Health and Safety Executive

NUCLEAR DIRECTORATE

GENERIC DESIGN ASSESSMENT – NEW CIVIL REACTOR BUILD

STEP 3 STRUCTURAL INTEGRITY ASSESSMENT OF THE WESTINGHOUSE AP1000

DIVISION 6 ASSESSMENT REPORT NO. AR 09/013-P

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EXECUTIVE SUMMARY

This report records my assessment of the nuclear safety-related structural integrity aspects of the Westinghouse AP1000 design, for Step 3 of the Generic Design Assessment (GDA). The Nuclear Directorate (ND) usage of the term ‘structural integrity’ covers metal pressure boundary components, their supports and some of the associated internal support structures (e.g. for a PWR, the core barrel).

In this GDA Step 3 assessment of the structural integrity aspects of the AP1000 design proposed for the UK, I have not identified any matters that would lead to a recommendation to raise a Regulatory Issue.

During GDA Step 3 I have raised a number of matters with Westinghouse; I have done this mostly through thirteen Regulatory Observations (RO). Some matters raised are relatively more significant than others. I consider useful progress has been made across all of these Regulatory Observations. Several aspects of these Regulatory Observations remain to be resolved. I consider there is a reasonable prospect of achieving such resolution by carrying these remaining open aspects forward into GDA Step 4.

For structural integrity aspects of the AP1000 and from an ND perspective I believe there has been a significant improvement in HSE understanding of the design.

For components where ‘the likelihood of gross failure is claimed to be so low it can be discounted’, Westinghouse has indicated a willingness to implement a method of achieving and demonstrating integrity consistent with UK practice. I regard this as substantial progress within GDA Step 3 and a basis for going forward. Execution of the programme of work will extend into GDA Step 4. Full implementation might extend beyond GDA Step 4. If so, some interim work might be done within GDA Step 4 to give confidence for final implementation. I believe ND will want to take an interest in the detail of this work and how the programme of works progresses.

The question of which components have the claim that the likelihood of gross failure is so low it can be discounted still remains to be completely resolved. Westinghouse has proposed a programme of work to address this matter. I regard this as substantial progress within GDA Step 3 and a basis for going forward. Execution of the programme of work will extend well into GDA Step 4. I believe ND will want to take an interest in the detail of this work and how the programme of works progresses.

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. For the purposes of this assessment, I have assumed the Reactor Pressure Vessel (RPV) will have set-in (also referred to as set-through) nozzles, rather than an integral nozzle shell course design. If an integral nozzle shell course was proposed for the RPV, this would require specific assessment.

For neutron irradiation embrittlement of regions of the RPV, the AP1000 design takes account of what is now known regarding chemical composition of the base materials and welds. However, the end of life maximum neutron dose to the forgings is quite high. I recommend ND seeks an As Low As Reasonably Practicable (ALARP) review of practical options for meaningful reduction of neutron dose to the RPV. As a minimum this should consider the locations of peak neutron dose, which occur in the forging. On the face of it, there is the potential for adopting a ‘low leakage core’ fuel management arrangement in service.

The basis of Reactor Coolant Pump Casing construction based on casting technology has been justified. However, there are still aspects to resolve in how to deal with large repairs to the castings

made by welding. The areas still open relate to how to obtain confidence that crack-like defects of a size of concern for integrity can be detected.

Useful progress has been made in understanding the approach to be used for an AP1000 in setting Pressure-Temperature limit curves for the RPV. However, there are aspects still to be resolved, for instance consideration of what is ALARP.

The design of the steel containment shell complies with the American Society of Mechanical Engineers (ASME) code. There are some general aspects that need further consideration: plate thickness available for corrosion allowance for most of the shell, the toughness properties of the plates and welds to meet the requirements for no post weld heat treatment and tolerance on plate thickness relevant to both corrosion allowance and no post weld heat treatment. There are a number of matters to take forward for further assessment.

The AP1000 Design Control Document (DCD) states that the Steam Generator tubing will be made using mill annealed Alloy 690 in the Thermally Treated (TT) condition. Based on my knowledge of UK experience of Thermally Treated Alloy 690 Steam Generator tubing and a general perception of international experience of this material, I had no particular concerns about its use. But, given the past interest in the UK of this aspect of Pressurised Water Reactor (PWR) structural integrity, I judged it prudent to give the matter some consideration. I decided to do this through a support contract to review PWR Steam Generator tube materials and manufacturing routes.

Overall, I conclude from the review and my general knowledge of this area that Alloy 690 in the Thermally Treated condition is a sound choice of material for Steam Generator Tubing. When supported by detailed manufacturing practice and in-service water chemistry control, Alloy 690TT tubing exhibits good resistance to stress corrosion cracking. However, material choice, manufacturing practice and in-service water chemistry are not a panacea. The general design and construction aspects of the Steam Generator as they affect the tubing also have a role. Important factors are the minimisation of ‘crevice’ conditions, support for the tubing to avoid vibration induced wear and support materials that themselves do not corrode. Most of these general design and construction factors have been understood for many years, and the AP1000 Steam Generator design takes these into account.

A number of matters are identified above for carrying forward in to GDA Step 4 and some will require significant effort and programmes of work on the part of Westinghouse (e.g. the work for Regulatory Observation RO-AP1000-19). In addition GDA Step 4 for structural integrity needs to move to the next level of detail and consider the content of documents such as:

- Design Specifications.
- Analyses for loading conditions (mainly thermal-hydraulics analyses - this will require involvement of other ND assessment functions).
- Design Reports.
- Equipment Specifications.

for a range of components.

From an ND perspective, I consider there has been a reduction in regulatory risk.

LIST OF ABBREVIATIONS

ALARP	As Low as Reasonably Practicable
ASME	American Society of Mechanical Engineers
BMS	(Nuclear Directorate) Business Management System
BSL	Basic Safety Level (in SAPs)
BSO	Basic Safety Objective (in SAPs)
BWR	Boiling Water Reactor
CFR	Code of Federal Regulations (USA)
CVS	Chemical and Volume System
DCD	Design Control Document
dpa	displacements per atom
EA	The Environment Agency
EPRI	Electric Power Research institute
FSER	Final Safety Evaluation Report (by USNRC)
GDA	Generic Design Assessment
GDC	General Design Criteria (or in singular Criterion). In Appendix A of 10 CFR 50
HSE	The Health and Safety Executive
IAEA	The International Atomic Energy Agency
IASCC	Irradiation Assisted Stress Corrosion Cracking
IGA	Intergranular Attack
IGSCC	Intergranular Stress Corrosion Cracking
ND	Nuclear Directorate (of HSE)
PCSR	Pre-construction Safety Report
PID	Project initiation Document (HSE / ND)
P-T	Pressure-Temperature
PWSCC	Primary Water Stress Corrosion Cracking
QA	Quality Assurance
RCC-M	Design and Construction Rules for Mechanical Components of PWR Nuclear Islands (AFCEN, France)
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

LIST OF ABBREVIATIONS

SAP	Safety Assessment Principle (plural SAPs)
SSC	System, Structure and Component
TAG	(Nuclear Directorate) Technical Assessment Guide
TQ	Technical Query
TT	Thermally Treated (referring to Steam Generator tubing)
US NRC (or NRC)	United States Nuclear Regulatory Commission
WEC	Westinghouse Electric Company LLC

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1 INTRODUCTION

1 This report records my assessment of the nuclear safety-related structural integrity aspects of the Westinghouse UK AP1000, for Step 3 of the Nuclear Directorate's (ND's) Generic Design Assessment (GDA). The ND usage of the term 'structural integrity' covers metal pressure boundary components, their supports and some of the associated internal support structures (e.g. for a Pressurised Water Reactor [PWR], the core barrel).

2 The specific aims of GDA Step 3 are to (Ref. 10. Page 14):

- improve HSE 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 a significant reduction in regulatory uncertainty.

3 It is expected that assessment will continue in GDA Step 4.

4 For GDA Step 3, my assessment has concentrated on the components likely to have a major influence on nuclear safety, particularly high consequence, low likelihood events. In practice this means my assessment has concentrated on the primary pressure boundary and to some extent the secondary pressure boundary of the AP1000 and mostly those components within the containment building. Examples of components included within this scope are:

Reactor Pressure Vessel

Pressuriser

Steam Generators

Reactor Coolant Pumps (pressure boundary)

Primary Coolant Loop Piping

5 For GDA Step 3, I have not considered components outside the containment building, or low pressure / low temperature systems. However, for the AP1000 design I have given some consideration to the steel pressure shell of the containment building, mainly the material of construction.

6 My assessment began in June 2008, based mainly on the AP1000 Design Certification Document Revision 16 (AP1000 DCD Revision 16, as submitted to the USNRC) (Ref. 1), and to some extent on the Final Safety Evaluation Report (FSER) (Ref. 2) produced by United States Nuclear Regulatory Commission (USNRC) in response to AP1000 Design Control Document (DCD) Revision 15 (sent by Westinghouse to ND in 2008). A UK Pre-Construction Safety Report (PCSR) for the AP1000 was sent by Westinghouse to ND, 15 December 2008 (Ref. 3). The PCSR depends extensively on the AP1000 DCD Revision 16.

7 The formal methods of interacting with the Requesting Parties (RPs) for technical aspects of their submissions are (in order of increasing significance):

Technical Queries

Regulatory Observations

Regulatory Issues

8 For this assessment, most of my formal, technical interactions with Westinghouse have been based on a number of Regulatory Observations. I sent Westinghouse a set of draft Regulatory Observations 31 October 2008, by email. These draft Regulatory Observations were the basis for a meeting with Westinghouse, 4-6 February 2009 (Ref.

4). Final versions of the Regulatory Observations were sent by ND to Westinghouse on 17 March 2009 (ND letter WEC70066R); see Table 1 for the list of Regulatory Observations.

9 Draft proposals for actions to answer the Regulatory Observations were sent from ND to Westinghouse on 23 March 2009, by email. Westinghouse agreed the way forward to answer the Regulatory Observations in an email sent 5 May 2009. The final wording of the actions to answer the Regulatory Observations was sent by ND to Westinghouse by letter on 7 May 2009 (WEC70077R).

10 A further meeting was held between ND and Westinghouse on 2/3 June 2009 (Ref. 5). The final meeting in GDA Step 3 with Westinghouse was held 16 September 2009 (Ref. 47).

11 For all the face-to-face meetings mentioned above, and progress meetings conducted by telephone, my view is all have been conducted in a professional, positive manner in an atmosphere of mutual respect.

12 Westinghouse issued a 'European Design Control Document' (European DCD) to ND 4 March 2009 (Ref. 6). This 'European DCD' is in effect US AP1000 DCD Revision 17. Westinghouse did not supply a USNRC Final Safety Evaluation Report (FSER) corresponding to US AP1000 DCD Revision 17. Westinghouse issued an updated version of the AP1000 UK Pre-construction Safety Report (PCSR) in April 2009 (Ref. 7); this revised PCSR depends extensively on the European DCD. So far as structural integrity is concerned, the updated AP1000 PCSR and the European DCD contain no changes compared with the 2008 equivalents.

2 WESTINGHOUSE CASE

2.1 UK AP1000 PCSR Overview of structure and relevant content

13 The UK AP1000 PCSR (Refs 3 and 7) in Section 1.2 states it is a generic PCSR and the head safety case document within the GDA, and as such provides the claims and arguments that the design is safe, referencing the appropriate supporting evidence.

14 The UK PCSR also states that the European AP1000 Design Control Document (DCD) provides the bulk of the evidence cited by the generic PCSR in the arguments that it develops to justify the claims it makes. Note that UK AP1000 PCSR Revision 0 (Ref. 3) in referencing the 'European DCD' refers to Ref. 1 here, which is Revision 16 of the US DCD, while the UK AP1000 PCSR Revision 1 (Ref. 7) refers to Ref. 6 here which is essentially Revision 17 of the US DCD. Note the small change in the DCD document numbers, UKP-GW-GL-700 (Ref. 1) and EPS-WL-GL-700.

15 I made a comparison between the UK AP1000 PCSR Revisions 0 and 1 (Ref. 11). My summary comments are:

> "*Based on the side-lining of Revision 1, at first sight there are a large number of changes. However the vast majority of the changes are text formatting matters, rather than changes in words. The summary below gives my quick analysis of the differences between Revisions 0 and 1*."
>
> My overall conclusion is:
>
> "*Putting the document into A4 paper size and improving the reference list are the only worthwhile, substantive changes. Fiddling with formatting may or may not be desirable, but it is not worth side-lining, that just makes it very difficult to detect real changes. I have found no substantive technical change (actually no technical change of any sort)*."

As there are no technical changes between the two revisions, in the following I make no distinction between the two.

16 The sections of the UK AP1000 PCSR relevant to structural integrity are listed in Table 2, and those sections of the DCD relevant to structural integrity are listed in Table 3.

17 For the significant pressure boundary components of interest, the most important part of the AP1000 DCD is Chapter 5.

18 ND seeks a 'safety case' based on a framework of 'Claims - Arguments - Evidence' (see Safety Assessment Principles (SAPs) SC.3, Paragraph 90 and SC.4 Paragraph 91(b), (Ref. 8), and G/AST/001, Paragraph 2.4 of Appendix - Mechanics of Assessment, where 'claims' are referred to as 'safety requirements' (Ref. 15)). One way of implementing such a framework is to:

> define, for each system / plant / function / operation the functional and integrity requirements relevant to safety ('safety design bases');
>
> describe the detailed way in which conformity with the above 'safety design bases' is achieved ('safety design approaches').

The description of how conformity with the safety design bases is achieved would be the majority of the text of such a safety case - i.e. information will be the majority of the text.

19 The PCSR and the DCD together are not a complete 'safety case' in a UK context. For a given component, such as for example the Reactor Pressure Vessel (RPV), there will be a number of significant documents that contribute to the safety case. Such documents will include the 'design report' and the 'equipment specification'. And to realise a component requires a system of quality assurance, with documentary evidence of satisfactory compliance with requirements. The contents of these additional documents are not appropriate for the PCSR, however they are part of the safety case. There should be a list of such supporting documents that, taken together constitute the 'safety case'. With this overall structure, the PCSR (and its successors, see below) and the DCD might provide the 'Claims and Arguments' end of the framework while the supporting documents provide the 'Evidence' end of the framework.

20 At the stage of the PCSR, the complete suite of documents constituting the safety case is not needed, and some will not be available. However, for a specific licensed site, as detailed design is completed and the station is constructed, supporting documents will be produced, and by the time the station enters service, the Station Safety Report will have evolved from the PCSR (SAP SC.3 Paragraph 90 (Ref. 8)). The Station Safety Report as a document might look overall similar in scope and extent to the PCSR, however there will need to be a system of referencing other documentation that taken together forms the 'safety case'. The operating plant 'safety case' needs to be a living document that takes account of modifications to plant or analyses which support the claims for safety. Many of the documents that are part of the safety case for an operating plant, will be 'lifetime records' retained at the plant.

2.2 UK AP1000 PCSR Outline of safety case claims for structural integrity

21 The UK AP1000 PCSR in Section 1.2 states it "*provides the claims and arguments that the design is safe, referencing the appropriate supporting evidence*". Section 1.4 of the PCSR is titled "Nuclear Safety Case Claims of the AP1000". There are 9 claims listed in section 1.4 of the UK AP1000 PCSR, see Table 4 here.

22 These are the only explicit claims made in the UK AP1000 PCSR, they are repeated in Section 10 of the PCSR.

23 The 9 claims in the UK AP1000 PCSR are valid, but they are very top-tier claims. The ND expectation is that a PCSR will use a structured 'claims-arguments-evidence' sequence

throughout, down to much lower tier levels. Specifically, the UK AP1000 PCSR contains no claims for structural integrity. The UK AP1000 PCSR does use the word 'claim' in quite a number of locations, but on review I do not think there is a structured 'claims-arguments-evidence' approach overall. The PCSR in places does repeat text from the DCD that sets out 'System Safety Functions', and it might be thought these could be classed as 'claims'. However, so far as the PCSR is concerned and structural integrity, the 'System Safety Functions' do not help. For example the UK AP1000 PCSR for the Reactor Coolant System and Associated Systems (Section 6.3) states the System Safety Functions as:

- The Pressuriser safety valves provide overpressure protection in accordance with Section III of the ASME Boiler and Pressure Vessel Code.
- The RCS provides automatic depressurisation during DBEs.
- The RCS provides emergency letdown during DBEs.
- The Automatic Depressurisation System (ADS) provides a controlled method to depressurise the reactor coolant system in case the core make up tanks have significantly drained. It must not spuriously initiate.

24 These are valid System Safety Functions, but do not constitute a complete set of safety-related 'claims' for the reactor coolant system; at least they do not touch on passive component integrity.

25 The DCD would not be expected to exhibit a 'claims-arguments-evidence' format because it was constructed starting from a different basis. Fundamentally, the DCD seeks to show compliance with US NRC Criteria, Regulations and Guidance.

26 But some of the approach in the DCD can be re-interpreted, to provide a set of claims as a starting point for the part of a safety case for the structural integrity aspects. In particular several of the US NRC General Design Criteria (GDC) can be re-cast in a 'claim' format. From this perspective and for structural integrity, the GDC are a mixture of those that contain high level claims and those that contain ways of achieving the claims, or arguments. Taking the US NRC GDC highlighted in Table 3, Criteria 4, 14 and 16 can be recast as claims while Criteria 15, 31, 32 and 51 can be recast as the basis of ways of achieving the claims, or arguments. This is summarised in Tables 5 (for claims) and Table 6 (for arguments).

27 On this basis, Criterion 4 becomes a claim on how internal hazards are dealt with, while Criterion 14 becomes a claim on how the reactor coolant pressure boundary is dealt with.

28 In the DCD Chapter 5.3 (Reactor Vessel) has 'Safety Design Bases' listed in 5.3.1.1 but the list is a combination of performance and safety design bases. It is not clear whether each item in the list is both a performance and a safety design basis. However as written, none appear to be claims. DCD Chapter 5.4 (Component and Subsystem Design), deals with the rest of the reactor coolant system and connected systems. Most of the sections dealing with each component simply refer to 'Design Bases'. The only instance where 'Safety Design Basis' is separated from 'Non-safety Design Basis' is in 5.4.7.1 for the Normal Residual Heat Removal System.

29 In DCD Chapter 5.4, buried in the 'Design Bases' section for the Reactor Coolant Pump Assembly is a very clear safety claim relevant to internal hazards which states:

> "*The reactor coolant pump pressure boundary shields the balance of the reactor coolant pressure boundary from theoretical worst-case flywheel failures*."

And goes on to outline the argument supporting this claim:

> "*The reactor coolant pump pressure boundary is analyzed to demonstrate that a fractured flywheel cannot breach the reactor coolant system boundary (impacted*

pressure boundary components are stator closure, stator main flange, and lower stator flange) and impair the operation of safety-related systems or components. This meets the requirements of General Design Criteria 4."

30 In Chapter 10 of the DCD there is a specific claim in the negative concerning the turbine condenser:

10.4.1.1.1 Safety Design Basis

"*The main condenser serves no safety-related function and therefore has no nuclear safety design basis*."

But the tubes and tubeplates of the condenser keep separate the water of the ultimate heat sink from the condensate / feedwater. There may be systems in the condensate and feed system to monitor and treat the condensate / feedwater, but it must be preferable to keep the water of the ultimate heat sink from entering in the first place. As an example of the sort of safety design basis I have seen elsewhere for the turbine condenser, the following wording is offered:

"*The main condenser does not perform any safety function but does have a potential to impact on the safety function of other systems. For this reason the following safety design basis has been adopted for the main condenser:*

1. the main condenser is designed to facilitate reactor heat removal via the turbine bypass subsystem post reactor trip subject to the condenser maintaining adequate vacuum;

2. the main condenser is designed to minimise ingress of ultimate heat sink water and its contents into the condensate."

31 With one exception, the UK AP1000 PCSR is not useful for the assessment of structural integrity in terms of 'claims-arguments-evidence', and the DCD was not constructed on that basis. However, the DCD does contain a significant amount of information relevant to the functional and integrity requirements of metal pressure boundary and other components of the UK AP1000 design.

