Licensee shall develop inspection procedures for the HI, and HSS forgings and justify their coverage and capability These procedures should specify the actions to be taken if defects are detected	Install RPV
AF-AP1000-SI-11	The Licensee shall produce a comprehensive material data set for the HSS, HI, Standard 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.	Hot Operations
AF-AP1000-SI-12	For the casting and forging manufacturing processes, the Licensee shall explain how the details of suppliers' procedures are assessed and provide the criteria used for deciding on whether they are acceptable. Examples of the aspects to be fully documented are the details of the casting process, control of segregated regions and material discarded, forging processes and forging ratios and heat treatment details.	Long Lead Item and SSC Procurement Specifications

Annex 1

Assessment Findings to Be Addressed During the Forward Programme as Normal Regulatory Business

Structural Integrity – AP1000

Finding No.	Assessment Finding	MILESTONE (by which this item should be addressed)
AF-AP1000-SI-13	The Licensee shall define and justify the chemical compositions of the main forgings regardless of whether the composition is based on ASME III compositions or on more restrictive limits. The justification shall take into account start-of-life materials properties and through-life changes.	Long Lead Item and SSC Procurement Specifications
AF-AP1000-SI-14	The Licensee shall specify reasonably practicable controls on the composition and variability (e.g. in segregated areas) of carbon and other elements which affect the likelihood of defects from welding.	Long Lead Item and SSC Procurement Specifications
AF-AP1000-SI-15	The Licensee shall ensure that the maximum value of sulphur content in the belt-line forgings is restricted, either by setting an upper limit not exceeding 0.005% or by setting a target value with a rigorous process for reviewing the acceptability of the sulphur content should the actual value be above 0.005%. For the other main forgings (outside the belt-line), the licensee shall also consider whether it is reasonably practicable to reduce the sulphur levels specified.	Long Lead Item and SSC Procurement Specifications
AF-AP1000-SI-16	The Licensee shall specify and justify limits on residual elements arsenic, antimony, cobalt and tin which take account of the precedent set by UK usage as listed in Tables 3 and 5.	Long Lead Item and SSC Procurement Specifications
AF-AP1000-SI-17	The Licensee shall supply evidence of the capability of the forgemaster's processes and procedures to achieve a satisfactory level of uniformity in mechanical properties.	Long Lead Item and SSC Procurement Specifications
AF-AP1000-SI-18	The Licensee shall review whether, in the light of other relevant good practice, it is reasonably practicable to improve the mechanical property values in the specification for the forgings.	Long Lead Item and SSC Procurement Specifications

Annex 1

Assessment Findings to Be Addressed During the Forward Programme as Normal Regulatory Business

Structural Integrity – AP1000

Finding No.	Assessment Finding	MILESTONE (by which this item should be addressed)
AF-AP1000-SI-19	The Licensee shall ensure that welding and cladding procedures are demonstrated to be consistent with modern good practice, that they include appropriate limits for the preheat and post-heat temperatures and that evidence is provided to ensure that grain size is adequately controlled.	Install RPV
AF-AP1000-SI-20	The Licensee shall ensure that sample ultrasonic inspections for underclad cracking are performed during manufacture of the RPV, SGs and PZR.	Install RPV
AF-AP1000-SI-21	The Licensee shall prepare an ALARP justification to support the proposed initial core design which take appropriate account to the benefits of reducing the flux to the RPV. Safety cases will also be required to support subsequent core designs and these will also need to consider the benefit of reducing the RPV flux.	Fuel Load
AF-AP1000-SI-22	The Licensee shall demonstrate that the damage correlation used to determine the shift in RT_{NDT} is suitable for the RPV materials. This needs to reflect on the current understanding of damage correlations and should be kept under review over the life of the plant as new data becomes available from surveillance specimens and from worldwide data.	Fuel Load
AF-AP1000-SI-23	A Licensee shall have access to an adequate database so that thermal ageing effects can be reliably predicted and, if necessary, a thermal ageing surveillance programme should be established for materials operating at temperatures experienced by the RPV outlet nozzles and the pressuriser.	Install RPV
AF-AP1000-SI-24	The Licensee shall propose P-T and LTOPS limits for a UK AP1000 and justify these.	Hot Operations
AF-AP1000-SI-25	The Licensee shall confirm that the containment vessel wall temperature does not rise above the design temperature in the event of a reactor coolant loop or main steam line failure or if it does justify that this is acceptable.	Long Lead Item and SSC Procurement Specifications