32 The exception for the UK AP1000 PCSR referred to above is in Section 5.5.5 where it is stated:

"*This section of the PCSR considers DBA safety arguments that have been developed using an approach outside the standard DBA process. There is one such specific area, concerning the safety argument demonstrating the structural integrity of primary systems pipe work to be acceptable. The approach specified in the DCD diverges from UK relevant good practice, as represented in Reference 42. [Ref. 12 here]*

To further align this approach with UK relevant good practice, consideration will be given to supplementing the established NRC basis for the integrity of the AP1000 piping systems using fracture analyses of selected welds based upon credible fabrication manufacturing defects and the capability of the inspections applied during manufacture and pre-service.

For the Leak before Break and the Break Exclusion Zone Categories, consideration will be given to presenting the overall arguments in terms of quality of design and manufacture, functional testing of the pipe work, quantification of the defect tolerance of the pipe work and indicators of forewarning of failure. The use of such an argument structure will expose the existing piping quality and integrity arguments in a way that is readily appreciable by a UK audience, and will allow judgements of the overall integrity of the systems to be made."

33 This is a potentially useful recognition of a point that has been raised during this assessment (see Section 5). However, it is a commitment to consider something, it is not an element of a safety case as it stands.

2.3 UK AP1000 PCSR Outline of arguments and evidence to support the claims for structural integrity

34 The nature of the type of arguments deployed in the UK AP1000 PCSR is summarised by the text in Section 6.3 "Reactor Coolant System and Associated Systems, Section 6.3.3 "Conformance with Design Requirements" and its sub-sections 6.3.3.1 and 6.3.3.2 where it is stated:

> "*6.3.3.1 Introduction*
>
> *General Design Criteria (GDC) 1, "Quality Standards and Records" requires that nuclear power plant System, Structure and Components (SSCs) important to safety be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety function to be performed. This requirement is applicable to both pressure-retaining and non-pressure-retaining SSCs that are part of the reactor coolant pressure boundary, as well as other systems important to safety. Where generally recognized codes and standards are used, they must be identified and evaluated to determine their adequacy and applicability.*
>
> *Pursuant to US Section 55a, "Codes and Standards", of Reference 54, components important to safety are subject to the following requirements:*
>
> - *Reactor coolant pressure boundary components must meet the requirements for ASME Class 1 (Quality Group (QG) A) components as specified in ASME Code, Section III, except for those components that meet the exceptions described in Section 55a(c) of Reference 54. These components may be classified as Class 2 (QG B), or Class 3 (QG C).*
> - *Components classified as QG B and QG C must meet the requirements for Class 2 and 3 components, respectively, as specified in ASME Code, Section III."*
>
> *"6.3.3.2 Integrity of Reactor Coolant Pressure Boundary*
>
> *Reactor coolant pressure boundary components are designed and fabricated in accordance with the ASME Boiler and Pressure Vessel Code, Section III. A portion of the CVS inside containment that is defined as reactor coolant pressure boundary uses an alternate classification in conformance with the requirements of Section 55a(a) of Reference 54.*
>
> *Fabrication, examination, inspection and testing requirements as defined in Chapters IV, V, VI and VII of the ASME B31.1 Code are applicable and used for the B31.1 (Piping Class D) CVS piping systems, valves and equipment inside containment*."

35 Section 6.3 of the UK AP1000 PCSR has short sections for each of the main components of the reactor coolant system. However, the substantive argument is straightforward; the components are designed and built to the ASME code. The PCSR refers to the DCD for further detail.

36 In the DCD some arguments supporting claims for structural integrity are summarised in Sub-Chapter 3.1 in the 'AP1000 Compliance' responses to each of the US NRC General Design Criteria.

37 For instance the response to GDC 30 includes the following:

"*Reactor coolant pressure boundary components are designed, fabricated, inspected, and tested in conformance with the ASME Code, Section III. A portion of the chemical and volume control system that is defined as reactor coolant pressure boundary uses an alternate classification in conformance with the requirements of 10 CFR 50.55a(a)(3). The alternate classification is discussed in Section 5.2*."

And the response to GDC 31 includes the following:

"*Control is maintained over material selection and fabrication for the reactor coolant pressure boundary components so that the boundary behaves in a nonbrittle manner. The portion of the chemical and volume control system that uses an alternate classification is not required to meet the requirements to prevent brittle failure. The reactor coolant pressure boundary materials exposed to the coolant are corrosion-resistant stainless steel or nickel-chromium-iron alloy. The nil-ductility transition reference temperature of the reactor vessel structural steel is established by Charpy V-notch and drop weight tests in accordance with 10 CFR 50, Appendix G (Reference 1). See Section 5.3 for additional information.*

The following requirements are imposed in addition to those specified by the ASME Code, Section III.

• A 100 percent volumetric ultrasonic shear wave test of reactor vessel plate and a post-hydrotest ultrasonic map of welds in the pressure vessel are required. Cladding bond ultrasonic inspection to more restrictive requirements than those specified in the ASME Code, Section III is also required in order to preclude interpretation problems during in-service inspection.

• In the surveillance programs, the evaluation of the radiation damage is based on pre-irradiation testing of Charpy V-notch and tensile specimens and post-irradiation testing of Charpy V-notch, tensile, and l/2T compact tension specimens. These programs are directed toward evaluation of the effect of radiation on the fracture toughness of reactor vessel steels based on the reference transition temperature approach and the fracture mechanics approach, and are in accordance with ASTM, E-185 (Reference 2).

• Reactor vessel core region material chemistry (copper, phosphorous, and vanadium) is controlled to reduce sensitivity to embrittlement due to irradiation over the life of the plant. The fabrication and quality control techniques used in the fabrication of the reactor coolant system are governed by ASME Code, Section III requirements.

Allowable pressure-temperature relationships for plant heatup and cooldown rates are calculated using methods derived from the ASME Code, Section III, Appendix G. The approach specifies that the allowable stress intensity factors for vessel-operating conditions do not exceed the reference stress intensity factor for the metal temperature. Operating specifications include conservative margins for predicted changes in the material reference temperatures due to irradiation."

38 Whatever the type of failure of a metal pressure boundary component, the basic argument is the same, that together:

- material characteristics (so obviously depending on material selection); conservative design;
- manufacturing quality controls;
- construction;
- operation;
- maintenance and inspection;

will ensure the structural integrity claim is met. The evidence to support this basic argument is summarised in the UK AP1000 PCSR mostly in the way of information about the design. For example in terms of:

Material selection and characteristics: the material for the Reactor Pressure Vessel (RPV) is identified as ASME SA-508 Grade 3 Class 1 (Table 5.2-1 of the DCD);

Conservative design: the design code for reactor coolant pressure boundary components is the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (B&PVC) Section III (DCD 5.2.1.1).

39 The nature of the arguments and evidence supporting the claims for structural integrity in the UK AP1000 PCSR / DCD (essentially the DCD) could be described as conformance with good nuclear engineering practice and sound safety principles - using the concept of ‘defence in depth’ and with safety margins (see SAPs SC.4 Paragraph 92(c) and (d) [Ref. 8]).

40 It is not appropriate here to repeat the information in the UK AP1000 PCSR / DCD that could be described as the evidence supporting the claims and arguments in the safety case. However, it can be stated that overall this is a relatively mature area of engineering for PWRs worldwide. That is, for the major nuclear safety significant pressure vessels and piping, the materials selected, the design rules used, the manufacture and fabrication methods used and the types of examination and tests conducted during manufacture are consistent with industry practice that has been largely stable for the past 20 years or more.

41 For example, the types of materials, design rules etc proposed for the UK AP1000 are similar to those used for the Sizewell B PWR, construction of which started in 1987, with commercial operation from late 1995. For Sizewell B, the materials of construction and design rules had been determined by 1984.

42 As will be seen later, in Section 5 of this assessment report, my assessment has concentrated on specific aspects of the proposed design for nuclear safety significant metal pressure boundary components.

3 STANDARDS AND CRITERIA

43 I have based my assessment of the structural integrity aspects of the UK AP1000 PCSR, (actually the supporting DCD) primarily on the following:

Safety Assessment Principles for Nuclear Facilities (the ‘SAPs’, Ref. 8);

Technical Assessments Guide - Integrity of Metal Components and Structures - T/AST/16 Issue 003 (Ref. 9).

44 For the SAPs (Ref. 8) the main relevant part is “Integrity of Metal Components and Structures” in paragraphs 238 to 279, involving Principles EMC.1 to EMC.34. Other parts of the SAPs have some relevance to this assessment. For example, another part of some relevance is “Safety Classification and Standards” in Paragraphs 148 to 161, involving Principles ECS.1 to ECS.5.

45 In carrying out their assessment, ND Inspectors are asked to consider whether risks have been reduced As Low as Reasonably Practicable (ALARP). The SAPs in Paragraph 14 state:

“*The principles are used in judging whether ALARP is achieved…Priority should be given to achieving an overall balance of safety rather than satisfying each principle or making an ALARP judgement against each principle. The judgement using the principles in the SAPs is always subject to consideration of ALARP*.”

46 SAPs Paragraph 93 states:

"*To demonstrate ALARP has been achieved for new facilities, modifications or periodic safety reviews, the safety case should:*

a) identify and document all the options considered;

b) provide evidence of the criteria used in decision making or option selection; and

c) support comparison of costs and benefits where quantified claims of gross disproportion have been made."

47 Some further guidance on ALARP is provided in the SAPs in the part on "Numerical targets and Legal Limits". The SAPs define "Basic Safety Levels" (BSL) and "Basic Safety Objectives" (BSO). In terms of numerical limits such as radiological dose and frequency of occurrence, BSOs are lower (that is more onerous) than BSLs.

48 SAPs Paragraph 571 states:

"*It is HSE's policy that a new facility or activity should at least meet the BSLs. However, in meeting the BSLs the risks may not be ALARP. The application of ALARP may drive the risks lower*…."

49 SAPs Paragraph 573 states:

"*The BSOs form benchmarks that reflect modern nuclear safety standards and expectations. The BSOs also recognise that there is a level beyond which further consideration of the case would not be a reasonable use of ND resources, compared with the benefit of applying the effort to other tasks…..The dutyholder, however is not given the option of stopping at this level. ALARP considerations may be such that the dutyholder is justified in stopping before reaching the BSO, but if it is reasonably practicable to provide a higher standard of safety, then the dutyholder should do so*."

50 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.

51 The UK 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 Paragraph 93, as quoted above). Of course design decisions will have been made, but it is difficult now to 'back fit' ALARP to the design. It might be possible to examine individual important areas to determine if the situation is consistent with ALARP.

52 In carrying out my assessment, I have based my judgements of the technical aspects of structural integrity on the guidance provided on ALARP. I have interpreted the guidance to reach a judgement to apply to the balance of all the factors which contribute to the structural integrity safety case.

53 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 highest claims of highest structural integrity, I have examined whether:

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

54 For these highest claims of structural integrity, I have not sought 'perfection'; rather I have sought to determine the design is 'adequately safe' (Ref. 8 SC.4 Paragraph 91[a]) within the context of the claims of the safety case.

4 GENERAL MATTERS RELATING TO ND ASSESSMENT

4.1 Outcome of Assessment in GDA Step 2

55 The assessment reported here follows on from the Step 2 assessment, the assessment report for which was completed in early 2008 (Ref. 13). The GDA Step 2 assessment for the UK AP1000 was based on Ref. 1 but at Revision 1. Ref. 1 Revision 2 differs from Revision 1 almost entirely due to the addition in Revision 2 of metric units, but both are essentially Revision 16 of the DCD.

56 The GDA Step 2 assessment in the structural integrity area was brief (about 4 staff-days). In the GDA Step 2 assessment of the UK AP1000 structural integrity, the sections have the following headings and some of the topics raised as likely to require further consideration were identified as:

1. Pressure Boundary Components - Achievement of Integrity as a Contribution to the Overall Safety Justification
2. Requirements Additional to Code Requirements
3. Reactor Pressure Vessel
4. Steam Generators
5. Pressuriser
6. Reactor Coolant Pumps
7. Core Make Up Tanks and Accumulators
8. Piping - General Comments
9. Piping - Leak Before Break Evaluation Procedure
10. Reactor Coolant Loop Primary Pipework and Surge Line
11. Main Steam Lines
12. Overpressure Protection
13. ASME Code Edition
14. Load Combination and Stress Limits
15. In-Service Inspection

57 The assessment report raised a number of questions in some of the above areas. The sections containing questions were: 1, 4, 5, 6, 11, 12 and 14. To close out the questions, at least for GDA Step 2, the questions were raised formally with Westinghouse and they provided responses. A summary of the questions and responses is given in Table 7.

58 The responses to the questions are reasonable. However, these were specific questions for clarification and not intended to be the end of assessment in any particular area.

59 It will be seen from the titles of my GDA Step 3 Regulatory Observations that to some extent or other, I have carried forward assessment in most of the above Step 2 areas. Areas that have not featured explicitly in Step 3 Regulatory Observations are 13, 14 and 15. Item 13 is dealt with in Section 5.4 below, item 14 is dealt with in Section 5.3 below and item 15 is dealt with in Sections 4.2 and 5.3 below in the discussion of coverage of pre-service examination.

4.2 GDA Step 3 Assessment Compared with Project initiation Document (PID)

60 My PID for GDA Step 3 (Ref. 14) was written as an overall plan to cover, at the time, three different designs (two Pressurised Water Reactors [PWRs] and a BWR).

61 Table 1 of my PID (Ref. 14) sets out the main topic areas and how they will be considered in detail in GDA Step 3 and an outline of how they will be dealt with in GDA Step 4. An amended version of Table 1 of my PID for Step 3 only and for an AP1000 type PWR only is given in Table 8 here.

62 In Table 8, the topic headings are (number 1 was not used in the original table, see footnote to Table 8):

2. Components and Systems to be Considered
3. Level of Integrity Required for Nuclear Safety Claim
4. Safety Classification and Standards - Including Quality Assurance
5. Potential Failure Modes
6. Potential In-Service Degradation Modes (linked with 17. below)
7. Analysis - Design Analysis, Fracture Mechanics Analyses
8. Loadings
9. Materials - Choice and Specifications
10. Fabrication Design and Processes
11. In-Manufacture Examinations - Scope, Extent. Qualification of Procedures, Equipment and Personnel
12. Procedural Control of Design, Manufacture and Installation
13. In-Manufacture Inspection
14. Pressure System - Discharge and Flow Aspects
15. Pre-Service Examination - Scope, Extent. Qualification of Procedures, Equipment and Personnel
16. Definition of Operating Envelope
17. Establish In-Service Monitoring, Examination and Testing Requirements (linked with 6 above)

63 The comments in Table 8 against each of the above topic headings indicate there are varying degrees of depth of information needed for assessment in GDA Step 3. Table 9 shows how I have dealt with each of the topic areas in Table 8, mostly in terms of the Regulatory Observations (ROs) I have raised. Note that the only topic area where no substantive assessment has been made is 15, Pre-Service Examination. This has been deferred to later in GDA, and can be dealt with as part of the consideration of in-service examination.

4.3 Requesting Party Response to What is Required for GDA Step 3

64 Ref. 10 Page 14 sets out what the Requesting Party (RP) is required to do for GDA Step 3. There are two fundamental requirements:

1. provide a Pre-Construction Safety Report (PCSR); and
2. respond to questions and points of clarification raised by ND during its assessment.

In Ref. 10 there is also a list of requirements for the PCSR, items 3.1 to 3.13.

65 From Section 2 above, it is clear Westinghouse has provided a PCSR, but only after the start of GDA Step 3, Ref. 3 being Revision 0 of the PCSR and dated December 2008. The subsequent Revision 1 of the PCSR (Ref. 7) was issued in March 2009 and contained only minor changes. The DCD document (Ref. 1) was available from the start

of GDA Step 3; this was revised in February 2009 (Ref. 6). By itself, the PCSR document (Refs 3 and 7) would not be sufficient for this assessment. The DCD does provide information relevant to this assessment; the PCSR makes extensive reference to the DCD.

66 From Section 1 above it is clear there have been some responses from Westinghouse to ND questions and points of clarification.

67 For this assessment of structural integrity, the requirements for the PCSR in Ref. 10 (3.1 to 3.13) have varying significance. The main requirements for the PCSR relevant to this assessment of structural integrity are given below, with commentary on the extent of meeting these requirements in italics after each item - on the basis of the combination of the PCSR and DCD documents:

3.1 definition of the documentary scope and the extent of the safety case:

As stated above, the PCSR as submitted (Refs 3 and 7) on its own would not be an adequate basis for this assessment. However the PCSR makes extensive reference to the DCD and the latter document contains useful information. However the DCD is not structured on the basis of 'claims-arguments-evidence'.

3.3 Responses to any issues outstanding from Step 2:

There were no issues outstanding from Step 2, in the sense of questions outstanding. Most of the topics raised in the Step 2 assessment report are continued as topics in GDA Step 3.

3.4 Sufficient information to substantiate the claims in Step 2 (in the Preliminary Safety Report - for the UK AP1000 this was called the UK AP1000 Safety, Security and Environmental Report - actually US DCD Revision 16):

The substantive basis of Step 2 was Revision 1of Ref. 1, and the substantive basis for GDA Step 3 was Ref. 1 (i.e. at Revision 2). As noted above the only real difference is the inclusion of metric units. So one might expect the claims and information to be the same between Steps 2 and 3. For structural integrity, my interpretation of the claims in Step 3 are the same as in Step 2 so the issue is whether the PCSR substantiates its own claims. This is a fundamental aspect of this assessment and is reported in another section.

3.5 Sufficient information to enable ND to assess the design against all relevant SAPs:

There is information in the DCD to provide a starting point for ND assessment. Clearly this assessment has raised questions of substance and clarification, so by definition the DCD as originally submitted was not by itself sufficient. And the PCSR did not fill the gaps.

3.6 A demonstration that the detailed design proposal will meet the safety objectives before construction or installation commences, and that sufficient analysis and engineering substantiation has been performed to prove that the plant will be safe:

For structural integrity, the DCD at a general level has a demonstration that the plant will be safe. There is a layer of design specific documentation below the level of the PCSR / DCD (e.g. equipment specifications, design reports, and supporting documents to the design reports) that need some examination to confirm sufficient analysis has been completed. For structural integrity, it is anticipated this lower level of documentation will be for GDA Step 4.

3.7 Detailed descriptions of system architectures, their safety functions and reliability and availability requirements:

> *For structural integrity, this is taken to mean a description of the components in terms of function, size, shape, materials of construction, design loadings, design codes used, and so on. In these terms the DCD provides sufficient descriptions as a starting point for GDA Step 3 assessment.*

3.8 Confirmation and justification of the design codes and standards that have been used and where they have been applied, non-compliances and their justification:

> *For structural integrity there is a clear statement as to the design code to be used - the American Society of Mechanical Engineers (ASME) design code. The applicable edition of the ASME design code varies somewhat depending on the component (generally for pressure parts the 1998 edition with addenda to 2000, but 1989 edition and 1989 addenda for pipework - DCD 5.2.1.1; for the steel containment shell the 2001 edition with 2002 addenda - DCD 3.8.2.2). In general terms this code is clearly justified as an appropriate code to use. The ASME III code has been used (with adaptation) in the UK for Sizewell B.*

3.10 Justification of the safety of the design throughout the plant's life cycle, from construction through operation to decommissioning, and including the on-site spent fuel and radioactive waste management issues:

> *For structural integrity, safety through life depends mainly on operating within the design envelope of the relevant pressure boundary components and monitoring for potential degradation mechanisms. Operation within the design envelope is primarily through compliance with 'Technical Specifications'. An obvious example of potential degradation is neutron irradiation embrittlement of the Reactor Pressure Vessel steel material adjacent to the reactor core. The DCD in Chapter 16 sets out Technical Specifications for the AP1000. And DCD Chapter 5.3 includes coverage of the RPV materials surveillance programme and the RPV pressure-temperature limit curves.*

3.11 Identification of potentially significant safety issues raised during previous assessments of the design by overseas nuclear safety regulators, and explanations of how their resolution has been or is to be achieved:

> *Neither the UK AP1000 PCSR nor the DCD covers this explicitly. However the material supplied for GDA Step 3 includes the United States Nuclear Regulatory Commission (US NRC) Final Safety Evaluation Report (FSER) (up to DCD Revision 15) and this incidentally provides some insight to 'issues' raised, mainly by reference to Requests for Additional Information (RAIs).*

3.12 Identification of the safe operating envelope and the operating regime that maintains the integrity of the envelope:

> *For structural integrity, this overlaps with Item 3.10 and the comments above for 3.10 apply.*

3.13 Confirmation of:

(a) which aspects of the design and its supporting documentation are complete and are to be covered by the Design Acceptance Confirmation;

(b) which aspects are still under development and identification of outstanding confirmatory work that will be addressed during Step 4.

> *Given the mature nature of design aspects of metal pressure boundary components it can be taken that as far as the Requesting Party is concerned*

the design is complete. Indeed procurement of pressure boundary components for at least one power station based on the AP1000 design is underway.

4.4 What HSE will do in GDA Step 3

68 Ref. 10 Page 15 sets out what HSE will do in GDA Step 3. There is one fundamental requirement:

> "*Undertake an assessment, on a sampling basis, primarily directed at the system level and by analysis of the Requesting Party's (RP's) supporting arguments. The scope will be partly defined by experience in Step 2 and the issues arising in that step*."

69 In Ref. 10 there is also a list of what this sampling assessment should include, Items 3.14 to 3.26.

70 It will be seen in Section 5 below that I have undertaken an assessment, on a sampling basis of the structural integrity aspects of Westinghouse's PCSR and supporting material. To consider even the general claims for structural integrity involves considering individual components and the term 'directed at the system level' is not the way I would describe this assessment. However, my assessment has started with the important claims for structural integrity and delved into the supporting arguments for these claims. My assessment has been on a sampling basis. I have addressed the most important components but have concentrated on specific technical aspects and delved into the detail of arguments and evidence to varying extents. I believe my sampling is qualitatively consistent with the nature of the claims, arguments and evidence put forward by Westinghouse.

71 For this assessment of structural integrity, the requirements for the assessment in Ref. 10 (3.14 to 3.26) have varying significance. The main requirements relevant for this assessment of structural integrity are given below, with commentary on the extent of meeting these requirements in italics after each item:

3.14 Consideration of whether the design is likely to meet the RP's design safety criteria and reduce risks As Low As Reasonably Practicable (ALARP):

This is in two parts. For this assessment of structural integrity, consideration of whether the design is likely to meet the design safety criteria is interpreted to mean will the design meet the claims made for structural integrity. Whether the design will reduce risks ALARP has already been discussed in Section 3 above. The design of the AP1000 has evolved over a number of years (more than 15 years if the AP600 is taken as the starting point) and did not explicitly include consideration of ALARP. Design options will have been considered and choices made, but that might not amount to reducing risks ALARP. The best that can be done now is to consider important aspects of the design and investigate whether other options exist and whether a change on the basis of ALARP is indeed now practical. This sort of investigation has been part of this assessment, in particular for the region of the Reactor Pressure Vessel subject to neutron irradiation and the manufacturing route for the Reactor Coolant Pump casings.

3.15 Undertaking an initial assessment of the scope and extent of the arguments in each of the technical areas, including the generic site envelope:

This is almost a restatement of the general requirement for an assessment. Here, assessment of the scope and extent of the arguments has been made for the structural integrity area.

3.18 Deciding on scope and plan of further assessment:

This assessment report provides recommendations on further assessment, in terms of closing out GDA Step 3 matters and also starting on the next level of detail in GDA Step 4.

3.19 Assessment of the quality assurance (QA) arrangements, including:

(a) QA arrangements for the early manufacture of long lead time items important to safety.

Several of the components falling under this assessment and important to safety are likely to be 'long lead time items'. QA arrangements are relevant to such components. ND has addressed this matter in a general way, producing an Assessment Guide (T/AST/077 Ref. 25). I have contributed to this Technical Assessment Guide. My understanding is details of QA arrangements for Long Lead Time Items are likely to be discussed with licensees or proto-licensees, rather than the RPs.

3.20 Identification of research needs and setting up of longer-term research or contract support to complement Step 4:

The structural integrity aspects of the design and safety case of the UK AP1000 are based on many years of Pressurised Water Reactor (PWR) experience of primary and secondary pressure boundary component technology and operating experience. There are no really novel aspects of the pressure boundaries of the UK AP1000, compared with this body of experience. For this technical area no significant research needs have been identified. There will almost certainly be a need for technical support contracts in Step 4, but these will not be in the nature of 'research'.

5 ND ASSESSMENT GDA STEP 3 - STRUCTURAL INTEGRITY

5.1 Overview of Assessment

72 The specific aims of GDA Step 3 assessment are listed in Section 1. An outline and overview of the nature of the UK AP1000 safety case is given in Section 2. The standards and criteria used for this assessment are explained in Section 3. Section 4 explains how this GDA Step 3 assessment relates to the earlier Step 2 assessment and how this Step 3 assessment aligns with the Project Initiation Document (PID). Section 1 also gives a summary of the main milestones in this GDA Step 3 assessment.

73 With the information provided in the AP1000 DCD and further information supplied in meetings and by correspondence I have been able to make a meaningful assessment of the structural integrity aspects of the safety case.

74 It has been clear from the outset of the GDA process (at least from the start of Step 2) that so far as metal pressure boundary components are concerned (both primary and secondary circuits), this is a relatively 'mature' technological area of Pressurised Water Reactors (PWRs). For this technical aspect, the AP1000 design in terms of overall design, materials, fabrication and operation, is not that different from PWRs that entered service 10 to 15 years ago, and that were therefore designed about 20 years ago, or more.

75 Given the maturity of this aspect of PWR design, I have not started this assessment from a 'zero base'. Instead I have sought to confirm that general good practice has been used, taking account of international experience, but including the consideration of ALARP. In the UK the Sizewell B PWR has operated commercially since October 1995. Construction of Sizewell B started in 1987 and was preceded by design work extending back a number of years. The safety of the Reactor Pressure Vessel for a prospective UK PWR had been

the subject of debate for a number of years, including industry study groups (Refs 20 to 22). The subject of PWR Reactor Pressure Vessel integrity also featured in the Public Inquires for the Sizewell B and Hinkley Point C stations (Refs 23 to 24).

76 The Sizewell B pressure boundary structural integrity, both in achievement and demonstration of integrity, introduced a number of additional requirements over and above the standard design code requirements (in the case of Sizewell B, the American Society of Mechanical Engineers [ASME] code). In terms of As Low As Reasonably Practicable (ALARP), i.e. what is reasonably practicable, I have to take the Sizewell B approach to structural integrity into account; especially as ND licensed the station in part on the approach to structural integrity. However, I have to have regard to subsequent experience and developments. What was implemented for Sizewell B was clearly practicable, because it was done, and almost certainly remains so; but whether all aspects are still *reasonably* practicable is something to consider in this assessment. In short, for this assessment I have taken the Sizewell B approach to structural integrity as a precedent, but not necessarily a paragon in every detail.

77 After an initial, general examination of the documentation, I concentrated on a specific number of aspects and formulated these into a number of ‘Regulatory Observations’ (ROs). The ROs are listed in Table 1 and are presented in Annex 2. The ROs were the basis of communicating initial outcomes of the assessment with Westinghouse.

78 The ROs do not all have equal significance. At the time of initiating the ROs I was aware they were not of equal significance. And as a result of responses from Westinghouse, the relative significance of some of the ROs has changed.

79 Some of the ROs are long, with a degree of ‘background’ information included to explain how matters have been dealt with in the UK in the past. This might be construed as ‘assistance to the licensee’ in the terms used in the ND guidance on the assessment process (Appendix to Ref. 15: The Mechanics of Assessment, paragraphs 1.21 to 1.23). Relevant extracts from this guidance are:

> “*1.21 A situation which often occurs after a legitimate objection has been raised by an assessor and has been accepted, is that a licensee will ask for assistance with respect to what needs to be done to satisfy the assessor's concern. This is a fair question for a licensee…… It is potentially a dangerous question for an assessor however, since it threatens their independence….. Hence assessors should try, at least initially, to restrict their advice to clarification of the safety principle which is being pursued rather than to the identification of specific engineering solutions.*
>
> *1.22 However it is unhelpful and against the principle of openness (and also understandably regarded as perverse by licensees) to insist on keeping a potential solution secret, in the hope that it will occur independently to the licensee.*
>
> *1.23 A way round this dilemma is to explain clearly the safety concern that underlies the objection, and to put forward one's idea for satisfying the concern on the strict understanding that although it* appears *to suffice, no guarantee of acceptability is to be assumed by the licensee, and that if it is taken up then it remains the licensee's responsibility to justify it, and it will be assessed as the licensee's own proposal*…..”

80 Here we are not dealing with a licensee, but a Requesting Party (RP) and obviously an RP that has not previously interacted with ND. Hence, in setting out the ROs and in discussions and correspondence, I have erred on the side of ‘being helpful’. I have not offered solutions. But where relevant, I have indicated how matters have been successfully dealt with in the UK in the past, and on occasion how things might have moved forward. Whether Westinghouse considers the information I have provided as being helpful is a matter for them.

81 Most of the significant activity for my assessment is captured in the ROs. However, I have also included a consideration of Steam Generator tubing integrity, mainly through a

Technical Framework Contract, see below. From my understanding of operational performance with the tube material for the AP1000, I did not have concerns about the integrity of the tubing. However, given past interest in the UK, I felt it prudent to review this matter. I have not pursued other historical issues, where the design takes account of past experience and current understanding of the steps to avoid such issues. Examples of such historical issues are:

- Pressuriser surge line behaviour due to thermal loadings.
- Reactor Pressure Vessel head control rod drive mechanism adapter tube degradation.
- Thermal fatigue due to mixing (often rapid cycling) of hot and cold water within pipework systems.

For the last in the above list, thermal fatigue in pipework, there is a good understanding of the general factors leading to such situations, though predicting individual areas of pipework systems that might be susceptible is not a precise science.

5.2 Technical Framework Contracts

82 It was clear from the outset of my assessment that there would be a need for some technical support in specialist areas. In the event I established five contracts relevant to the assessment of the UK AP1000 design. These contracts cover the following subjects, with an indication of their relevance to the Regulatory Observations (ROs) I have raised:

1. Qualification of manufacturing examinations (2 contracts) (RO-AP1000-19);
2. Neutron irradiation embrittlement of the Reactor Pressure Vessel (RO-AP1000-22);
3. Metallurgy of ferritic steels for main primary circuit pressure vessels (RO-AP1000-21);
4. Steam Generator tubing material and manufacturing processes.

83 Items 1 to 3 above are associated with aspects of some of the more significant Regulatory Observations. I had no particular concerns about the choice of Steam Generator tubing for the AP1000. But I judged it prudent to include a contract for Item 4 above, given interest in previous consideration in the UK of this aspect of PWR structural integrity.

84 The work addressing Items 1 and 4 above is generic to both designs under consideration in GDA Step 3. Items 2 and 3 are such that it is necessary to consider each design specifically. One contractor provided support for Item 2 and produced separate reports for each of the two designs under consideration. The same approach has been used for Item 3.

85 The work for the contracts covering Items 1 to 4 above has been completed and final reports received. These are Refs 16 and 17 (Item 1) and Ref. 18 (Item 2), Ref. 45. (Item 3) and Ref. 44 (Item 4).

5.3 Summary of Assessment

86 In general, I have dealt with the Topics in Table 9 through the Regulatory Observations (ROs) listed in Table 1. These ROs are not all of equal significance. Factors that determine the significance of these ROs are:

1. the safety significance;
2. the amount of work required on the part of the Requesting Party (RP) to respond to the points raised in a Regulatory Observation (RO);

3. the implications for potential changes to design, specifications, or safety case claims, arguments and evidence.

The overall significance of an RO is clearly somewhat subjective and can involve any combination of the above three factors.

87 In my view, the relative ranking of the Regulatory Observations in Table 1 is as follows, going from highest to lowest:

1st Rank: RO-AP1000-18, RO-AP1000-19

2nd Rank: RO-AP1000-22, RO-AP1000-21, RO-AP1000-20

3rd Rank: RO-AP1000-29, RO-AP1000-30

4th Rank: RO-AP1000-23, RO-AP1000-24

5th Rank: RO-AP1000-25, RO-AP1000-26, RO-AP1000-27, RO-AP1000-28

In the 1st, 2nd and 3rd rank lines above, I have ordered the relative significance with the highest mentioned first, though in the case of the 1st Rank, I regard both ROs as of equal significance in part because they are inter-linked.

88 The nature of the responses by Westinghouse to the ROs could range from providing further or better arguments and evidence to support a claim, acknowledgement of a claim that was in the safety case but not explicitly declared, or a commitment to consider making a physical change to an aspect of the design or manufacturing specification of the plant.

89 It will be seen from Table 9 that the Regulatory Observations address most of the topics listed in my PID (Ref. 14). The exceptions are dealt with below.

90 A notable exception is Topic 8 - Loadings in Table 9. The AP1000 DCD (Refs 1 and 6) deals with loading conditions in Sub-Chapter 3.9. The DCD Sub-Chapter 3.9, Section 3.9.1.1 describes Design Transients including the definition of operating conditions, and the terminology of Level A, B, C and D Service Conditions and Testing Conditions. These have essentially the same meaning as previous usage in the ASME code of Normal, Upset, Emergency, Faulted and Test Conditions. By definition, all are transient loading conditions, that is involving a rate of change with time of pressure and temperature or mechanical load.

91 In sections 3.9.1.1.1, 3.9.1.1.2, 3.9.1.1.3, 3.9.1.1.4 and 3.9.1.1.5 of DCD Sub-Chapter 3.9, there are lists of specific operating conditions for (with number of transients in brackets):

Level A Service Conditions (Normal Conditions) (19),

Level B Service Conditions (Upset Conditions) (11 - one with 3 cases),

Level C Service Conditions (Emergency Conditions) (6),

Level D Service Conditions (Faulted Conditions) (5),

Test Conditions (3)

92 Table 3.9-1 in the AP1000 DCD Sub-Chapter 3.9 lists the transients for Level A and B Service Conditions and lists the number of assumed occurrences over plant life of each transient; the latter are needed for fatigue evaluations.

93 Table 3.9-5 of AP1000 DCD Sub-Chapter 3.9 sets out the minimum design loading combinations for ASME Class 1, 2, 3 and CS systems and components. This list is of loads which cause primary stresses in terms of the ASME code (so thermal stresses are not mentioned). The various design transients are described in terms of overall plant behaviour, for example ‘Plant Heatup and Cooldown’. From my understanding of PWR design, the design transients included in the design of the AP1000, as set out in the DCD,

are a recognisable set, with no obvious omissions. Detailed pressure and temperature variations for each of these design transients are needed for the design analyses of the pressure boundary components. The determination of these pressure and temperature variations is outside the scope of the structural integrity area. This will be for others to assess. I do not see this as an area of substantial regulatory risk and I believe the detailed assessment can await GDA Step 4.

94 Overall I conclude the AP1000 DCD provides some information for the structural integrity aspects of loading conditions, and I am satisfied this gives an indication of coverage within the design of this aspect, at least for compliance with the ASME code.

95 In Table 9, Topic 4 mentions Quality Assurance (QA). An important part of past usage of the ASME code in the UK has been the adaptation to UK practice of those aspects in the ASME code relating to QA, including organisational relationships and responsibilities. Clearly, the ASME code has no legal basis in the UK (unlike in the USA through reference in individual State statutes and in the Code of Federal regulations). In the past, UK practice and the general nature of the ASME requirements for quality assurance have aligned quite closely. However, it could be argued the UK nuclear industry, and ND, have not explicitly defined specific quality assurance and organisational requirements in the past.

96 Following discussion within ND of organisational and quality assurance requirements in general (not just for pressure boundary components) it was decided to produce a Technical Assessment Guide on procurement. At the time of writing this Assessment Report, the Technical Assessment Guide is still in draft (Ref. 25). However, as this matter is being dealt with as a general matter, I have not pursued this aspect separately in my assessment (I have contributed to the Technical Assessment Guide, including an appendix specific to metal pressure boundary equipment).

97 As noted in Section 4.2 above, my GDA Step 3 assessment has not dealt with Topic 15 in Table 4 - Pre-Service Examination. This can be dealt with in GDA Step 4 along with in-service examination. For now the main point is there should be access to components for pre- and in-service examination. I have not found anything in the documentation to suggest access for in-service examination would be a significant or general problem.

98 In Sections 5.5 to 5.16 below, I describe the progress made with dealing with the Regulatory Observations (ROs), for simplicity I do this in number sequence rather than the ranking described above. The next section covers use of the ASME Code.

5.4 ASME Code

99 There is some experience with the ASME code within UK industry and ND has dealt with the ASME code in terms of the Sizewell B project.

100 The applicable edition of the ASME design code for the AP1000 varies somewhat depending on the component (generally for pressure parts the 1998 edition with addenda to 2000, but 1989 edition and 1989 addenda for pipework - DCD 5.2.1.1; for the steel containment shell the 2001 edition with 2002 addenda - DCD 3.8.2.2).

101 The ASME code provides a substantial set of rules for design and construction of mechanical components (essentially pressure boundary components, their supports and internals) of nuclear facilities. This body of rules has changed in specific areas over the years, but the basic technical content and breadth of coverage have not changed much since the 1983 edition that was the edition of reference for Sizewell B. Hence, the fact that the AP1000 design is based on ASME code editions a number of years old is not in itself significant.

102 In the case of the Sizewell B project, the ASME code was the basis of the design, but additional requirements were applied to some aspects of pressure boundary component

design and manufacture. This approach of building on the ASME code is not unique to the Sizewell B project, particular nuclear steam supply system vendors have for their own reasons added requirements over those of ASME, usually set out in the Equipment Specification for a component.

103 Since 1983, the ASME III code has changed in one notable area with the introduction of definitions for reversing and non-reversing dynamic loads. These terms are then used in revised rules for design analysis of piping. However, 10 CFR 50.55a regulations stipulate that for piping design the code editions which contain these revised rules cannot be incorporated by reference. Presumably this leaves open application of the revised rules for piping design analysis for dynamic loading on a case-by-case basis. However, Westinghouse claims the AP1000 piping design criteria are consistent with the ASME code up to the 1989 Addenda, with specific stress limits invoked for anchor motion in seismic events (DCD Tables 3.9-6, 3.9-7) supplementing the requirements in the ASME code up to the 1989 Addenda (TQ-AP1000-000028).

104 The ASME code requirements are comprehensive for materials, design, fabrication, manufacturing examination, testing and overpressure protection. However, there are Regulatory Observations (ROs) where final resolution might involve adopting specifications or procedures that are different from what is in the relevant parts of the ASME code. I see this as no different to how the ASME code has been used in the UK in the past, nor in principle different to how vendors using the ASME code have added their own requirements.

5.5 RO-AP1000-18. Categorisation of Safety Function, Classification of Structures, Systems and Components

105 This Regulatory Observation (RO) is linked to RO-AP1000-19; this RO is essentially concerned with identifying those components to which RO-AP1000-19 might apply. As indicated in Section 5.3 above, RO-AP1000-18 and RO-AP1000-19 together are overall the most significant ROs of those listed in Table 1.

106 RO-AP1000-18 is concerned with identifying those components which require the highest reliability - where the claim is the likelihood of gross failure is so low it can be discounted. For such components, SAPs (Ref. 8) Paragraphs 243 to 253 apply along with the associated EMC.1 to EMC.3, and with SAPs ECS.3 and EMC.4 to EMC.34 are applicable with maximum stringency.

107 The Safety Function Categorisation and Classification of Systems, Structures and Components are important, fundamental foundations in the development of a deterministic safety case. This Regulatory Observation has the more limited aim of identifying the sub-set of metal pressure boundary components, and potentially a few other metal components.