Annex 1

Assessment Findings to Be Addressed During the Forward Programme as Normal Regulatory Business

Structural Integrity – AP1000

Finding No.	Assessment Finding	MILESTONE (by which this item should be addressed)
AF-AP1000-SI-26	The Licensee shall include planned periodic visual inspection of the CV, its protective coatings and the moisture barrier in its arrangements for periodic inspections. Particular attention should be given to the concrete embedment transition.	Containment Pressure Test
AF-AP1000-SI-27	The licensee shall demonstrate the protective coating applied to the containment vessel is capable of protecting it against extended exposure to the potentially corrosive chemicals to which it may be exposed.	Containment Pressure Test
AF-AP1000-SI-28	The Licensee shall include the guidance on coating application, repair of coating defects and the qualification of staff for application and inspection of coatings in its procedures and arrangements.	Containment Pressure Test
AF-AP1000-SI-29	The Licensee shall ensure that the safety case for the structural integrity components on the individual site reflects the actual build and operation on that site.	Hot Operations
AF-AP1000-SI-30	The Licensee shall demonstrate that, for each stage of the procurement, manufacturing and construction process, the hierarchy of documents for the structural integrity components relevant to that stage is in place before the work commences. This Assessment Finding shall be completed before the generic milestone of RPV Installation, although in practice it will need to be completed at various times to suit the construction programme.	RPV Installation
AF-AP1000-SI-31	The Licensee shall ensure that all the Design/Fabrication Actions in the ISI Inspectability Reports are either completed, or the issue addressed in an alternative way.	Install RPV
AF-AP1000-SI-32	The Licensee shall ensure that all the Pre PSI/ISI Actions in the ISI Inspectability Reports are either completed, or the issue addressed in an alternative way	Cold Ops

Annex 1

Assessment Findings to Be Addressed During the Forward Programme as Normal Regulatory Business

Structural Integrity – AP1000

Finding No.	Assessment Finding	MILESTONE (by which this item should be addressed)
AF-AP1000-SI-33	The Licensee shall prepare justified PSI/ISI inspection procedures for each of the HSS and HI welds.	Cold Ops
AF-AP1000-SI-34	The Licensee shall set up suitable arrangements to ensure that leaks are reliably and promptly detected and subsequently managed.	Hot Operations
AF-AP1000-SI-35	The Licensee shall review the need to carry out hardness testing on welds and HAZ and indentify which measurement technique will be used.	Install RPV
AF-AP1000-SI-36	The Licensee shall ensure that where a thermal method is used to remove metal that sufficient additional material is removed by non-thermal method to ensure that the risk of cracking is acceptably low.	Install RPV
AF-AP1000-SI-37	The Licensee shall ensure that either the fabrication procedure for the welding of non-austenitic materials specifies “extra low hydrogen” welding consumables or, if this is not possible, a robust qualification process is specified.	Install RPV
AF-AP1000-SI-38	The Licensee shall ensure that the measurement of the weld pre-heat is specified in sufficient detail to ensure that it is adequate.	Install RPV
AF-AP1000-SI-39	The Licensee shall undertake a fatigue design evaluation for locations in stainless steel and ferritic components that are in contact with the wetted environment to ensure that the effects of environment have been properly accounted for in the fatigue design analysis.	Hot Operations
AF-AP1000-SI-40	The Licensee shall carry out a review the changes to the design which would be required if the current version of ASME III were used and either make these changes or justify why these changes are not practical.	Long Lead Item and SSC Procurement Specifications

Annex 1

Note: It is the responsibility of the Licensees / Operators to have adequate arrangements to address the Assessment Findings. Future Licensees / Operators can adopt alternative means to those indicated in the findings which give an equivalent level of safety.

For Assessment Findings relevant to the operational phase of the reactor, the Licensees / Operators must adequately address the findings <u>during</u> the operational phase. For other Assessment Findings, it is the regulators' expectation that the findings are adequately addressed no later than the milestones indicated above.

Annex 2

GDA Issues – Structural Integrity – AP1000

WESTINGHOUSE AP1000® GENERIC DESIGN ASSESSMENT

GDA ISSUE

STRUCTURAL INTEGRITY – AVOIDANCE OF FRACTURE

GI-AP1000-SI-01 REVISION 0

<table>
<tr><td colspan="2">Technical Area</td><td colspan="2">STRUCTURAL INTEGRITY</td></tr>
<tr><td colspan="2">Related Technical Areas</td><td colspan="2">None</td></tr>
<tr><td>GDA Issue Reference</td><td>GI-AP1000-SI-01</td><td>GDA Issue Action Reference</td><td>GI-AP1000-SI-01.A1</td></tr>
<tr><td>GDA Issue</td><td colspan="3">Avoidance of Fracture - Margins Based on Size of Crack-Like Defects

Demonstration of defect tolerance and the absence of planar defects in the components for which the likelihood of gross failure is claimed to be so low it can be discounted.