108 Based on my interpretation of explicit and implicit claims in the AP1000 DCD (Ref. 6), I offered Westinghouse the following list of potential candidate components that might fall under RO-AP1000-18:

1. Reactor Pressure Vessel
2. Core Support Structure - Lower Internals
3. Main Coolant Loop Pipework
4. Reactor Coolant Pump Bowl Casings
5. Pressuriser
6. Steam Generator Channel Head Shell, Tubesheet and Secondary Shell Pressure Boundary

7. Main Steam Lines Inside and Outside Containment
8. Accumulator Tanks
9. Reactor Coolant Pump Flywheels

109 The Main Coolant Loop Piping and the Main Steam Lines inside containment are identified in the AP1000 DCD as having a 'leak before break' methodology applied to eliminate the need to consider local dynamic force effects of pipe failure (i.e. guillotine pipe failure). This was raised as a topic in my GDA Step 2 assessment report (Ref. 13). The UK AP1000 PCSR (Ref. 7) in section 5.5.5 contains a useful acknowledgement of this as a matter which will require attention (see Section 2.2 above).

110 The draft of this RO was discussed with Westinghouse at the meeting on 4-6 February 2009 (Ref. 4), and again at the meeting held 2-3 June 2009 (Ref. 5). Westinghouse offered a schedule for responding to this RO. The schedule envisaged finalising the approach to the response and then holding a meeting with ND to discuss.

111 In the event the meeting with ND was held on 16 September 2009 (Ref. 47). At this meeting, Westinghouse explained an overall plan for a programme of work that will perform a hazard / consequences analysis the aim of which is to categorise pressure boundary components as either:

- covered by a 'consequences of failure' claim-argument-evidence basis, meaning that nuclear safety does not depend solely on the structural integrity of the component;
- nuclear safety rests on the claim-argument-evidence basis of the structural integrity of the component; such components are candidates for being dealt with under Regulatory Observation RO-AP1000-19 (see next section).

112 From the presentation in the meeting on 16 September 2009 and the document support to the presentation, I regard this as substantial progress within GDA Step 3 and a basis for going forward. Westinghouse confirmed their plan for the programme of work in letter WEC00101N (15 October 2009) (Ref. 48). Execution of the programme of work will extend well into GDA Step 4. I believe ND will want to take an interest in the detail of this work and how the programme of works progresses. So far as GDA Step 3 is concerned I think this RO has achieved its objective.

5.6 RO-AP1000-19. Avoidance of Fracture - Margins Based on Size of Crack-Like Defects. Integration of Material Toughness Properties, Non-Destructive Examinations During Manufacture and Analyses for Limiting Sizes of Crack-Like Defects

113 This Regulatory Observation (RO) gives my interpretation of ND's expectations, based on the Safety Assessment Principles, of the arguments and evidence for components required to have the highest structural integrity.

114 For the materials and components in question, there are two basic failure modes due to tensile stress:

1. plastic deformation, where the applied load exceeds the combination of material strength and wall thickness / shape, either by single load application or repeated loading causing incremental distortion;
2. propagation of a pre-existing crack-like defect in either a 'brittle' or 'ductile' mode.

Failure Mode 1 above is well controlled by the traditional, long-established requirements of design codes.

Failure Mode 2 above is unlikely but arguably is not as well controlled as mode 1 by design codes. This RO deals with this failure mode.

115 Avoidance of failure by propagation of crack-like defects is based on a ‘defence in depth’ approach of:

1. absence of crack-like defects at the end of the manufacturing process - confirmed by examinations during manufacture;
2. material toughness offering good resistance to propagation of crack-like defects - underpinned by minimum material toughness requirements in equipment specifications;
3. absence of in-service sub-critical crack growth mechanisms that could lead to the increase in the size of pre-existing defects; or in the extreme, nucleation and growth of defects from an essentially defect-free initial condition.

In Item 1 above the role of manufacturing examinations is emphasised. 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.

116 This approach requires manufacturing examinations that are shown to be capable of detecting and sizing crack-like defects of concern. 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.

117 For this approach, there are some fundamental supporting requirements:

<u>Materials Toughness:</u> The needs to be a basis for a conservative (lower bound) value of fracture toughness for end of life conditions. In some cases (e.g. shells of Reactor Pressure Vessel, 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;

<u>Qualification of Manufacturing Examinations:</u> Ultrasonic examination is the predominant means of examination for crack-like defects. The European Network on Inspection Qualification (ENIQ) provides a framework for such qualification.

As input to the qualification, a definition is required for the nature and size of defects to be found with high confidence. Usually, the qualification requirement will not be set at the theoretical smallest defect the technique can find. Instead the requirement is to set the qualification defect size less than the limiting defect size, by some margin.

Defect aspect ratios included in the qualification, and those used in the fracture mechanics analyses for limiting defect sizes should be consistent;

<u>Limiting Defect Size Analyses:</u> All relevant materials are ductile thus the analyses need to make use of elastic-plastic fracture mechanics methods.

118 The draft of this RO was discussed with Westinghouse at the meeting on 4-6 February 2009 (Ref. 4), and again at the meeting held 2-3 June 2009 (Ref. 5). Westinghouse offered a schedule for responding to this Regulatory Observation. The schedule envisaged finalising the approach to response and then holding a meeting with ND to discuss.

119 In the event the meeting with ND was held on 16 September 2009 (Ref. 47). At this meeting, Westinghouse explained an overall plan for a programme of work that will address the three aspects of this RO (Materials Toughness, Qualification of Manufacturing Examinations and Limiting Defect Size Analyses, see above).

120 The total programme will need time and resource to execute, and some aspects of full implementation will be dependent on one or more utilities making some form of commercial commitment to a UK AP1000.

121 From the presentation in the meeting on 16 September 2009 and the document support to the presentation, I regard this as substantial progress within GDA Step 3 and a basis for going forward. Westinghouse confirmed their plan for the programme of work in letter WEC00101N (15 October 2009) (Ref. 48). Execution of the programme of work will extend into GDA Step 4. Full implementation might extend beyond GDA Step 4. If so, some interim work might be done within GDA Step 4 to give confidence for final implementation. I believe ND will want to take an interest in the detail of this work and how the programme of works progresses. So far as GDA Step 3 is concerned I think this Regulatory Observation has achieved its objective.

5.7 RO-AP1000-20. Manufacturing Method for Reactor Coolant Pump Casings

122 The SAPs (Ref. 8) in Paragraph 262 indicates a general preference for forged austenitic stainless steel components over cast stainless steel.

123 The AP1000 DCD indicates the Reactor Coolant Pump casings are made as stainless steel castings. It should be noted that the AP1000 Reactor Coolant Pump casing is slightly unusual compared with previous PWR technology in that the pump casings are welded direct to the channel head of the Steam Generators (two pumps for each Steam Generator). The choice of casting as the manufacturing route for the casings is not related to this direct connection to the Steam generators. As will be seen below, casting has historically been the favoured manufacturing route worldwide for Reactor Coolant Pump casings of PWRs.

124 RO-AP1000-20 was raised to address two general matters regarding the manufacturing route for the Reactor Coolant Pump casings, namely:

1. Had an options study been conducted for the manufacturing route of the reactor coolant pump casings? What were the pros and cons of casting versus forging?
2. Assuming a casting manufacturing route, what specific measures were implemented to ensure a sound final product, in particular what measures were in place to confirm the structural integrity of any large repair welds that might need to be incorporated in the final product?

Westinghouse responded to Regulatory Observation RO-AP1000-20 in letter WEC00084N (20 August 2009) (Ref. 26).

125 The Reactor Coolant Pump design is the responsibility of Curtiss-Wright Electro-Mechanical Corporation, and much of Westinghouse's response to this Regulatory Observation is from Curtiss-Wright.

126 In terms of casting versus forging, this was reviewed by Curtis-Wright in early 2007. In summary the reasons for preferring a casting manufacturing route are:

- The casing can be made in one piece as a casting, hence no main seam welds.
- It is highly improbable that a pump casing could be forged in one piece. Even if a casing could be forged in one piece, it would require extensive machining of the inside and outside surface.
- If a pump casing was made from two or more forgings, they would have to be joined by main seam welds.
- A carbon steel forging would require the inside surface to be clad in stainless steel.

- There are only one or two suppliers worldwide who could in principle supply a forged one-piece pump casing, cost could be very high.
- Westinghouse and Curtiss-Wright together have experience of supplying over 300 Reactor Coolant Pumps with cast stainless steel casings.

127 I am not strongly persuaded by unquantified claims of cost. However, I agree that extensive experience of a manufacturing route is important and a one-piece casting which did not contain large repair welds could be the basis for a claim of high structural integrity. And there is only value in using a forging manufacturing route if it is capable of providing a sufficient forging reduction ratio and hence obtain a better metallurgical structure (a point not explicitly put in Westinghouse's response).

128 A single AP1000 Reactor Coolant Pump casing has a mass of about 16 tonnes, this is somewhat less than (about 55%) the mass of pump casing castings I have dealt with in the past. The smaller mass might ease some of the potential manufacturing issues raised here.

129 If a casting could be made with no need for repair work to achieve the final product, this sort of austenitic stainless steel casing as a casting would be unexceptional. However, historically, such castings have been known to require large repairs to remove various sorts of initial casting defects. The form of repair is to locally remove material from the casting in order to remove the defect and then to replace the removed material by depositing weld metal. Such repair work is a standard feature of casting production, and permitted by relevant codes. The size of excavations for such repairs can range from relatively minor depths (say 40mm in a 160mm thick wall) up to half wall thickness or even through wall. Such repair welds are done at a time when no further heat treatment of the component is possible. This means the deposited weld metal is left in the as-welded condition. The nature of the repair process is such that it the residual stresses in the weld metal would be expected to be tensile. And it might be expected that the fracture toughness of the deposited weld metal could be lower than the parent casting material.

130 The issue of the structural integrity of any large weld repairs in cast casings of Reactor Coolant Pumps would need to be addressed to some extent. But if the claim for Reactor Coolant Pump casings is that gross failure is so unlikely it can be discounted, then particular attention needs to be given to the integrity of any large repair welds.

131 In terms of base material of the casting, it is noted that the material selected could be CF8A from the ASME SA-351 Specification. The DCD, Ref. 6, in Table 5.2-1 indicates either CF3A or CF8A might be used. CF3A is also in the ASME SA-351 Specification. CF8A is the 'casting equivalent' of 304 stainless steel and CF3A is the 'casting equivalent' of 304L stainless steel. The 'A' designation represents high tensile strength. The chemical composition of CF3A and CF8A is further restricted within the composition limits of CF3 and CF8, respectively, to obtain a ferrite / austenite ratio that results in higher ultimate and yield strengths.

132 The main difference between CF8A and CF3A is the maximum content for carbon - 0.08% and 0.03% respectively. Carbon content can be a factor in thermal ageing during service of this type of stainless steel casting (Ref. 27). One way high carbon content can influence thermal embrittlement is when a preferential failure path occurs on the ferrite-austenite phase boundaries due to the presence of large phase boundary carbides (Ref. 27, Executive Summary).

133 Westinghouse contends that the main factors controlling thermal ageing are the Molybdenum content and the ferrite content (it is the ferrite structure which is changed by thermal ageing). Westinghouse cites EPRI work (MRP-175, Ref. 28) to claim that with a Molybdenum content of less than 0.5% and a ferrite content less than 20%, thermal embrittlement analysis is not required. The work in Ref. 27 seems to indicate a relatively

small difference between the age-saturated fracture toughness of CF3 and CF8 materials.

134 I am satisfied there is no compelling reason to select CF3A over CF8A as the casting base material, assuming the ferrite content is controlled as intended.

135 The ASME SA-351 Specification includes a number of possible Supplementary Requirements, some of which relate to manufacturing examination or repair welds. Some of these Supplementary Requirements are in SA-703 (Specification for Steel Castings, General Requirements, for Pressure-Containing Parts) which is referenced by SA-351. Westinghouse noted that the Equipment Specification for the Reactor Coolant Pump casings only cites one of these Supplementary Requirements, S17 “Tension Test from Castings”. But the ‘Material Ordering Document’ for the castings does require the following, which are similar to some of the Supplementary Requirements:

- Radiographic and liquid penetrant testing is required (cf S5 and S6).
- Purchaser approval of major weld repairs is required (cf S12).
- Weld repair charts are required (cf S20).
- Ferrite content is specified as in the range 8 to 20 FN (based on Hull’s equivalent factors) (cf S24);
- Some destructive testing of the first article poured was carried out (cf S2).

The above are important aspects, it does not matter that they are not called explicitly from the Supplementary Requirements of the Specification.

136 Radiography is the only volumetric examination method applied during manufacture, for both the base casting and any repair welds. One question put to Westinghouse concerns the capability of radiography to find crack-like defects. This question was put without distinguishing between base casting and repair welds, but is important for both. The response from Westinghouse was that the radiographic procedure complies with the requirements of the ASME Code. Depending on the claims for structural integrity of the Reactor Coolant Pump casings (i.e. whether the claim is gross failure is so unlikely it can be discounted - see Regulatory Observation RO-AP1000-18) I conclude this response might not be enough.

137 A further question relating to manufacturing examinations was in terms of whether ultrasonic examination would provide an additional element in the justification of integrity, at least for near surface regions. This might be targeted at any large repair welds. The response from Westinghouse was that the ASME code does not require ultrasonic examination for this material and Westinghouse does not currently have any additional requirements. Depending on the claims for structural integrity of the Reactor Coolant Pump casings (i.e. whether the claim is gross failure is so unlikely it can be discounted - see Regulatory Observation RO-AP1000-18) I conclude this response might not be enough.

5.8 RO-AP1000-21. Materials Specifications and Selection of Material Grade - Reactor Pressure Vessel, Pressuriser, Steam Generator Shells

138 This Regulatory Observation addresses materials specifications and selection of materials for the following major pressure vessels (materials defined in AP-1000 DCD, Ref. 6, Table 5.2-1):

Reactor Pressure Vessel (forgings SA 508 Grade 3 Class 1)

Pressuriser (forgings SA 508 Grade 3 Class 2)

Steam Generator Shells (primary and secondary circuit sides) (forgings SA 508 Class 1A or Grade 3 Class 2)

Steam Generator Channel Head (forgings SA508 Grade 3 Class 2).

139 In line with international practice for PWRs, the above vessels in the AP-1000 are specified to be made using a quenched and tempered low-alloy ferritic steel. The AP-1000 specifies forgings as the base material of construction.

140 I note the DCD (Ref. 6) in Sub-Chapter 5.3, Section 5.3.4.1, page 5.3-15, regarding the RPV inlet and outlet nozzles states:

"*These nozzles are forged into the ring or are fabricated by 'set in' construction*."

141 Where the nozzles are forged as one with the nozzle shell course, the nozzle shell course is referred to as an 'integral' design. From other, detailed information supplied by Westinghouse, I have assumed for this assessment that the Reactor Pressure Vessel (RPV) main nozzles will be of the set-in (also known as set-through) and not forged into the nozzle shell course. If an integral design was proposed, it would require specific assessment.

142 As an aside, the DCD (Ref. 6, Page 5.4-10) states the primary and secondary sides of the steam generator pressure boundaries are designed to ASME III NB (Class 1) even though the secondary side could be designed to ASME III NC (Class 2). But for ASME XI in-service inspection, the steam generator secondary side shells are treated as ASME III Class 2 components.

143 A point of reference for this part of my assessment is the material specification for the same vessels in Sizewell B. Before construction of Sizewell B commenced, a good deal of time and effort was put into reviewing the ASME SA 508 material specification. As a result of this review, several amendments were made to the chemical specification, see Table 1 of RO-AP1000-21 in Annex 2. Comparing ASME SA 508 Grade 3 Class 1 with the Sizewell B specification ('UK Usage of SA508 Class 3'), it will be seen that for the Sizewell B specification there is a lower maximum Carbon content, a lower maximum Nickel content, lower limits on Sulphur, Phosphorus and Silicon, a lower limit on Chromium and a limit on Cobalt (for activation reduction and hence lower dose to operators in outages). The right-most column of Table 1 of RO-AP1000-21 in Annex 2 shows the additional, vendor imposed, chemical composition restrictions for the AP1000 Reactor Pressure Vessel beltline forging material (listed in the DCD, Ref. 6, Table 5.3-1).

144 A complicating factor in comparing the material specifications of SA-508 through time is the change in grade / class nomenclature that was made some years ago. For the relevant material forms, the following table gives the current and former nomenclature:

SA-508 Current	SA-508 Former
Grade 1A	Class 1A
Grade 3 Class 1	Class 3
Grade 3 Class 2	Class 3A

145 The current Grade 3 Class 1 and Class 2 have the same chemical composition requirements in ASME SA-508 (2007 edition). Grade 3 Class 2 has higher minimum strength requirements than Grade 3 Class 1, slightly lower elongation and reduction in area requirements and slightly higher Charpy Impact energy requirements but at a higher temperature but slightly higher (and this was the distinction between the former Class 3 and Class 3A).

146 The DCD in 5.2.1.1 states that for most pressure boundary components the 1998 edition of the ASME code applies. The 1998 version of the SA-508 Specification is almost the same as the 2007 edition. The notable difference is the 2007 edition in the chemical composition includes Titanium, see later in this section.

147 For the Sizewell B plant, SA 508 Class 3 (designation then, now SA 508 Grade 3 Class 1) was also used for the Pressuriser and Steam Generator shells (as for the Reactor Pressure Vessel) but with a slightly different set of additional requirements, see Table 4 of RO-AP1000-21 in Annex 2.

148 These low alloy, quenched and tempered ferritic steel forging materials have been used for many years, and the chemical composition within each standard has remained substantially the same. However, some changes have been made since the Sizewell B vessels were specified and procured. For example, the chemical composition of ASME SA 508 for all Grades/Classes changed after 2004 to include a maximum on Titanium (Ref. 29).

149 According to Ref. 29, the role of Titanium is to act to control prior austenite grain size, which encourages a small grain size in the final quenched and tempered condition (helpful in both tensile and fracture toughness properties). Micro-alloying elements like Titanium can form carbide, nitride or carbonitride particles, stable at high temperatures and effective in pinning austenite grain boundaries. Other micro-alloying elements that have a similar effect are Vanadium and Niobium, at temperatures up to 1200^{o}C. Above 1200^{o}C Titanium still has the ability to control grain size. However, too much Titanium can have deleterious effects, hence the maximum level specification.

150 It will be noted that the historical UK usage of ASME SA508 Class 3 did not include Titanium.

151 It is worth recalling that the detailed chemical specifications of these materials, and their quench and temper heat treatments are intended to provide large, thick-walled finished forgings with minimum tensile and toughness properties that can be depended upon to apply reasonably well throughout the volume of the forging. Ref. 30 provides examples of how variations in heat treatment practice and the location where test specimens are taken can give rather different results.

152 There will inevitably be some variation through the volume of such large forgings in terms of tensile strength and fracture toughness properties. The key is to be sure of the minimum properties wherever they occur in the finished product.

153 Several matters have been clarified through dialogue with Westinghouse.

154 The vast majority of the Steam Generator primary and secondary side pressure boundary is made from SA-508 Grade 3 Class 2. Minor nozzles are specified as SA-508 Grade 1A.

155 For the Reactor Pressure Vessel it was confirmed that the Copper content of the following forgings is restricted to 0.06%: upper shell, lower shell, transition ring and the two circumferential welds at the top and bottom of the lower shell. With reference to Figure 1 here, I interpret these parts and welds as Upper Shell Course, Lower Shell Course, Transition Ring of Lower Head and Welds 3 and 2 respectively. This was confirmed by review of the Design Specification for the AP1000 RPV and the Material Purchase Specifications for the various forgings of the RPV.

156 I asked a question whether there was any substantive reason why a chemical composition similar to Sizewell B could not be used in AP1000. In reply, Westinghouse noted their specifications for SA-508 Grade 3 Class 1 for ‘core region forgings’ (as listed above) has additional restrictions on the following chemical elements:

- Phosphorous ≤ 0.01%.
- Vanadium ≤ 0.05%.

- Sulphur ≤ 0.01%.
- Nickel ≤0.85%.
- Chromium ≤ 0.15%.

and for non-core regions the above limits on Sulphur and Chromium are required by Westinghouse specifications.

157 The above reply from Westinghouse essentially reiterates information in the DCD which I had used illustrating the difference in material specifications in Table 1 of the Regulatory Observation RO-AP1000-21 (see Annex 2). The only new information above is the Westinghouse additional restriction on Chromium content. See the conclusion at the end of this section in paragraph 164.