This requires integration of qualified non-destructive examinations during manufacture and analyses for limiting sizes of crack-like defects using conservative material fracture toughness properties.</td></tr>
<tr><td>GDA Issue Action</td><td colspan="3">Support assessment of the fracture analysis approach by providing adequate responses to any questions arising from assessment by ONR of documents submitted during GDA Step 4 but not reviewed in detail at that time.

A number of important fracture assessment reports arrived much later in the Step 4 assessment timeframe than had been originally planned. ONR undertook a high level review of the reports to gain confidence in the approach but was unable to undertake a full assessment within the timescales allowed for GDA Step 4. This GDA Issue Action has been created to support the full assessment of these reports.

Activities by Westinghouse should comprise:

• Provide adequate responses to questions arising from ONR assessment of documents submitted during GDA Step 4 or in response to this Action.

With agreement from the Regulator this action may be completed by alternative means.</td></tr>
</table>

Annex 2

WESTINGHOUSE AP1000® GENERIC DESIGN ASSESSMENT

GDA ISSUE

STRUCTURAL INTEGRITY – AVOIDANCE OF FRACTURE

GI-AP1000-SI-01 REVISION 0

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<tr><td>GDA Issue Reference</td><td>GI-AP1000-SI-01</td><td>GDA Issue Action Reference</td><td>GI-AP1000-SI-01.A2</td></tr>
<tr><td>GDA Issue Action</td><td colspan="3">Demonstrate that there are qualifiable inspection techniques capable of detecting the limiting defects with adequate margin in a representative range components for which the likelihood of gross failure is claimed to be so low it can be discounted.

A number of the important reports on inspection qualification arrived much later in the Step 4 assessment timeframe than had been originally planned. As a result ONR has been unable to undertake a full assessment of all inspection qualification reports within the timescales allowed for GDA Step 4, but has undertaken a high level review of the reports where a full assessment was not possible in order to gain confidence in the apporach, This GDA Issue Action has been created to support the full assessment of the reports not yet fully assessed..

Activities by Westinghouse should comprise:
Provide adequate responses to questions arising from ONR assessment of documents submitted during GDA Step 4 or as a result of this action.
With agreement from the Regulator this action may be completed by alternative means.</td></tr>
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Annex 2

WESTINGHOUSE AP1000® GENERIC DESIGN ASSESSMENT

GDA ISSUE

STRUCTURAL INTEGRITY – AVOIDANCE OF FRACTURE

GI-AP1000-SI-01 REVISION 0

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<tr><td colspan="2">Related Technical Areas</td><td colspan="2">None</td></tr>
<tr><td>GDA Issue Reference</td><td>GI-AP1000-SI-01</td><td>GDA Issue Action Reference</td><td>GI-AP1000-SI-01.A3</td></tr>
<tr><td>GDA Issue Action</td><td colspan="3">Provide formalised proposals for additional materials testing to underpin the fracture toughness values used in the fracture analyses.

Westinghouse have acknowledged that testing over and above the standard testing required by ASME will be required to underpin the fracture toughness values used in the fracture analyses, however formalised proposals for additional materials testing have not yet been provided.

Activities by Westinghouse should comprise:

• Provide formalised proposals for additional materials testing to underpin the fracture toughness values used in the fracture analyses.
• Provide adequate responses to questions arising from ONR assessment of documents submitted during GDA Step 4 or as a result of this Action.