158 I asked whether the Pressuriser and Steam Generator pressure shells could be manufactured from SA-508 Grade 3 Class1, rather Grade 3 Class 2. The response was this would be a significant change. SA-508 Grade 3 Class 2 has a higher strength (by about 12%) than Grade 3 Class 1. A change of material would require a complete re-design of the components, and their supports and surrounding modules and floors, because of the increased weight. Given this is the case, it puts emphasis on an adequate materials database for SA-508 Grade 3 Class 2, for use in the type of approach outlined in RO-AP1000-20.

159 For the Reactor Pressure Vessel I agree that SA-508 Grade 3 Class 1 is appropriate. I conclude the Westinghouse specification for the chemical composition of the forgings for the Reactor Pressure Vessel take account of the evolution of understanding of the factors of most importance in reducing susceptibility to neutron irradiation embrittlement. The exact ASME Code edition for material supply is still slightly vague; the Design Specification states the 1998 Edition with 2000 Addenda, the Material Purchase Specification states that the Code Edition will be identified in the Purchase Order.

160 For the Pressuriser and Steam Generator shells, Westinghouse has chosen SA-508 Grade 3 Class 2, this has the same chemical composition as Grade 3 Class 1 but has a higher specified strength (about 12% higher). I consider the main issue here to be the extent of available fracture toughness data for SA-508 Grade 3 Class 2 compared with Grade 3 Class 1. The fracture toughness properties of SA-508 Grade 3 Class 1 (formerly SA-508 Class 3) have been studied extensively worldwide for many years and reasonably extensive databases exist which allow lower bound toughness values to be determined with some confidence. I am not aware of the extent of fracture toughness data for SA-508 Grade 3 Class 2 (formerly SA-508 Class 3A). Perhaps there are metallurgical arguments to support a claim that toughness data for SA-508 Grade 3 Class 1 is applicable to SA-508 Grade 3 Class 2. The fracture toughness properties of SA-508 Grade 3 Class 2 are relevant to consideration of the Steam Generator and Pressuriser shells under RO-AP1000-19.

161 Forging material chemical composition and manufacturing detail have evolved through time and are specialised metallurgical matters. Given the nature of this topic and its fundamental importance, I decided it was necessary to take authoritative advice on the topic, using a support contract. The contract, with Prof J F Knott, has provided a report with clear advice (Ref. 45).

162 My summary of the main conclusions in Ref. 45 is:

1. Confirmation that the steels will be fully-killed and the process for de-oxidation should be sought.
2. Ingot casting practice and how much material is discarded (to remove segregation or inclusions) needs to be specified and available for review.

3. The degree of forging reduction achieved in manufacture of the various forged parts should be confirmed. It might be argued that the outcome of mechanical property tests justifies whatever forging ratio is achieved. However, it is not possible to comprehensively test every forging. The amount of forging reduction provides general support to an argument for 'quality of material'.
4. The question of limits on Arsenic, Antimony, Tin and Hydrogen should be taken further.
5. A maximum Carbon level of 0.2% is satisfactory for the cylinder region of the RPV. A slightly higher Carbon level may be necessary for deep hardenability of thicker sections, such as the vessel flange.
6. The maximum Carbon content of 0.3% of SA 508 Grade 1A gives concern with respect to welding - however only minor nozzles are specified in this material.
7. For the RPV materials (SA 508 Grade 3 Class 1), a Chromium level of 0.15% max is acceptable. For the Steam Generators and Pressuriser, the SA 508 Grade 3 Class 2 Chromium limit of 0.25% max is not significant (except as part of a general assessment of the propensity for stress relief cracking).
8. 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.
9. The Calcium content is a consequence of the steel making practice and cannot be considered as an element to be controlled in isolation (but low levels in the steel may be a good guide to effective removal of Phosphorous and Sulphur as slag).
10. A maximum level of Aluminium of 0.025% is reasonable. Maximum limits on Titanium and Niobium are reasonable. There is a need to have some micro-alloy present to control grain growth during welding.
11. 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.
12. A limit on Phosphorous of 0.01% is reasonable. There are indicators of thermal ageing embrittlement at lower Phosphorous levels; this points to the need for a comprehensive materials surveillance programme through life.
13. The proposed limit on Copper is crucial for minimising neutron irradiation embrittlement in those regions of the Reactor Pressure Vessel that receive a significant neutron fluence through life.
14. The maximum level for Nickel of 0.85% is appropriate.
15. The heat treatment sequences are as would be expected for this class of material.
16. Information on tensile properties at operating temperature and scatter in tensile properties within and between forgings should be considered.
17. The required minimum Charpy impact energy values are not ambitious but might be argued to be 'fit-for-purpose'.

163 For welding processes, Ref. 45 recommends details of weld and cladding qualification tests should be reviewed, the corrosion properties of cladding confirmed.

164 From these overall conclusions of Ref. 45, there may be a number of 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.

5.9 RO-AP1000-22. Reactor Pressure Vessel Body Cylindrical Shell Forging Material and Associated Circumferential Welds - Effects of Irradiation

165 Neutron irradiation embrittlement of the base materials and welds of PWR Reactor Pressure Vessels has historically been a significant issue, and remains so for older PWRs.

166 This is arguably the most significant ageing effect for PWR metal pressure boundary components. Ref. 19 agrees this is the key degradation mechanism to consider for a PWR RPV. The effect of neutron irradiation on the ferritic steels used in such RPVs is a shift in the brittle to ductile fracture transition temperature to higher temperatures. In the past, some have supposed that neutron irradiation has little effect on upper shelf toughness. A different view is expressed in support work completed for this assessment (see discussion of Ref. 19 later). But in any case, the shift in transition temperature probably has more general significance than reduction in upper shelf toughness.

167 If the shift in transition temperature is significant (or the initial transition temperature is high), the region over which the transition from brittle to ductile fracture behaviour occurs can approach temperatures of operation. For instance start-up conditions involve a change of metal temperature of the RPV from ambient temperature to full operating temperature. A high toughness transition temperature can mean the early phase of startup occurs with the RPV metal (adjacent to the core) temperature in the transition region. The same applies at the end of a shutdown sequence.

168 The issue of operation of ferritic steel nuclear reactor pressure vessels has been a significant regulatory issue around the world for 40 years or so. As an indication of this significance, ND published a Statement on the matter in 1995 (Ref. 31). Section 5 of Ref. 31 gives the ND position as:

- Clear safety benefits derive from operating on the upper shelf of the toughness transition curve to ensure ductile behaviour.
- RPVs must, for normal steady-state operation, operate on the upper shelf.
- For other conditions the RPVs should be on the upper shelf wherever possible. However, where upper shelf conditions cannot be achieved - e.g. during shutdown, start-up or limited duration transients - it is important that all uncertainties and conditions are considered and that adequate margins on toughness are shown.

169 There is now a good understanding of the factors that influence the response of the base and weld materials to neutron irradiation.

170 For a new PWR design, it is reasonable to expect the design of the Reactor Pressure Vessel to take account of the accumulated knowledge regarding neutron irradiation embrittlement. So far as reasonably practical, it would be expected the design would minimise the effect of neutron irradiation embrittlement.

171 Paragraph 262 of the SAPs (Ref. 8) Item (c) states that designs should consider avoiding welds in high neutron radiation locations.

172 Figure 1 here shows the locations of three circumferential welds in the body of the Reactor Pressure Vessel, Welds 1, 2 and 3. These welds join the ring forgings identified as the Upper Shell Course, Lower Shell Course and Transition Ring of Lower Head; there are no axial welds in the reactor Pressure Vessel. Welds 2 and 3 receive significant neutron irradiation over the 60 year design life, that is higher than $1x10^{19}$ n/cm^2 (E>1MeV). Weld 1 receives a much lower dose over the 60 year life.

173 The peak neutron dose occurs about mid-way between Welds 2 and 3, at core mid-height. From Figure 1 it will be seen there is no weld between Welds 2 and 3. This means there is no weld at the location of peak neutron dose. Clearly this in part meets the SAPs expectation mentioned above. However, Welds 2 and 3 do receive a life time dose that

can be expected to change the toughness properties of the welds. Of course the forging material will be subjected to the peak neutron dose, and as a result, its properties will also change.

174 In summary the peak neutron doses to the inner surface of the RPV after 60 years with 90% availability (54 effective full power years) are:

Location	Peak Neutron Dose neutrons/cm^2 (E>1MeV)
Mid-height (approx) of Lower Shell Course	9.76×10^{19} n/cm^2 (Ref. 2) 8.9×10^{19} n/cm^2 (WEC Ref. 26)
Weld 2	2.84×10^{19} n/cm^2 (Ref. 2) 2.5×10^{19} n/cm^2 (WEC Ref. 26)
Weld 3	1.25×10^{19} n/cm^2 (WEC Ref. 26)

As they are the most recent and direct from Westinghouse (WEC), I take the Ref. 26 values in the above table as the most representative.

175 The neutron dose level falls rapidly above and below the core. Weld 1 receives a neutron dose much lower than Weld 2. As it does not feature in dose-damage estimates, I assume the dose to Weld 1 is of order 1×10^{17} n/cm^2 or lower.

176 The neutron dose to the RPV varies significantly with axial and circumferential location, only a limited fraction of the vessel inner surface is subject to the peak neutron dose.

177 The AP1000 includes neutron shield pads on the outside of the core barrel to attenuate the neutron flux falling on the wall of the RPV. These neutron shield pads, the width of the water gap as now defined for the design and some limits on maximum power in fuel assemblies at the periphery of the core, all contribute to the peak neutron dose at end of life being slightly less than the maximum dose of 10×10^{19} n/cm^2 for the dose-damage correlation in US NRC Regulatory Guide 1.99 Revision 2 (Ref. 32, Figure 1). Another limit in Ref. 32 is 6×10^{19} n/cm^2 being the upper limit in the correlation for the decrease in upper shelf energy (Charpy impact energy) (Figure 2 of Ref. 32).

178 Looking to international precedent, I note that historically the approach in Germany for PWRs over 1000MWe (Konvoi series) was for a design limit on the neutron fluence to the RPV wall of about 0.5×10^{19} n/cm^2 after 40 years operation. This was achieved by using a relatively large diameter RPV and using a large water gap (Ref. 34). A limit of 1×10^{19} n/cm^2 appears in the RSK Guidelines (Ref. 35). The German Konvoi 1300Mwe (design dating from mid-1980s) RPV has an internal diameter of about 5000mm, the AP1000 RPV internal diameter is about 4040mm. I understand the actual in service limit for Konvoi series plants is about half the design limit, that is about 0.25×10^{19} n/cm^2 at 40 years, due to the implementation from early in service of a low leakage core configuration. Of course the Konvoi series RPV has a circumferential weld at about core mid-height.

179 I understand the amount of data supporting the US NRC Regulatory Guide 1.99 correlation (and other similar correlations) declines as the dose increases. In particular the extent of data to support the correlation at doses higher than 3×10^{19} n/cm^2 is relatively small. A statement attributed to a speaker at a Workshop in 2008 (Ref. 33) is that for US surveillance data, 92% of the data is at doses below 3×10^{19} n/cm^2.

180 The upper limit on neutron dose of 10×10^{19} n/cm^2 (E>1MeV) might be thought of as a 'Basic Safety Level' (BSL) in terms of the application of 'As Low As Reasonably Practicable' (ALARP) in this area. A 'Basic Safety Objective' (BSO) is more difficult to judge. On the basis of extent of data, a BSO of 3×10^{19} n/cm^2 might be reasonable, though in terms of what has been achieved elsewhere, a lower value might be argued.

181 I noticed Weld 2 has a higher predicted dose than Weld 3, and this is the other way round compared to the PWR I am familiar with (Sizewell B). The neutron dose to Weld 2 is influenced by the AP1000 core being positioned relatively low in the RPV and the use of fuel assemblies about 0.6m longer than for Sizewell B.

182 Although the dose estimates above include a restriction on fuel assembly power levels at the periphery of the core, these restrictions do not amount to a ‘Low Leakage’ or ‘In / Out’ fuel management strategy. Low Leakage core strategies for fuel management have now been used for many years in operating PWRs. In response to a question, Westinghouse estimated the reduction in peak neutron dose to the RPV by implementation of a ‘Low Leakage’ core configuration would be somewhat less than a factor of 2, depending on when such a configuration was implemented in service (TRIM Ref. 2009/263292). Implementation of a ‘Low Leakage’ core configuration in service could usefully decrease the neutron dose to the vessel, especially the maximum dose. This is obviously of most use for the forging locations, and there is less concern about the potential incidence of defects in the forgings. The neutron dose to the Welds 2 and 3 is lower but the Low Leakage core configuration would also have a beneficial effect for these welds.

183 There is no ‘cliff-edge’ effect anticipated if the operating life increased beyond 60 years. The correlations for changes in material toughness properties are smooth functions with increasing neutron dose (typically related to neutron fluence raised to a power less than 0.5). The neutron dose to any location on the vessel increase in direct proportion to the number of years operation.

184 In response to a question, Westinghouse stated (Ref. 26) that the neutron energy spectrum at both the RPV wall and the surveillance capsule locations are essentially the same as existing Westinghouse type PWRs.

185 According to Ref. 26 the AP1000 RPV materials surveillance programme now includes 1/2T-CT (0.5 inch thick compact tension) fracture toughness specimens, covering base material and weld metal, but apparently not heat affected zone material (HAZ). The plan is for many more Compact Tension (CT) specimens in each capsule than stated in Table 5.3-4 of the DCD (Ref. 6). This is a recent change from the original design of the surveillance scheme which did not include such CT specimens. Addition of these specimens means the surveillance capsules are now full. Adding further specimens would require an increase in the size of the holders and this would not be trivial.

186 On the basis of revised surveillance capsule locations on the outside of the core barrel (Ref. 26), Westinghouse states the lead factors for irradiation of the specimens in the capsules compared to the RPV wall lies between 1.8 and 2.3. These lead factors appear to be with reference to the maximum neutron dose location for the RPV wall, which is a forging or base material location. Weld metal specimens in the surveillance capsules will therefore have a higher lead factor (as the neutron flux to Welds 2 and 3 is lower than the peak for the forging). The lead factor for the weld specimens in the surveillance capsules over the welds in the RPV will be in the approximate range of 7 to 21. Ref. 19 states that for low Copper / low Phosphorous materials the surveillance scheme lead factors are acceptable (neutron flux affects the rate of Copper precipitation).

187 With a given neutron dose, minimisation of embrittlement centres on the chemical composition of the material of the base forgings and the welds. The importance of chemical composition is recognised in the AP1000 DCD. In particular, for irradiation embrittlement, the role of Copper, Phosphorous and Nickel and the need for upper limits on these elements is recognised in the DCD. In general, chemical composition of materials has been handled under Regulatory Observation RO-AP1000-21 (see above). To summarise, for the regions affected by neutron irradiation, the Westinghouse additional requirements for SA-508 Grade 3 Class 1 and the associated welds are (from DCD Table 5.3-1, Ref. 6):

Element	Base Metal Forging %	As Deposited Weld metal %
Copper	0.06	0.06
Phosphorous	0.01	0.01
Vanadium	0.05	0.05
Sulphur	0.01	0.01
Nickel	0.85	0.85

188 Even if there is still debate about details of the specification, clearly the above table shows that the effects of Copper, Phosphorus and Nickel have been taken into account in the Westinghouse additional requirements.

189 As pointed out at the beginning of this section, the main issue is where the toughness transition temperature lies at end of life, compared with the operating temperature region, the latter including start-up and shutdown and possibly other less likely transients. The end of life toughness transition temperature depends on the start of life toughness transition temperature and the shift to higher temperature due to neutron irradiation over the life of the plant. The widely used definition of toughness transition temperature is RT_{NDT} (Reference Temperature for Nil Ductility). Put simply, the lower the end of life RT_{NDT} value, the better.

190 An example of how one measure of fracture toughness (K_{Ic}) changes with temperature is shown in Figure 2. This curve would be expected to be a lower bound to actual test data. The temperature axis is indexed to RT_{NDT}.

191 To determine the end of life RT_{NDT}, Westinghouse follows the method set out in US NRC Regulatory Guide 1.99 revision 2 (Ref. 32). For end of life RT_{NDT} Ref. 32 requires:

$$RT_{NDT(EOL)} = RT_{NDT(SOL)} + \Delta RT_{NDT} + M$$

where

$RT_{NDT(EOL)}$ is end of life RT_{NDT} (also termed RT_{PTS})

$RT_{NDT(SOL)}$ is start of life RT_{NDT}

ΔRT_{NDT} is the mean predicted shift in RT_{NDT} through life

M is a Margin term added to obtain an overall upper bound estimate - this is twice the square root sum of squares combination of the standard deviation in $RT_{NDT(SOL)}$ and ΔRT_{NDT}. The standard deviation for ΔRT_{NDT} is 15.6^{o}C for weld and 9.4^{o}C for base metal, but no more than 0.5x the mean value of ΔRT_{NDT}.

192 Westinghouse specifies the start of life RT_{NDT} as -23.3^{o}C for “beltline forgings”, and -28.9^{o}C for “beltline welds”. I have confirmed that “beltline forgings” means the Upper Shell Course, Lower Shell Course and Transition Ring of Lower Head (TRIM Ref. 2009/289088) and “beltline welds” are Welds 2 and 3, see Figure 1.

193 The change or shift in RT_{NDT} with neutron irradiation over life is one of the most intensively studied topics in the field on nuclear power plant structural integrity. Over the years, several ‘dose-damage’ correlations have been established, generally associated with individual country nuclear power programmes, with some international comparisons. This is a complex area and I decided to obtain the advice of specialists. I established a technical support contract to consider dose-damage relationships and their applicability to a UK AP1000 RPV. The result of this contract work is set out in Ref. 19.

194 Ref. 19 notes that shifts in RT_{NDT} are based on dose-damage correlations and these depend on statistical analyses of databases of materials test results. Materials tests are done either using material samples irradiated in test reactors or using specimens from operating plant surveillance schemes. Historically, such databases and resulting correlations have been generated on a national basis. Ref. 19 makes the point that as these correlations are empirical, 'national' details of steel specifications or steel making and welding practice might be embedded in the correlations. Also some correlations are based on data from a mixture of high and low Copper content materials, whereas other correlations are based on data from mainly low Copper content materials.

195 Ref. 19 raises concerns about the chemical composition limits of the forgings above Weld 3 and below Weld 2. As noted above, I have more recently confirmed that the forgings above and below both Welds 2 and 3 have the same "beltline" limits on Copper and Phosphorous.

196 Ref. 19 notes that in a US context, the dose-damage (i.e. RT_{NDT} shift) correlation in US NRC Regulatory Guide 1.99 Revision 2 (Ref. 32) was derived from a statistical analysis of US surveillance data in the 1980s. Ref. 19 states this correlation is known to be outdated, as both mechanistic understanding of embrittlement and the amount of data within the surveillance database, have increased since its derivation, but a replacement correlation has not yet been agreed.

197 Ref. 19 cites 3 more recent US dose-damage correlations:

ASTM - the 'E-900-02' correlation (2002);

ORNL - the 'EONY' correlation (2007);

US NRC - the 'RM-9' correlation (2007).

It seems the last two correlations above are both under consideration for inclusion in Revision 3 to regulatory Guide 1.99 (Ref. 19).

198 Ref. 19 considers how the AP1000 design parameters compare with the range of parameters of the surveillance databases that are the basis for the various US correlations. Ref. 19 concludes that, overall one may use correlations derived from the US database to predict embrittlement in the AP1000, though with caution. One specific area of concern in Ref. 19 is the high fluence to areas of the mid-height of the Lower Shell Course (Figure 1), even if a low leakage core configuration was to be adopted.

199 Table 4.1 of Ref. 19 provides a comparison of RT_{NDT} shifts calculated using various correlations and making a range of assumptions about fluence (location, whether or not low leakage core), irradiation temperature and Copper content. It is unfortunate I could not provide clearer information at the time for this work regarding chemical composition of the base forgings above and below Welds 2 and 3 (Figure 1). The concerns raised in Ref. 19 concerning chemical composition for the forgings above Weld 3 and below Weld 2 (Figure 1) are valid on the basis of information available at the time, but as seen above, this is not an issue given the Material Purchase Specification information I have now seen.