With agreement from the Regulator this action may be completed by alternative means.</td></tr>
</table>

Annex 2

WESTINGHOUSE AP1000® GENERIC DESIGN ASSESSMENT
GDA ISSUE
STRUCTURAL INTEGRITY – FATIGUE ANALYSIS
GI-AP1000-SI-02 REVISION 0

<table>
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<tr><td colspan="2">Related Technical Areas</td><td colspan="2">None</td></tr>
<tr><td>GDA Issue Reference</td><td>GI-AP1000-SI-02</td><td>GDA Issue Action Reference</td><td>GI-AP1000-SI-02.A1</td></tr>
<tr><td>GDA Issue</td><td colspan="3">Fatigue Analysis of ASME III Class 1 Piping.</td></tr>
<tr><td>GDA Issue Action</td><td colspan="3">Provide sufficient evidence to show that ASME III Class 1 pipework has an adequate fatigue life for the 60 year design life of the reactor.
Activities by Westinghouse should comprise:
• Provide sufficient evidence to show that ASME III Class 1 pipework has an adequate fatigue life for the 60 year design life of the reactor.
• Provide adequate responses to any questions arising from assessment by ONR of documents submitted.
With agreement from the Regulator this action may be completed by alternative means.</td></tr>
</table>

Annex 2

WESTINGHOUSE AP1000® GENERIC DESIGN ASSESSMENT

GDA ISSUE

STRUCTURAL INTEGRITY – REACTOR COOLANT PUMP

GI-AP1000-SI-03 REVISION 0

<table>
<tr><td colspan="2">Technical Area</td><td colspan="2">STRUCTURAL INTEGRITY</td></tr>
<tr><td colspan="2">Related Technical Areas</td><td colspan="2">None</td></tr>
<tr><td>GDA Issue Reference</td><td>GI-AP1000-SI-03</td><td>GDA Issue Action Reference</td><td>GI-AP1000-SI-03.A1</td></tr>
<tr><td>GDA Issue</td><td colspan="3">Reactor Coolant Pump – Pump Bowl Integrity Case and Flywheel Disintegration Case.</td></tr>
<tr><td>GDA Issue Action</td><td colspan="3">Supply a technical report addressing the structural integrity considerations related to a clad ferritic pump bowl casing.and support the assessment of the pump bowl integrity case.
Activities by Westinghouse should comprise:
Provide a report on the structural integrity considerations related to clad ferritic casing material which includes consideration of:
Pump casing design,
Material specification (Chemical composition and mechanical properties),
Forging manufacturing process,
Pump Casing fabrication,
Weld geometry, process and procedures for safe end weld to pipework and attachment weld for steam generator,
Clad process including process, specification and inspections.
ASME analyses of the design.
Provide adequate responses to any questions arising from the assessment by ONR of the pump bowl integrity case.
With agreement from the Regulator this action may be completed by alternative means.</td></tr>
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Annex 2

WESTINGHOUSE AP1000® GENERIC DESIGN ASSESSMENT
GDA ISSUE
STRUCTURAL INTEGRITY – REACTOR COOLANT PUMP
GI-AP1000-SI-03 REVISION 0

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<tr><td>GDA Issue Reference</td><td>GI-AP1000-SI-03</td><td>GDA Issue Action Reference</td><td>GI-AP1000-SI-03.A2</td></tr>
<tr><td>GDA Issue Action</td><td colspan="3">Support the ongoing assessment of the Flywheel Disintegration Case
Activities by Westinghouse should comprise:
• Provide adequate responses to any questions arising from assessment by ONR of documents submitted.
With agreement from the Regulator this action may be completed by alternative means.</td></tr>
</table>

Annex 2

WESTINGHOUSE AP1000® GENERIC DESIGN ASSESSMENT
GDA ISSUE
STRUCTURAL INTEGRITY – CONTAINMENT VESSEL
GI-AP1000-SI-04 REVISION 0

<table>
<tr><td>Technical Area</td><td colspan="3">STRUCTURAL INTEGRITY</td></tr>
<tr><td>Related Technical Areas</td><td colspan="3">None</td></tr>
<tr><td>GDA Issue Reference</td><td>GI-AP1000-SI-04</td><td>GDA Issue Action Reference</td><td>GI-AP1000-SI-04.A1</td></tr>
<tr><td>GDA Issue</td><td colspan="3">Fracture Analysis of Containment Vessel.</td></tr>
<tr><td>GDA Issue Action</td><td colspan="3">Provide sufficient evidence to show that the containment vessel has adequate tolerance to the thermal shock due to the flow of PCS water onto the top head.
Activities required to be carried out by Westinghouse are:
• Provide a report with the structure proposed during GDA to show that the containment vessel has adequate tolerance to the thermal shock due to the flow of PCS water onto the top head.
• Provide adequate responses to any questions arising from assessment by ONR of documents submitted.
With agreement from the Regulator this action may be completed by alternative means.</td></tr>
</table>

Annex 2

WESTINGHOUSE AP1000® GENERIC DESIGN ASSESSMENT

GDA ISSUE

STRUCTURAL INTEGRITY – CONTAINMENT VESSEL

GI-AP1000-SI-04 REVISION 0

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<tr><td>GDA Issue Reference</td><td>GI-AP1000-SI-04</td><td>GDA Issue Action Reference</td><td>GI-AP1000-SI-04.A2</td></tr>
<tr><td>GDA Issue Action</td><td colspan="3">Provide sufficient evidence to show that the containment vessel has adequate tolerance to small defects given the high residual stress associated with welds which have not undergone post weld heat treatment.

It is anticipated that simple fracture mechanics calculations will be required to show adequate defect tolerance. It may be necessary to critically review the input parameters (design temperatures, pressures, residual stresses, likely manufacturing flaws etc) to ensure that they are self consistent and realistic.

Activities required to be carried out by Westinghouse are:
• Provide sufficient evidence to show that the containment vessel has adequate tolerance to small defects in the absence of post weld heat treatment.
• Provide adequate responses to any questions arising from assessment by ONR of documents submitted.

With agreement from the Regulator this action may be completed by alternative means.</td></tr>
</table>

Annex 2

WESTINGHOUSE AP1000® GENERIC DESIGN ASSESSMENT

GDA ISSUE

COMPLIANCE OF AP1000 MAIN STRUCTURAL COMPONENTS WITH ASME III DESIGN RULES

GI-AP1000-SI-05 REVISION 0

<table>
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<tr><td colspan="2">Related Technical Areas</td><td colspan="2">MSQA</td></tr>
<tr><td>GDA Issue Reference</td><td>GI-AP1000-SI-05</td><td>GDA Issue Action Reference</td><td>GI-AP1000-SI-05.A1</td></tr>
<tr><td>GDA Issue</td><td colspan="3">Provide evidence to show that the design of the Main Structural Vessels is compliant with the ASME III code.</td></tr>
<tr><td>GDA Issue Action</td><td colspan="3">Support the assessment of Westinghouse’s response to ONR’s findings on the AP1000 Stress Analysis.

The review of the reactor pressure vessel report identified a number of areas where it was unclear why specific assumptions and approximations had been made. In their response to this review Westinghouse justified these. The review of the pressuriser report identified errors in the calculations for the safety relief nozzle however a revision of this report was in preparation during ONR’s review; this corrected all the main errors.

The response to the comments on the reactor pressure vessel report and the revision of the pressuriser report were both supplied too late for ONR to undertake a full assessment of these documents within GDA Step 4.

Activities by Westinghouse should comprise:

• Provide adequate responses to questions arising from ONR assessment of documents submitted during GDA Step 4 or in response to this Action.

With agreement from the Regulator this action may be completed by alternative means.</td></tr>
</table>

Annex 2

WESTINGHOUSE AP1000® GENERIC DESIGN ASSESSMENT

GDA ISSUE

COMPLIANCE OF AP1000 MAIN STRUCTURAL COMPONENTS WITH ASME III DESIGN RULES

GI-AP1000-SI-05 REVISION 0

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<tr><td>Related Technical Areas</td><td colspan="3">None</td></tr>
<tr><td>GDA Issue Reference</td><td>GI-AP1000-SI-05</td><td>GDA Issue Action Reference</td><td>GI-AP1000-SI-05.A2</td></tr>
<tr><td>GDA Issue Action</td><td colspan="3">Provide evidence that there will not be similar errors elsewhere in the design support documentation.

ONR have identified errors on a sample review of the design calculations. The calculations were verified and issued, and referred to within the GDA submissions, but not approved as the formal issue (Rev 0) of the report. In this circumstance the formal issue of the report corrected the errors in the calculational route of ‘design by rule’, and in this case, even if error had not been detected, the design was still secure because the design route ‘design by analysis” had also been followed. Nevertheless, since a sample review identified significant errors in a verified document, evidence is required to demonstrate that the process in raising design reports to Rev 0 is sufficiently robust to ensure that errors missed by the author and verifier of the earlier revisions will be reliably detected.