200 In examining the information in Table 4.1 of Ref. 19, I have concentrated on results from the following: Regulatory Guide 1.99 Revision 2, EONY and RM-9. These are respectively, the current US correlation and the two correlations under consideration in the US. To simplify matters, I have concentrated on the temperature shift estimates assuming the RPV wall is at the 'cold leg' water temperature. Clearly the lower circumferential weld (Weld 2, Figure 1) bounds Weld 3, so I have concentrated on Weld 2 and the cylinder forging (Lower Shell Course, Figure 1) at core mid-height. Table 10 summarises the relevant information.

201 From Table 10 I note:

- Regulatory Guide 1.99 Revision 2 predicts higher shifts for the weld than for the forging.
- EONY and RM-9 both predict higher shifts for the forgings than for the weld.
- Regulatory Guide 1.99 Revision 2 predicts higher shifts for the weld than either EONY or RM-9.
- Regulatory Guide 1.99 Revision 2 predicts lower shifts for the forging than either EONY or RM-9.

The main concern from the above is that the more modern correlations predict higher shifts for the forging than the current Regulatory Guide 1.99 Revision 2. The absolute difference is about 20^{o}C, or around half the overall predicted shift across the three correlations.

202 The AP1000 RPV 'beltline' base metal and welds specifications are characterised by low Copper content. Historically the proportion of US plant surveillance data from 'low Copper' material has been relatively low. In other words, correlations based on US data could be dominated by relatively 'high Copper' material. Most operating French PWRs would be characterised as having relatively 'low Copper' material and Ref. 19 provides calculations of RT_{NDT} shift using the French FIM dose-damage correlation, which is a mean shift correlation like the three US correlations. Overall, the French FIM results in Ref. 19 Table 4.1 are similar to the two more recent US correlations, EONY and RM-9, and the FIM results show the same type of difference compared to the Regulatory Guide 1.99 Revision 2 results, as the EONY and RM-9 results.

203 Although the shifts are important, the most important matter is the end of life value of RT_{NDT} and this depends on the start of life value and any margin to be added (given all three US correlations are for mean shift in RT_{NDT}). Table 11 provides end of life RT_{NDT} values for Weld 2 and the cylinder forging (Lower Shell Course) without a low-leakage core configuration. Using the Regulatory Guide 1.99 Revision 2 shifts in Ref. 19, the Margin values in the Regulatory Guide and the start of life RT_{NDT} values in the AP1000 DCD Table 5.3-3, the end of life RT_{NDT} temperatures are 59^{o}C for Weld 2 and 26^{o}C for the forging. Actual start of life RT_{NDT} values could of course be lower than the specification and this would reduce the predicted end of life RT_{NDT} values.

204 I note the predicted end of life RT_{NDT} values in the AP1000 DCD (Table 5.3-3 footnote) are higher, being 64.4^{o}C for the weld and 34.4^{o}C for the forging. It is not obvious why the values in the DCD are higher; two reasons could be including in the Margin term M, the standard deviation of the initial measured RT_{NDT} (taken as zero in Table 11 as no data is available) and possibly slightly higher neutron fluence levels for the results in DCD Table 5.3-3.

205 With the more recent EONY or RM-9 correlations, the equivalent end of life RT_{NDT} values in Table 11 would be about 5 to 10^{o}C lower for Weld 2, but about 20^{o}C higher for the forging.

206 Table 10 indicates a low leakage core would result in modest reductions in the predicted shift in RT_{NDT} for both weld and forging. All three correlations show about a 10^{o}C reduction in shift for the weld. For the forging location the current Regulatory Guide 1.99 Revision 2 only shows a 3^{o}C reduction, but the two newer correlations show a reduction in shift for the forging of more than 10^{o}C.

207 Whatever the value of end of life RT_{NDT}, there is the question of what criterion to use to evaluate it? There are two obvious approaches to the question of a criterion:

1. the SAPs approach of ALARP and the concept of a Basic Safety Level (BSL) as an upper limit and a Basic Safety Objective (BSO) as a practical expression of modern expectations and beyond which further consideration by ND would not be

appropriate (SAPs Paragraph 573, Ref. 8), coupled with the ND Statement (Ref. 31);

2. the criteria used by the designer, that is the US NRC requirements.

208 The relevant US NRC criteria are summarised in Table 12. The Regulatory Guide 1.99 Revision 2 criterion is in terms of the temperature shift at a position a quarter the way through the wall from the inner surface of the vessel. The 10 CFR 50.61 criterion is in terms of the inner surface of the ferritic material of the vessel wall (i.e. at the "clad to base metal interface") and is in terms of a screening criterion for Pressurised Thermal Shock (PTS).

209 In terms of a criterion on the inner surface, the two sets of criteria in Table 12 are probably roughly equivalent, i.e. they allow the end of life RT_{NDT} to reach up to about 130°C (148°C for circumferential welds). It is noted the Regulatory Guide 1.99 Revision 2 criterion for new plants mentions limits on chemical composition of material, but not measures to reduce neutron fluence. 10 CFR 50.61 (which is essentially for operating plant) invokes flux reduction if the PTS screening criteria are predicted to be exceeded.

210 The AP1000 DCD end of life RT_{NDT} values are comfortably less than either sets of US NRC criteria, and the estimates made here on the basis of Ref. 19 calculations are slightly lower. The PTS screening criteria may be based on complex arguments and analysis, they are concerned with an unlikely emergency / fault condition. During normal full power operation, the metal of the RPV will be well on the upper shelf of toughness, even at end of life.

211 From some point during plant life, every startup and shutdown will take RPV metal temperatures into the transition region of toughness (see Figure 2), though pressure and thermal stress may be modest when metal temperatures are low (this is a matter for Pressure-Temperature Limit curves and is dealt with in RO-AP1000-29 later).

212 From a UK perspective, the question remains, has the extent of operation (including startup and shutdown) with metal temperatures in the transition range of toughness been reduced ALARP. My concern is not so much with the welds, but with the base metal forging in the region of peak neutron dose. The peak dose to the forging is toward the upper end of dose-damage correlations, where there is least data to fit the correlation. By comparison, the neutron dose to Weld 2 (Figure 1) without a low-leakage core configuration is just within the neutron fluence level where the extent of data begins to decline (Weld 2 EOL fluence 2.5×10^{19} n/cm^2 versus about 3×10^{19} n/cm^2 for the start of the decline in extent of data).

213 Set against the concerns about the data behind the dose-damage correlations, one would expect it rather less likely that a crack-like defect with a size to be of concern could exist in the base forging compared with the welds.

214 Reduction in fluence to the RPV from current estimates would require physical changes to the plant design, for example more shielding round the core or a larger water gap. The dominant effect of a larger water gap could be to change the energy spectrum of the neutron flux, so there are fewer neutrons with energy >1MeV; they become neutrons with low or thermal energies rather than being 'absorbed'. The latter way of reducing high energy neutron flux then raises the issue of how to assess the damage due to the whole neutron spectrum; among other things it would be more important to assess neutron dose on the basis of 'displacements per atom' (dpa).

215 The sorts of changes involving more shielding round the core or a larger water gap would probably require an increase in the RPV inner diameter and a consequent increase in outer diameter. These might be considered major changes and not ALARP at this stage of the design; though at an earlier stage in the design process, such changes might have been easier and therefore at that stage ALARP.

216 One option still available would be to adopt in-service a low-leakage core management scheme. Depending on the dose-damage correlation used, this would lead to a modest reduction in predicted shift in RT_{NDT} by end of plant life. More importantly though it would bring the end of life neutron fluence for the welds to the region of dose damage correlations where they are based on most data. It would tend to move the base material neutron fluence toward the region of the dose-damage correlation where it is based on most data, but it would still be above 3×10^{19} n/cm^2. This would improve confidence in predictions.

217 Although Westinghouse is well able to complete analyses on the basis of a low-leakage core configuration, this may be a topic which can only be fully resolved by engaging with the licensee, once one is identified.

218 Overall, I recommend ND seeks an ALARP review of practical options for meaningful reduction of neutron dose to the RPV. As a minimum this should consider the locations of peak neutron dose, which occur in the forging. On the face of it, there is the potential for adopting a 'low leakage core' fuel management arrangement in service. The effect on neutron dose to the RPV of adopting a 'low leakage core' fuel management arrangement in service can be determined by the designer, though the implementation would be for the licensee.

219 It is possible, even likely, that dose-damage correlations will be updated in the US over the next few years, for instance Regulatory Guide 1.99 Revision 2 will be changed to Revision 3. I would recommend future ND assessment relating to the effects of neutron irradiation to take account of whatever is considered the most appropriate information.

220 Westinghouse routinely uses displacement per atom (dpa) in its analyses, as well as neutrons/cm^2. If need be, they should then be able to assess neutron energy spectrum differences between RPV wall locations, the AP1000 surveillance specimen locations, and possibly surveillance specimens from other operating PWRs whose data is used for the dose-damage correlations.

221 Ref. 19 notes the AP1000 DCD and other documents do not explicitly consider thermal ageing and strain ageing. For the irradiated locations of the RPV, the surveillance specimens will intrinsically include any thermal ageing that occurs at inlet leg coolant temperatures. The Pressuriser will operate at higher temperature and if thermal ageing is a factor, it would be expected to be greater for the Pressuriser. Ref. 19 (Section 2.1) notes a UK review in 1987 suggested allowing a 30^{o}C shift for thermal ageing during 40 years operation at 324^{o}C. Ref. 19 states it would be useful to review the allowance made for such processes and also the case for including relevant samples in surveillance schemes (for thermal ageing of unirradiated parts, a surveillance scheme would most easily be done ex-vessel).

222 In the UK, an attempt has been made to monitor for strain ageing effects, by including pre-strained specimens in the in-vessel surveillance scheme. However, the level of pre-strain needs to be representative and if values recommended elsewhere are typical (e.g. RCC-M Code 2007 Edition Annex Z G), it could be difficult to distinguish strain ageing shifts from other causes of shift.

223 Ref. 19 expresses concern about the relevance of a dose-damage relationship which is based on historical data largely from operating plants in one country when for the future, vessel base forgings and welds for assembly might be done in a number of countries. This is a question of the extent to which material and manufacturing procedures constrain the final manufactured base forgings and then the welds that join them into complete assemblies. This is a topic which might be explored generically. However, as noted in Ref. 19 and mentioned here, Japanese and French dose-damage correlations yield results similar to the US correlations, especially if one limits comparison to the two most recent US correlations (EONY and RM-9). This suggests any country-to-country variation might be minor based on the most recent correlations. But in any event, the intent should

be to use correlations that are appropriate to the materials in question. Ultimately any notable discrepancies should be brought to light by the results of the in-service surveillance programme. As the surveillance programme leads conditions in the RPV materials, early warning of an issue should be possible. If there was a difference between prediction and surveillance data, this might have economic consequences, but not safety consequences.

5.10 RO-AP1000-23. Primary Circuit Vessel Nozzle to Safe End Welds

224 As usual with a PWR that uses stainless steel pipework, for the AP1000 the connection between the pipework and the ferritic pressure vessels is made by means of stainless steel 'safe ends' attached to the ends of the vessel nozzles. The safe ends are welded to the vessel nozzles in the fabrication shop; the welds between the safe end and the pipework are made at site.

225 This RO raised a number of questions regarding details relating to the pipework safe ends. It also raised a question regarding the dissimilar metal joint that joins each of the Reactor Coolant Pump casings (austenitic stainless steel) to the channel head of the Steam Generators (low alloy ferritic steel). Responses to the questions were provided in Ref. 26.

226 Westinghouse confirmed that the attachment of the safe ends to vessel nozzle ends for the Reactor Pressure Vessel, Steam Generators and Pressuriser use buttering. The weld metal for this buttering and the weld to join the safe end to the nozzle is one or more of the following ASME Section II Part C specifications:

- SFA-5.11 ENiCrFe-7 (Alloy 152 - note purchaser can specify Boron (0.005% max) and Zirconium (0.02% max) additions).
- SFA-5.14 ERNiCrFe-7 (Alloy 52).
- SFA-5.14 ERNiCrFe-7A (Alloy 54 - chemical specification as for Alloy 52, but with Cobalt (0.12%), Boron (0.005% max) and Zirconium (0.02% max).

227 SFA-5.11 is for welding electrodes for 'Shielded Metal Arc' welding, 'Manual Metal Arc' welding in UK terminology. SFA-5.14 is for bare welding electrodes and rods. SFA-5.14 weld consumables would be used for manual or automatic 'Tungsten Inert Gas' (TIG) welds in UK terminology ('Gas Tungsten Arc Welding' in US terminology). Alloy 52 and 152 are weld consumable 'equivalents' of the base high alloy material, Alloy 690.

228 SFA-5.14 ERNiCrFe-7A first appeared in the ASME Code in the 2006 Addenda to the 2004 Edition; that is somewhat later than the default 1998 Edition with Addenda to 2000.

229 For the connection of the Reactor Coolant Pump (RCP) casings to the Steam Generators, Westinghouse stated the RCP casings will be welded to the Steam Generators prior to shipment of the Steam Generators to the plant site. In other words, the joint between the RCP casing and the Steam generator channel heads will be a shop weld. The ends of the Steam Generator outlet nozzles will be buttered and then the RCP casings welded to the buttered end. The buttering and weld consumables will be as for pipework welds above. The RCP casing to Steam Generator joint does not use a safe end. Westinghouse noted that two full size mock-ups of this weld are required ahead of production welding. These mock-ups will also be used for training of operators for non-destructive examination.

230 In-service examination of the weld joining each RCP casing to a Steam Generator is via the Steam Generator channel head, the canned motor pump does not have to be removed for this examination.

231 Westinghouse provided an indication of the intended ultrasonic examination of the weld joining the RCP casings to the Steam Generator outlet nozzles. This was a useful

extension of a response to a GDA Step 2 issue relating to developing a qualification programme for examination of the RCP casing to Steam Generator outlet nozzles. The drawings provided for detection and sizing show the narrow-gap nature of the weld.

232 The information provided in Westinghouse's responses is sufficient for my GDA Step 3 assessment. In GDA Step 4 the weld procedures might be sampled, and the results of the mock-up welds for the reactor Coolant Pump casing to Steam Generator outlet nozzle assessed.

5.11 RO-AP1000-24. Information on Reactor Internals

233 The AP1000 DCD (Ref. 6) in Sub-Chapter 3.9 describes the Lower Reactor Internals and the Upper Core Support Structure. The reactor vessel internals in the AP1000 are similar in size and overall configurations to previous Westinghouse designed three loop PWRs. The major materials of construction of the reactor internals is 300 series austenitic stainless steel. The Core Barrel of the Lower Reactor Internals (which supports the weight of the core as well as providing part of the downcomer flow path of reactor coolant) is made of three cylindrical sections, the sections being joined by circumferential welds. Each cylindrical section has one longitudinal weld. The Lower Core Support Plate is joined to the lower end of the Core Barrel by a circumferential weld. The neutron shield pads and the surveillance capsule baskets are fixed to the Core Barrel by bolts and dowel pins. Some items are also bolted to the Lower Core Support Plate. The Core Shroud is made from plates welded together.

234 The original reference plant for the Westinghouse three loop PWR reactor internals so far as flow-induced vibration is concerned is the H. B. Robinson plant.

235 Since the reference plant, several successive design changes have been incorporated, all of which have been tested on subsequent operational plants. For example the neutron shield panels (see Section 5.9 above) were tested on the Trojan 1 plant and the core shroud within the core barrel (see later) was tested on the Yonggwang 4 plant.

236 The AP1000 reactor internals design also includes features which have not been tested on operational plants. For instance the core barrel is about 280mm longer than standard for a three loop extended fuel (length) design, and a flow skirt is included in the reactor vessel lower head.

237 In response to questions, Westinghouse provided a number of responses (Ref. 26); these are summarised below.

238 Clearly the Lower reactor Internals, especially the Core Barrel which carries the weight of the core will be subjected to significant neutron irradiation over plant life. Westinghouse notes that the estimated peak neutron fluence to the AP1000 core shroud is dominated by the power density in the peripheral fuel assemblies of the core. The AP1000 peak neutron fluence is expected to be similar to that in currently operating plants.

239 Westinghouse states that susceptibility to irradiation-assisted stress corrosion cracking and void swelling in reactor internals have been identified by the nuclear industry as potential degradation mechanisms in current PWR plants. These degradation mechanisms are being addressed for currently operating PWR plants through reactor material reliability programmes. Westinghouse states it is expected the AP1000 will be affected by similar ageing effects. Westinghouse states that for the AP1000, ageing management provisions include inspectability and reliability requirements to address these ageing mechanisms. In the limit, so long as there is forewarning of degradation the reactor internals could in general be replaced; they are removed from the RPV at major re-fuelling outages.

240 Approximate peak estimated neutron fluence values to the core shroud and the core structural components are:

Core Shroud: 1.28×10^{23} n/cm^2 (E>1MeV), 2.74×10^{23} n/cm^2 (E>0.1MeV), or 185dpa (this only in a small volume, a representative average might be 5×10^{22} n/cm^2 or 65dpa);

Lower Core Support Plate: 2.41×10^{21} n/cm^2 (E>1MeV) ~3.5dpa;

Support Plate Flange: ~1×10^{20} (E>1MeV) <0.2dpa;

Core Barrel: 1.5×10^{22} n/cm^2 (E>1MeV) about 25dpa;

Top of Core Barrel (ledge of RPV) < 1×10^{17} n/cm^2 (E>1MeV) <<0.01dpa.

The core shroud is not considered a core support structure.

241 Westinghouse notes that Irradiation Assisted Stress Corrosion Cracking (IASCC) requires both fluence and sustained stress; void swelling is highly temperature sensitive and of course depends on neutron fluence level. Based on analysis and PWR operating plant experience to date, Westinghouse contends that the conditions for IASCC and void swelling do not exist in the AP1000 core support structures. This contention is supported by a technical report.

242 Westinghouse confirms there is no fracture toughness requirement for the stainless steel core support structures at start of life. In general, annealed and moderately cold worked austenitic stainless steels show a large resistance to ductile tearing. Irradiated stainless steels tend to show sharp decreases in ductility and fracture toughness when subjected to neutron irradiation between 0.1 and 10dpa. There is variation in irradiation sensitivity and prediction is difficult. Therefore lower bound toughness curves are commonly used in analysis of highly irradiated components. Above 10dpa, a characteristic lower bound toughness might be about 45 MPa√m.

243 Overall, I conclude the AP1000 reactor internals, and particularly those elements that support the core are similar to existing PWR plants. The degradation mechanisms are well-known and can be managed. The overall claims on structural integrity of the reactor internals that support the weight of the core are a matter for other Regulatory Observations (RO-AP1000-18 and RO-AP1000-19).

5.12 RO-AP1000-25. ASME Design Specifications and Design Reports - Current Status and As-Built Status

244 This Regulatory Observation in general was concerned with ND access to Design Reports and Design Specifications. There was a useful discussion of the nature of various documents. To move the subject forward, I identified a sample of documents related to the RPV (TRIM Ref. 2009/289088). The purpose of this sample was to understand the scope of coverage in each document, rather than a detailed analysis of the content of each document.

245 I now have a better understanding of Westinghouse's document structure relating to pressure boundary components. This review of scope of documents for the RPV should be applicable to the same range of documents that a produced for other major component such as the Pressuriser and Steam Generators. This is something to take forward in GDA Step 4.

246 Some of the content of these documents has already been useful in clarifying a number of points, and has contributed to other parts of this assessment report.

247 A specific question raised concerned the reconciliation of design reports for pipework with the as-built condition. I was interested to understand how significant such reconciliation might be. Westinghouse explained that this reconciliation is based on geometric check of the installed pipework. If the pipework has been placed in the as-analysed location, within a tolerance, the reconciliation exercise should be a minor matter. The installed

configuration of the pipework will be determined by a 'plant walkdown' process and that will take some time. However, there should be little need to actually re-analyse pipework.

248 This RO did not set out to 'solve' any particular technical issue. It started with a request for some documents, which eventually led to a better understanding of Westinghouse's document structure, and hopefully a better understanding of what documents to select as a sample for assessment in future.

5.13 RO-AP1000-26. Fatigue Crack Initiation - Conservatism in ASME III Appendix I S-N Curves for Stainless Steel Material

249 This Regulatory Observation is concerned with design fatigue analyses that use so-called S-N curves. S-N curves are Stress - Number of Cycles curves usually provided in design codes. The ASME code contains such design curves; the design curves are supposed to have a factor of 2 conservatism on stress and a factor of 20 conservatism on number of cycles; these are compared to the experimental data on which the curves are based. These factors of conservatism reflect the sort of variability found in tests for fatigue endurance. The analysis uses predicted load cycles to check for the potential for fatigue crack initiation in components. That is the analysis assumes an initially defect-free component and determines the number of cycles required to start a defect, usually at the surface at stress concentrations.

250 In practice several different loading cycles will affect individual locations of components, and a method is used to sum the contribution of each type of loading cycle. The usual method of summation in an ASME design context is linear summation of load cycle ratios, where the ratios are number of cycles applied compared to the number of cycles of that type that would be required for reaching the S-N curve limit. This is known as Miner's summation. This S-N based fatigue analysis procedure has been used for many years.

251 Ref. 36 reviews the margins in the ASME Code fatigue design curves. Ref. 36 reviews carbon and low alloy steels and austenitic stainless steels. The overall conclusion in Ref. 36 is:

> "*The results indicate that the current ASME Code requirements of a factor of 2 on stress and 20 on cycles are quite reasonable, but do not contain excess conservatism that can be assumed to account for the effects of LWR environments*."

252 However, Table 9 of Ref. 1 also shows that for stainless steel in PWR primary water environment, the required factor on cycles can range from 19 to 31. The factor on strain or stress for all forms of material considered was 1.6 to 1.7.

253 One interpretation of the findings in Ref. 36 is that for stainless steel component locations with a high predicted usage factor, the actual level of conservatism could be lower than the expected factors of 2 on stress and 20 on cycles.

254 As a starting point to determine the significance of the above interpretation, I asked how many stainless steel component locations have a usage factor exceeding 0.75.

255 Westinghouse responded with a short list of reactor internals locations (three locations) where the usage factor was predicted to exceed 0.75. I do not regard these locations as critical, and as noted in the response by Westinghouse, the fatigue analysis is on the basis of 'umbrella transients' and this introduces a separate element of margin.

256 Westinghouse had indicated that fatigue usage factors for the Pressuriser surge line would be available for checking against the 0.75 level. However, the detailed analyses for the surge line are apparently still in hand. Assessment of fatigue usage factors for the surge line will have to be dealt with in GDA Step 4; I do not see this as a critical matter.

5.14 RO-AP1000-27 and RO-AP1000-28. Reactor Internals - Testing and Inspection Programme of First AP1000 as "Prototype" & Pressuriser Surge Line Stratification Evaluation -AP-1000 First Plant Only Test

257 These two Regulatory Observations are taken together because the fundamental point is they both envisage tests on the first AP1000 being then applicable to all future AP1000s.

258 For the reactor internals, the AP1000 DCD (Ref. 6) in Chapter 3.9, Section 3.9.2.3 states:

> "*The vibration assessment program for the AP1000 reactor internals determines, prior to testing the first AP1000, that the internals are not expected to be subjected to unacceptable flow-induced vibrations….*
>
> *The AP1000 core barrel and core shroud will be instrumented during pre-operational testing of the first plant to determine the shell mode and beam mode frequencies and amplitudes*."

259 The AP1000 DCD (Ref. 6) in Section 3.9.2.4 gives some more detail regarding the testing of the first AP1000 i.e:

> "*The pre-operational vibration test program for the reactor internals of the AP1000 conducted on the first AP1000 is consistent with the guidelines of Regulatory Guide 1.20 for a comprehensive vibration assessment program…..*
>
> *With respect to the reactor internals preoperational test program, the first AP1000 plant reactor vessel internals are classified as prototype as defined in Regulatory Guide 1.20. The AP1000 reactor vessel internals do not represent a first-of-a-kind or unique design based on the arrangement, design, size or operating conditions…*
>
> *The pre-operational test program of the first AP1000 plant includes a limited vibration measurement program and a pre- and post-hot functional inspection program. This program satisfies the guidelines for a Regulatory Guide 1.20 Prototype Category plant…*
>
> *Since the most notable differences with previously tested designs are in the lower internals, instrumentation is concentrated on the lower internals*…."

260 I note US NRC Regulatory Guide 1.20 Revision 3 (Ref. 37), defines 'prototype' as:

> "*A 'prototype' is a configuration of reactor internals that, because of its arrangement, design, size or operating conditions, represents a first-of-a-kind or unique design for which no 'valid prototype' exists*."

It appears that although Westinghouse does not consider the AP1000 reactor internals as meeting the US NRC definition of 'prototype', they will nevertheless be treated as 'prototype'.

261 Westinghouse provided a useful clarification of the classification of AP1000 plant reactor internals beyond the first plant. The planned designation of the reactor internals after the first plant is 'Non-Prototype Category I' as defined in Regulatory Guide 1.20.

262 Westinghouse explained that the pre-operational vibration testing on the first AP1000 will be performed without fuel loaded and at a uniform no-load reactor coolant system temperature of about 290^{o}C. Westinghouse has found that testing without fuel assemblies in place provides bounding vibration responses compared to testing with fuel assemblies installed. It is expected that debris screens will be installed in the reactor internals and incidentally these will simulate about 1/3 of the expected core pressure drop.

263 I raised a question about the pre-operational test results on the first AP1000 reactor internals being available as supporting evidence in subsequent AP1000 plant Station Safety Reports (also termed Operational Safety Reports). Westinghouse indicated it is closely involved in performing and evaluating the tests and in the past similar test reports have been made available to subsequent plant owners. This is a useful response. Of

course, if for some reason the pre-operational test results on the first AP1000 reactor internals were not made available to subsequent plants, a solution would be to carry out the same test on a subsequent plant.

264 For the Pressuriser surge line, the AP1000 DCD (Ref. 6) in Chapter 3.9, Section 3.9.3.1.2 states:

> "*A monitoring program will be implemented as discussed in subsection 3.9.8.5 at the first AP1000 to record temperature distributions and thermal displacements of the surge line piping, as well as pertinent plant parameters such as pressuriser temperature and level, hot leg temperature and reactor coolant pump status. Monitoring will be performed during hot functional testing and during the first fuel cycle. The resulting monitoring data will be evaluated to show that it is within the bounds of the analytical temperature distributions and displacements*."

265 And AP1000 DCD (Ref. 6) Chapter 14.2 Section 14.2.9.1.7 (on Page 14.2-30) states:

> "…*temperature sensors are installed on the pressuriser surge line and pressuriser spray line for monitoring thermal stratification and thermal cycling during power operation. Testing is performed to verify proper operation of these sensors. Note that this verification is required only for the first plant*."

266 In response to questions, Westinghouse confirmed that the surge line monitoring on the first AP1000 is planned to continue through to the end of the first fuel cycle, with interim assessments recommended during heatup and after three months of normal operation.

267 I have seen a technical report from Westinghouse that provides guidelines for the pressuriser surge line monitoring, including the number and location of monitoring points. The guidelines appear to be based on appropriate analysis and are comprehensive.

268 The results and conclusions of the monitoring of the pressuriser surge line on the first AP1000 could be relevant information for the Station Safety Reports of subsequent AP1000 plants. Westinghouse has indicated the results will be available for its design review purposes (e.g. to adjust surge line transient, stress and fatigue analysis models). Availability of information to subsequent plant operators would require appropriate commercial arrangements.

269 The AP1000 surge line design takes account of previous operating experience and issues arising on operating plant. Nevertheless, it would be useful to know that the design has effectively eliminated known issues on surge lines. For a UK AP1000 this re-assurance could be provided by results and conclusions from the first AP1000. However, if commercial arrangements are not possible, it would always be practically possible to conduct the same monitoring on a UK AP1000. This is a topic for interaction with a UK AP1000 licensee.

5.15 RO-AP1000-29. Reactor Pressure Vessel and Primary Circuit Pressure - Temperature Limits and Low Temperature Overpressure Protection

270 AP1000 DCD (Ref. 6) in Chapter 5.3, Section 5.3.3 and 5.3.6.1 deals with Pressure - Temperature (P-T) limits for the RPV. Although P-T limits are determined for the RPV, they are claimed to be applicable to the rest of the reactor coolant system. That the RPV is limiting is reasonable on the basis that P-T limits are mainly relevant to the ferritic steel vessels of the reactor coolant system, the start of life fracture toughness properties are likely to be similar across the main ferritic steel vessels, and the largest change in toughness properties over life is likely to be due to neutron radiation to parts of the body of the RPV.

271 The P-T limit curves are derived on the basis of:

The methodology in ASME XI Appendix G;

The methodology in US NRC Regulatory Guide 1.99 Revision 2 for material degradation due to the effects of neutron irradiation (see section 5.9 for RO-AP1000-22).

272 Regions of the RPV considered in setting P-T limits include the cylindrical region adjacent to the core (Lower Shell Course and Welds 2 and 3 in Figure 1).

273 Regarding the Lower Shell Course and Welds 2 and 3, the AP1000 DCD in section 5.3.3.1 states:

- The fluence values used are calculated values, not best-estimate values.
- The K_{Ic} critical stress intensities are used in place of the K_{Ia} critical stress intensities.

274 The 1996 edition of ASME XI Appendix G is used rather than the AP1000 default of the 1989 edition.

275 Table 13 here summarises the main features of the methodology for determining allowable pressure as set out in ASME XI Appendix G.

276 The overall methodology used to determine P-T limits for the RPV are long-standing. For example much the same methodology is contained in ASME III Appendix G in the 1983 edition of the Code. The main change over the years has been the change in criterion from K_{Ia} to K_{Ic}.

277 The difference between K_{Ic} and K_{Ia} versus temperature (referenced to the RT_{NDT} temperature) is shown in Figure 3 here. The change from use of K_{Ia} to K_{Ic} was introduced in the ASME Code firstly by Code Case N-640 (Ref. 38), then again in Code Case N-641 and was incorporated in the main code, ASME XI Appendix G in the 1998 Addenda. Background to this change is contained in Refs 39 and 40.

278 The ASME XI Appendix G methodology might be characterised as a nominal, deterministic procedure to determine P-T limit curves. It is nominal in the sense that some aspects, e.g. choice of defect depth and aspect ratio are not based on any obvious criterion, other than it defines a large defect that is unlikely to occur in practice. The procedure has a number of explicit and implicit margins. It might also contain some negative margins, for example only the deepest point of the postulated defect on the inside surface is considered, along with the corresponding neutron fluence at that point. For the inside surface crack, the crack tip at the surface (along with the higher neutron fluence at the inner surface) is not explicitly considered.

279 I note that the P-T limit curves for the now closed UK Magnox steel RPV stations were based on a combination of (Ref. 46):

- reference, postulated surface defect 1/4 or 1/3 wall thickness (25mm deep in either 100 or 75mm thick wall) with extended geometry (rather than 6:1 aspect ratio);
- material fracture toughness based on lower bound K_{Ic};
- irradiated material properties (strength, toughness) determined at the inside surface of the vessel wall (not mentioned in Ref. 47, but known to me through past assessment work);
- pressure reserve factor of 1.2 (compared with 2 for ASME or RCC-M, but note difference in reference, postulated defect).

So there is precedent for use in the UK of a lower bound to K_{Ic} as the measure of fracture toughness in determining P-T limits for ferritic steel Reactor Pressure Vessels.

280 In response to questions on the current methodology for determining P-T limit curves, Westinghouse noted the following:

1. The analysis could use:

(i) K_{Ia} rather than K_{Ic}; and / or

(ii) toughness at the inner surface of the RPV but applied to a crack tip at the ¼ wall depth location;

but this would be contrary to US practice (obviously). It would result in more restrictive P-T limits that could reduce overall plant safety.

2. For pressure loading, the highest applied K_I is at the deepest point of a 1/4T thickness postulated crack and for P-T curves, pressure is the dominant loading. Apparently this also applies to locations such as the vessel flange.

3. An example of the difficulty with more restrictive P-T limits is if a Low Temperature Overpressure Protection (LTOP) relief valve opened and stuck open. This would result in a small loss of coolant accident. This would be a more severe transient to the Reactor Coolant System than a normal heatup and cooldown. The claim is this could result overall in lower plant safety.

4. Axial flaws are postulated for axial welds, plates and forgings; circumferential flaws are postulated for circumferential welds.

281 For the AP1000, the use of a postulated axial flaw at the location of highest neutron fluence in the forging is a notable conservative factor. Based on technical reports I have seen (TRIM Ref. 2009/289088) the combination of axial postulated flaw located in the forging and the forging end of life RT_{NDT} is the limiting combination (i.e. compared with a circumferential postulated flaw located in Weld 2, even though the predicted end of life RT_{NDT} is higher for the weld compared to the forging).

282 There are a number of factors relating directly to the basis for determining P-T limit curves and the relative conservatism of different options. I agree that the consequences on overall plant safety should be considered. I think this is an area for further ND consideration in GDA Step 4, particularly focussed on what is ALARP in determining the P-T limit curve. In practice, concerns about the P-T curves imposing undesirable restrictions on other aspects of plant operation, might be associated with the vessel flange limits rather than the irradiated region of the vessel.

5.16 RO-AP1000-30. Containment Pressure Shell ASME SA738 Grade B

283 The AP1000 DCD (Ref. 6) in Chapter 3.8, Section 3.8.2 describes the steel containment. The containment pressure shell is a steel membrane. The containment shell comprises a cylindrical section (axis vertical) with ellipsoidal closure domes at the top and bottom. The bottom closure dome is embedded in the concrete structure of the shield building. According to Ref. 2, Chapter 3, Page 3-106, the cylindrical section has a diameter of 39.6m and length of about 42.6m, the overall height of the steel shell is 65.6m (crest to crest of the domes). The design pressure and temperature are 0.407MPa (gauge) and 149^{o}C. Following a design basis accident, the long term pressure and temperature in the containment are about 0.16MPA gauge and 110^{o}C. To put these design and 'operating' values in some perspective, a domestic pressure cooker on 'high' pressure is designed to operate at about 0.1MPas gauge and 120^{o}C. The steel containment shell is designed and constructed to ASME III Subsection NE (Class MC Components). The negative design pressure differential (i.e. higher pressure on the outside) is 0.02MPa.

284 The material of construction of the AP1000 steel containment shell is ASME SA-738 Grade B. The wall thickness of most of the containment shell is 44.4mm with the lowest cylindrical course 47.6mm. The axial welds of this lowest cylindrical course are post-weld heat treated, all other seam welds are left in the as-welded condition, as allowed by the ASME code.

285 The containment pressure shell is clearly a large structure. Steel containment shells (cylinders or spheres) have been used in nuclear power plants previously, though not of this overall size in the UK. The Dounreay Fast Reactor (DFR) sphere has a diameter of about 41m (completed May 1957), which makes it comparable to the cylindrical section of the AP1000 containment shell. The DFR shell is made of steel plate about 25mm thick. Most Magnox steel Reactor Pressure Vessels are spheres with a diameter of about 20m; their design pressure means the wall thickness is between 75 and 100mm. The Magnox steel RPVs were welded together on site and the welds subjected to post-weld heat treatment on site.

286 In assessing the structural integrity of the containment shell, it is noted that the AP1000 shell is unlikely ever to see its design loading. In practice it will only be subjected to test loads on a few occasions through plant life.

287 On the basis of the latest ASME III Subsection NE stress limits, the US NRC FSER (Ref. 2, Page 3-106) states the main part of the steel containment has a margin on thickness of 0.6mm available as a corrosion allowance. The lower part of the cylindrical region enclosed in concrete has a thickness margin of about 3mm.

288 Overall, my attention was drawn to the material of construction for the pressure containment; SA-738 Grade B. A material specification for ASME SA-738 has existed in the ASME code since about 1980. The specification has evolved over the years, but fundamentally a material like the current SA-738 Grade B could have been procured at any time since 1980.

289 Up to 2002, SA-738B was only permitted in the ASME code for ASME Section VIII Division 1 vessels (non-nuclear). An important change occurred in 2002 when the ASME code was amended to allow SA-738 Grade B to be used in ASME III vessels (nuclear); prior to the 2002 Addenda, only Grades A and C were allowed (Grade B has a higher tensile strength specification than Grades A and C, there are variations in chemical compositions between the grades). Another relevant change in 2002 was the change in maximum allowable stress limits. The Maximum Allowable Stress defined for SA-738 Grade B was altered (ASME II Part D Subpart 1, Table 1A). The limit of 24.3ksi (167MPa) which had applied up to 200°F (93°C) was extended to 500°F (260°C). Recall the design temperature of the AP1000 containment pressure shell is 300°F (149°C). As with other specified material in ASME III Subsection NE, the Allowable Stress Intensities and Stress Values for Class MC components are the Class 2 and 3 Allowable Stress Values multiplied by 1.1 (NE-3112.4), but no greater than 90% of the material's yield strength at temperature.

290 For the AP1000 containment pressure shell, the original ASME code edition chosen was the 1998 Edition with 2000 Addenda. Subsequently this was changed to the 2001 Edition with 2002 Addenda. This allowed SA-738 Grade B to be used and with the higher maximum Design Stress.

291 The main examination requirement for the butt welds of the containment shell is full radiography. Given the plate thicknesses are less than 50mm, it appears (ASME III NE-5130) that examination of weld edge preparations is not required.

292 Another aspect of ASME III Subsection NE for SA-738 is the heat treatment requirement. ASME III Subsection NE is specific in the pressure retaining materials that may be used for Class MC Components (NE-2120). SA-738 is included in Table NE-2121(a)-1. Post-weld heat treatment is required by default (NE-4622.1(a)) but there are exemptions as set out in NE-4622.7 and Table NE-4622.7(b)-1.

293 ASME uses a system of classifying materials that is used, among other things, to determine exceptions to post weld heat treatment; SA-738 is ASME P-Number 1 and Group 3. In ASME III Subsection NE, Table NE-4622.7(b)-1 most materials with P-Number 1 are exempted from post weld heat treatment if the nominal thickness is less

than 38mm. However, for four specific materials, the limit on thickness for exemption is raised to 44mm; the materials are SA-299, SA-516, SA-537 and SA-738. For the exemption to apply for these four materials, the maximum reported Carbon content must be less than 0.24% and the exemption from impact testing in NE-2311(a)(8) is not permitted (full details in Note 2 to Table NE-4622.7(b)-1).

294 Exemptions from post weld heat treatment in ASME III Subsection NE, table NE-4622.7(b)-1 are in terms of 'nominal thickness'. ASME III NE, NE-4622.3 defines 'nominal thickness' as:

> "*the thickness of the weld, the pressure retaining material for structural attachment welds or the thinner of the pressure retaining materials to be joined, whichever is least. It is not intended that nominal thickness include material provided for forming allowance, thinning, or mill overrun when the excess material des not exceed 1/8 in (3mm)*…"

Tolerance on specified plate thickness is covered below in the summary of SA-20.

295 As noted earlier, SA-738 has been a permitted material in ASME VIII (non-nuclear vessels) for a number of years. In ASME VIII, the exemptions from post-weld heat treatment requirements (Table UCS-56 in ASME VIII Division 1 and Table 6.8 in ASME VIII Division 2), for P-Number 1 Group 3 materials is for welded joints less than 38mm. It is noted that in British Standard Published Document PD5500:2006 (Ref. 41), a Group 1.3 material (the relevant Group for SA-738B in PD5500, Table 2.1-1 of Ref. 41) would not normally require post-weld heat treatment for nominal thickness less than 35mm (Table 4.4-1, Ref. 41) where vessels are designed to operate above 0^oC (4.4.3.1, Ref. 41). For ferritic steel vessels designed to operate below 0^oC, PD5500 post weld heat treatment requirements are contained in its Annex D.