Activities by Westinghouse should comprise:
• Provide evidence that the process for raising verrified douments to Revision 0 is sufficiently robust..
• Provide adequate responses to any questions arising from assessment by ONR of the response

With agreement from the Regulator this action may be completed by alternative means.</td></tr>
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Annex 2

WESTINGHOUSE AP1000® GENERIC DESIGN ASSESSMENT
GDA ISSUE
STRUCTURAL INTEGRITY CATEGORISATION AND CLASSIFICATION
GI-AP1000-SI-06 REVISION 0

<table>
<tr><td colspan="2">Technical Area</td><td colspan="2">STRUCTURAL INTEGRITY</td></tr>
<tr><td colspan="2">Related Technical Areas</td><td colspan="2">None</td></tr>
<tr><td>GDA Issue Reference</td><td>GI-AP1000-SI-06</td><td>GDA Issue Action Reference</td><td>GI-AP1000-SI-06.A1</td></tr>
<tr><td>GDA Issue</td><td colspan="3">Provide evidence to show that categorisation and classification has been applied in an appropriate manner to components with an important structural integrity claim.</td></tr>
<tr><td>GDA Issue Action</td><td colspan="3">Provide evidence to show that the principal design and construction codes adopted for Class 2 Pressure Equipment and Storage Tanks are consistent with ONR’s expectations as detailed within the SAPs, particularly ECS.3 and supporting paragraphs 157-161. In particular, where non-nuclear Pressure Equipment and Storage Tank design and construction codes are used in the design of Class 2 components Westinghouse will need to fully justify each case to show the arguments and evidence which support the use on non-nuclear codes. The arguments and evidence should take account of:
the safety significance of the component;
the demands that are placed on the system in terms of loadings, fatigue, temperature etc, and;
the consequences of failure of pressure boundary in terms of both the loss of system function and on the Internal Hazards safety case.
With agreement from the Regulator this action may be completed by alternative means.</td></tr>
</table>

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WESTINGHOUSE AP1000® GENERIC DESIGN ASSESSMENT

GDA ISSUE

STRUCTURAL INTEGRITY CATEGORISATION AND CLASSIFICATION

GI-AP1000-SI-06 REVISION 0

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<tr><td>GDA Issue Reference</td><td>GI-AP1000-SI-06</td><td>GDA Issue Action Reference</td><td>GI-AP1000-SI-06.A2</td></tr>
<tr><td>GDA Issue Action</td><td colspan="3">Provide evidence to show that components in AP1000 Equipment Class C have been assigned a class that is consistent with their intended duty and implied reliability. In particular Westinghouse need to provide arguments and evidence to show why its is appropriate to design and construct the Accumulator Tanks in the Passive Core Cooling System to ASME III Class 3 when previous designs of reactor would have designed and constructed the Accumulators to ASME III Class 2 in line with the guidance provided in ANS-51.1-1983. The arguments and evidence should address:
• the intended duty and implied reliability of the vessel, and;
• provide evidence to justify why the AP1000 design has apparently downgraded the classification of the core cooling system from the criteria set in ANS-51.1-1983.
With agreement from the Regulator this action may be completed by alternative means</td></tr>
</table>

Annex 2

WESTINGHOUSE AP1000® GENERIC DESIGN ASSESSMENT
GDA ISSUE
STRUCTURAL INTEGRITY CATEGORISATION AND CLASSIFICATION
GI-AP1000-SI-06 REVISION 0

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<tr><td>GDA Issue Reference</td><td>GI-AP1000-SI-06</td><td>GDA Issue Action Reference</td><td>GI-AP1000-SI-06.A3</td></tr>
<tr><td>GDA Issue Action</td><td colspan="3">Provide arguments and evidence to show that catastrophic failure of a reactor coolant pump bowl would not challenge the effectiveness vertical support for the Steam Generator.

The reactor coolant pump bowl has been assigned a Standard Class 1 structural integrity classification. It will be designed and constructed to ASME III, but this is not sufficient in its own right to discount the possibility of gross failure. As a result it is necessary to address the consequences of failure of the pump bowl.

Due to the proximity of the reactor coolant pump bowl to the Steam Generator vertical support it is not obvious that failure of the support can be discounted as not credible without sufficient evidence.

Thus Westinghouse will need to provide the evidence that the effectiveness of the Steam Generator vertical support will not be challenged by the failure of the pump bowl in order to support the assignment of a Standard Class 1 structural integrity classification for the pump bowl.

With agreement from the Regulator this action may be completed by alternative means</td></tr>
</table>

Further explanatory / background information on the GDA Issues for this topic area can be found at:	
GI-AP1000-SI-01 Revision 0	Ref. 206.
GI-AP1000-SI-02 Revision 0	Ref. 207.
GI-AP1000-SI-03 Revision 0	Ref. 208.
GI-AP1000-SI-04 Revision 0	Ref. 209.
GI-AP1000-SI-05 Revision 0	Ref. 210.
GI-AP1000-SI-06 Revision 0	Ref. 211.