296 Westinghouse explained (TRIM Ref. 2009/289088) the material for the AP1000 containment pressure shell was the subject of a detailed study performed for Westinghouse by an experienced contractor. The selection of SA-738 Grade B was a direct result of the study. Based on the outcome of this study, the interested organisations initiated ASME code committee action to get SA-738 Grade B included as a containment pressure boundary material, initially as a Code Case (N-655, Ref. 42) and subsequently incorporated into the ASME Code. I have produced a note giving a summary of the history of SA-738 in the ASME code (Ref. 43). The updated version of N-655, N-655-1 (21 September 2007) makes SA-20 Supplementary Requirements S1 and S20 mandatory for this material (S1: vacuum treatment, S20: maximum carbon equivalent for weldability).

297 The SA-738 specification states that the material shall also conform to the requirements of specification SA-20, "Specification for General Requirements for Steel Plates for Pressure Vessels". Particular requirements in SA-20 of interest here relate to types and numbers of mechanical tests and the thickness tolerance of plates.

298 For quenched and tempered plates SA-20 states two tension test coupons are required for each plate, taken from opposite ends of the plate. One Charpy impact test (i.e. 3 specimens) is required for each quenched and tempered plate as heat-treated. The impact test coupons are taken adjacent to the tension test coupons.

299 The permissible variations in plate thickness (metric) are defined in Table A2.1 of SA-20. The permissible excess thickness varies with specified thickness and plate width. For plate with a specified thickness of 45mm, the permissible excess thickness can be up to about 2 to 3mm. A footnote to Table A2.1 of SA-20 states the permissible variation under specified thickness is 0.3mm.

300 In response to questions, Westinghouse noted that specific weld consumables are not specified, but the testing and mechanical properties requirements are specified. The design specification limits welding to:

automatic welding to gas shielded Flux Cored Arc Welding (FCAW) and Submerged Arc Welding (SAW);

manual welding Shielded metal Arc Welding (SMAW), i.e. Manual Metal Arc (MMA) welding in UK terminology.

301 Mechanical strength and fracture properties of the plate material and the welds are as required by the ASME code. Weld tests are done for both the as-welded and post weld heat treated conditions.

302 The lowest service metal temperature (-26.1^{o}C) is based on assuming an ambient temperature of -40^{o}C and normal plant power operation. However, in analysing the design basis accident, the initial internal containment temperature was assumed to be at the Technical Specification limit of 49^{o}C (120^{o}F) and the external temperature was assumed to be 46^{o}C (115^{o}F).

303 Between 1996 and 2004, a number (more than 10) of quite large storage vessels have been built for the petro-chemical industry around the world, using SA-738 Grade B. It is noted that the majority of these storage tanks received a post weld heat treatment.

304 As noted at the beginning of this section, the containment shell is unlikely ever to see its design loading. In practice it will only be subjected to test loads on a few occasions through plant life. The pressure rise and fall in the design basis accident is quasi-static, taking about 1 hour to reach peak pressure and then falling to about half the peak pressure after a total elapsed time of about 3 hours.

305 In assessing the structural integrity of the containment pressure shell, it is important to keep in mind the design basis of the vessel. With high likelihood it will never be loaded by the main design basis loading, in normal plant operation it is essentially at zero load. This means although the steel pressure shell is a pressure vessel, it is not subject to constant load as is the case for most pressure vessels. The material of construction has been chosen as optimum for this specific application, and although the post weld heat treatment requirements are more relaxed for the material of construction compared with similar P-Number materials, there are countervailing requirements.

306 The design of the steel containment shell complies with the ASME code. There are some general aspects that need further consideration:

- plate thickness available for a corrosion allowance for most of the shell;
- the toughness properties of plates and welds to meet the requirements for no post weld heat treatment;
- the tolerance on plate thickness (over and under thickness), relevant to corrosion allowance and no post weld heat treatment.

One aspect of welding is the attachment of the crane girder which is welded to the inside surface of the shell to provide support for the polar crane rail and in turn the polar crane itself (potential for lamellar tearing of the shell plate). Matters to take forward for further assessment include:

- the number of Charpy tests required for each plate;
- the weld procedures and their qualification;
- the manufacturing examination requirements for weld edge preparations and significant attachment welds;
- the specifications for the procurement of the plate material, especially the thickness tolerance requirements;
- any recommendations for in-service monitoring and maintenance of the protective coating on the surfaces of the steel shell (inorganic zinc paint).

5.17 Steam Generator Tubing

307 The AP1000 DCD states that the Steam Generator tubing will be made using Alloy 690 in the Thermally Treated (TT) condition (Ref. 6, Sub-Chapter 5.4 Sections 5.4.2.2 and 5.4.2.4.3). This material was used for the Sizewell B Steam Generators and has been widely used around the world since for PWR Steam Generators, mostly replacement Steam Generators.

308 Based on my knowledge of UK experience of Thermally Treated Alloy 690 Steam Generator tubing and a general perception of international experience of this material, I had no particular concerns about its use. But, given the past interest in the UK of this aspect of PWR structural integrity (Refs 23 and 24), I judged it prudent to give the matter some consideration. I decided to do this through a support contract to review PWR Steam Generator tube materials and manufacturing routes.

309 The review (Ref. 44), focuses primarily on mill annealed Alloy 690 TT but includes some comparison with Alloy 600 and Alloy 800. Using open literature sources, the review deals with:

- material selection (which mainly affects resistance to stress corrosion cracking);
- manufacturing routes for tubing (emphasising the details of manufacturing that can also influence resistance to in-service degradation);
- factors which can affect in-service degradation that are not inherently due to material selection or manufacture;
- available information on in-service performance.

310 Historically, PWR Steam Generator tubing has often used mill annealed Alloy 600, Alloy 600 in the Thermally Treated condition, or Alloy 800, Nuclear Grade (NG). Alloy 800 has notably been used in the German Konvoi PWR series and the Canadian CANDU Steam Generators.

311 Sections of Ref. 44 deal with:

- Tube Specifications and Manufacturing Methods.
- Water Chemistry, General Corrosion, Cation Release and Fouling.
- Stress Corrosion Cracking, Fatigue and Wear.

Appropriately, the section on Stress Corrosion Cracking, Fatigue and Wear is the longest by some margin.

312 The main conclusions in Ref. 44 follow.

313 The main manufacturing routes of interest for Alloy 690 TT PWR steam generator tubing, pilgering and cold drawing, produce today high quality tubes with no significant differences in corrosion behaviour. Moreover, some initial problems with eddy current inspection of cold pilgered tubes were solved years ago.

314 Control of microstructure and of surface condition are key factors affecting in-service performance in terms of cation release and primary circuit activation as well as maximizing resistance to stress corrosion cracking from either the primary or secondary sides.

315 Mill annealing and thermal treatment conditions (temperature and atmosphere) in combination with the alloy carbon content have been optimized to ensure good intergranular carbide morphology, minimize intragranular carbide precipitation and ensure maximum resistance to stress corrosion cracking. High mill anneal temperatures (often

~1080° C) followed by a thermal treatment of 5 to 15 h at 700-738° C are now universally used to obtain these microstructures.

316 Cation release is strongly influenced by surface condition and by the water chemistry during the build up of protective oxide layers on new tube surfaces. Attention has to be paid to cleaning processes all along the manufacturing route in order to avoid ID surface carburization or nitriding by incompletely removed cleaning solutions. Grit blasting of the ID is avoided as it results in rougher and cold worked surfaces.

317 In addition, in order not to impair resistance to stress corrosion cracking and to reduce cation release to a minimum, contact with deleterious chemical products has to be avoided (unless the surface can be efficiently cleaned), throughout the tube manufacturing and assembly processes.

318 Water chemistry is also an important factor regarding cation release. Optimization of hot functional tests may contribute to decreasing the activity of the primary circuit both by building up a 'good' protective layer in water chemistry conditions and at redox potentials similar to those that prevail during normal operation, and by efficiently eliminating most of the corrosion products formed before criticality so that no activation of them occurs. Attention should also be paid in the future to the primary water hydrogen concentration whose effect is not well understood; further studies are known to be underway. Finally, zinc injection from the very beginning of operation may be an effective way to decrease cation release.

319 The considerable literature on resistance to stress corrosion cracking of Alloy 690 TT in PWR primary water (PWSCC) shows that no cracking has been observed in long term, constant load or constant deformation laboratory tests in heats with good intergranular carbide morphology and few intragranular carbides. Some cracking has been observed in a few cases during constant deformation or slow strain rate tests of Alloy 690 with anomalous microstructures (i.e. non-optimized carbide distribution and morphology and perturbed, contaminated surface layers). Thus, the high chromium content of Alloy 690 cannot be relied on alone to confer complete PWSCC resistance. An intergranular network of fine carbides, which is an important objective of the manufacturing sequence for Alloy 690 TT, is essential to ensure optimum resistance to PWSCC. Nuclear grade Alloy 800 is equally resistant.

320 A recent quantitative assessment of PWSCC resistance in Alloy 690 TT relative to Alloy 600 in the mill annealed condition shows an average factor of improvement of between 40 and 100. These are minimum values dictated by the maximum testing periods used without any PWSCC being detected.

321 Alloy 690 TT has also been extensively tested in the various concentrated chemical environments that have been hypothesized as accumulating in secondary side superheated crevices in PWR steam generators by hideout of secondary water contaminants, and as responsible for secondary side attack (IGA / IGSCC) of mill annealed Alloy 600 tubing. Alloy 690 TT displays superior resistance to all other candidate steam generator tube alloys and is only vulnerable in very caustic environments or in the presence of lead (Pb) and possibly, to a lesser extent, to those contaminated with reduced sulphur species. Long-term future performance is, therefore, linked in part to adequate secondary water chemistry control as well as to steam generator design improvements that limit the locations and extent of the impurity concentration process.

322 Improvement factors for IGA / IGSCC resistance of Alloy 690 TT relative to Alloy 600 in the mill annealed condition depend on the particular concentrated chemical environment concerned. When these results are weighted according to the frequency of calculated crevice pH values in the field deduced from hideout return, the resulting overall weighted improvement factor for Alloy 690 TT relative to mill annealed Alloy 600 is 7.6. For Alloy 800, the equivalent figure is 4.5. The weighted improvement factor for Alloy 690 TT

compares to 7.8 deduced from operating experience to 2005 without any IGA / IGSCC being observed in the field.

323 Operating experience of Alloy 690 TT tubing shows that no tubes have been damaged or plugged due to corrosion-related phenomena after up to 20 years in service. Only wear and fatigue have so far been responsible for the limited number of tubes that have been plugged in service. Alloy 800 NG steam generator tubing has also had an excellent record in service and only in the last few years have a small number of tubes been found with secondary side IGA / IGSCC attack after up to 26 years in service. A significant proportion of these cases of IGA / IGSCC is probably due to relaxation of the tube expansion near the top of the tube sheet although recently there have been some observations of cracking above the tube sheet and at the first tube support level, presumably under sludge deposits. As in the case of Alloy 690 TT, once corrosion related problems are minimal, fretting wear becomes the dominant cause of tube plugging of a relatively small number of tubes.

324 Compared to Alloy 800 NG, Alloy 690 TT is a better material regarding most physical properties (i.e. mechanical strength, thermal expansion coefficient and thermal conductivity), and several aspects of corrosion resistance (on the secondary side). Since Alloy 800 NG is an iron based material, it can be better than Alloy 690 TT regarding its contribution to activity build-up in primary circuit because it releases much less nickel and its corrosion products have a positive solubility coefficient with temperature. However, experience of Sizewell B and most recent French plants show that Alloy 690 TT can compete with Alloy 800 NG provided the ID surface condition (i.e. manufacturing process) and hot functional tests have been optimized.

325 Overall, I conclude from the review (Ref. 44) and my general knowledge of this area that Alloy 690 in the Thermally Treated condition is a sound choice of material for Steam Generator Tubing. When supported by detailed manufacturing practice and in-service water chemistry control, Alloy 690TT tubing exhibits good resistance to stress corrosion cracking. However, material choice, manufacturing practice and in-service water chemistry are not a panacea. The general design and construction aspects of the Steam Generator as they affect the tubing also have a role. Important factors are the minimisation of ‘crevice’ conditions, support for the tubing to avoid vibration induced wear and support materials that themselves do not corrode. Most of these general design and construction factors have been understood for many years, and the AP1000 Steam Generator design takes these into account.

326 Perhaps the most telling statement in the review (Ref. 44) is:

> “*Operating experience of Alloy 690 TT tubing shows that no tubes have been damaged or plugged due to corrosion-related phenomena after up to 20 years in service. Only wear and fatigue have so far been responsible for the limited number of tubes that have been plugged in service*.”

6 CONCLUSIONS

327 The specific aims of GDA Step 3 are to:

- improve HSE 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 a significant reduction in regulatory uncertainty.

328 For structural integrity aspects of the AP1000, and from an ND perspective I believe there has been a significant improvement in HSE understanding of the design.

329 Using the ND Safety Assessment Principles and the relevant Technical Assessment Guide, I believe I have identified the significant matters for structural integrity. I have articulated these matters in a number of Regulatory Observations. Westinghouse's responses to these Regulatory Observations have been useful in making progress toward resolution.

330 For components where 'the likelihood of gross failure is claimed to be so low it can be discounted', Westinghouse has indicated a willingness to implement a method of achieving and demonstrating integrity consistent with UK practice. I regard this as substantial progress within GDA Step 3 and a basis for going forward. Execution of the programme of work will extend into GDA Step 4. Full implementation might extend beyond GDA Step 4. If so, some interim work might be done within GDA Step 4 to give confidence for final implementation. I believe ND will want to take an interest in the detail of this work and how the programme of works progresses. This is the subject of Regulatory Observation RO-AP1000-19.

331 There is of course the question of which components have the claim that the likelihood of gross failure is so low it can be discounted. Westinghouse has proposed a programme of work to address this matter. I regard this as substantial progress within GDA Step 3 and a basis for going forward. Execution of the programme of work will extend well into GDA Step 4. I believe ND will want to take an interest in the detail of this work and how the programme of works progresses. This is the subject of Regulatory Observation RO-AP1000-18.

332 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 is the subject of Regulatory Observation RO-AP1000-21. 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. For the purposes of this assessment, I have assumed the Reactor Pressure vessel will have set-in (also referred to as set-through) nozzles, rather than an integral nozzle shell course design. If an integral nozzle shell course was proposed, this would require specific assessment.

333 For neutron irradiation embrittlement of regions of the Reactor Pressure Vessel (RPV), the AP1000 design takes account of what is now known regarding chemical composition of the base materials and welds. However, the end of life maximum neutron dose to the forgings is quite high. I recommend ND seeks an As Low As Reasonably Practicable (ALARP) review of practical options for meaningful reduction of neutron dose to the RPV. As a minimum this should consider the locations of peak neutron dose, which occur in the forging. On the face of it, there is the potential for adopting a 'low leakage core' fuel management arrangement in service. This is the subject of Regulatory Observation RO-AP1000-22.

334 The basis of Reactor Coolant Pump Casing construction based on casting technology has been justified. However, there are still aspects to resolve in how to deal with large repairs to the castings made by welding. The areas still open relate to how to obtain confidence that crack-like defects of a size of concern for integrity, can be detected. This is the subject of Regulatory Observation RO-AP1000-20.

335 Useful progress has been made in understanding the approach to be used for an AP1000 in setting Pressure-Temperature limit curves for the Reactor Pressure Vessel. However, there are aspects still to be resolved, for instance consideration of what is ALARP.

Pressure-Temperature limit curves are the subject of Regulatory Observation RO-AP1000-29.

336 The design of the steel containment shell complies with the American Society of Mechanical Engineers (ASME) code. There are some general aspects that need further consideration: plate thickness available for corrosion allowance for most of the shell, the toughness properties of the plates and welds to meet the requirements for no post weld heat treatment and tolerance on plate thickness relevant to both corrosion allowance and no post weld heat treatment. There are a number of matters to take forward for further assessment. This is the subject of Regulatory Observation RO-AP1000-30.

337 The AP1000 DCD states that the Steam Generator tubing will be made using mill annealed Alloy 690 in the Thermally Treated (TT) condition. Based on my knowledge of UK experience of Thermally Treated Alloy 690 Steam Generator tubing and a general perception of international experience of this material, I had no particular concerns about its use. But, given the past interest in the UK of this aspect of Pressurised Water Reactor (PWR) structural integrity, I judged it prudent to give the matter some consideration. I decided to do this through a support contract to review PWR Steam Generator tube materials and manufacturing routes.

338 Overall for Steam Generator tubing, I conclude from the review and my general knowledge of this area that Alloy 690 in the Thermally Treated condition is a sound choice of material for Steam Generator Tubing.

339 A number of matters are identified above for carrying forward in to GDA Step 4 and some will require significant effort and programmes of work on the part of Westinghouse (e.g. the work for RO-AP1000-19). In addition GDA Step 4 for structural integrity needs to move to the next level of detail and consider the content of documents such as:

- Design Specifications.
- Analyses for loading conditions (mainly thermal-hydraulics analyses - this will require involvement of other ND assessment functions).
- Design Reports.
- Equipment Specifications;

for a range of components.

340 From an ND perspective, I consider there has been a reduction in regulatory risk.

7 RECOMMENDATIONS

341 In this GDA Step 3 assessment of the structural integrity aspects of the AP1000 design proposed for the UK, I have not identified any matters that would lead to a recommendation to raise a Regulatory Issue.

342 During GDA Step 3 I have raised a number of matters with Westinghouse and I have done this mostly through thirteen Regulatory Observations. Some matters raised are relatively more significant than others. I consider useful progress has been made across a number of these Regulatory Observations. Several aspects of these Regulatory Observations remain to be resolved. I consider there is a reasonable prospect of achieving such resolution by carrying these remaining open aspects forward into GDA Step 4.

8 REFERENCES

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"This Design Control Document (DCD) for the Westinghouse AP1000 simplified passive advanced light water reactor plant is incorporated by reference into the Design Certification Rule for the AP1000 design (Section II.A) of Appendix D to 10 CFR Part 52. Revision 16 to the DCD is submitted to the NRC for review and approval application for an amendment to the Design Certification Rule for the AP1000."

2 *UK Compliance Document for AP1000 Design*, Section F - NRC AP1000 Final Safety Evaluation Report (issued as NUREG-1793, September 2004). WEC document UKP-GW-GL-710, Revision 2 (28 July 2008). FSER transmitted to WEC on 13 September 2004. According to Section F Chapter 1, the most recent version of the AP1000 DCD, as of 7 September 2004, was Revision 14. Section F, FSER Supplement 1 is a US NRC Supplement to their FSER (supplement dated December 2005 original issued as NUREG-1793 Supplement No 1), based on the AP1000 DCD up to Revision 15.

3 *AP1000 Pre-Construction Safety Report,* UKP-GW-GL-732, Revision 0, (11 December 2008)

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"Revision 0 of the 'AP1000 European Design Control Document' (DCD) is based on the U. S. AP1000 DCD, Revision 17. In addition, changes were made that provide clarification to allow the document to be applied generically throughout Europe. The following revision history and the change bars within the body of the DCD will enable the reader to identify the changes made from U.S. DCD Revision 16 to Revision 17."

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Table 1

List of Regulatory Observations. UK AP-1000 ND Generic Design Assessment - Step 3 Structural Integrity - Metal Components and Structures

RO Number and TRIM Reference	Regulatory Observation Title
RO-AP1000-18 2009 /106360	Categorisation of Safety Function, Classification of Structures, Systems and Components
RO-AP1000-19 2009 /106436	Avoidance of Fracture - Margins Based on Size of Crack-Like Defects Integration of Material Toughness Properties, Non-Destructive Examinations During Manufacture and Analyses for Limiting Sizes of Crack-Like Defects
RO-AP1000-20 2009 /106521	Manufacturing Method for Reactor Coolant Pump Casings
RO-AP1000-21 2009 /106565	Materials Specifications and Selection of Material Grade - Reactor Pressure Vessel, Pressuriser, Steam Generator Shells
RO-AP1000-22 2009 /106602	Reactor Pressure Vessel Body Cylindrical Shell Forging Material and Associated Circumferential Welds Effects of Irradiation
RO-AP1000-23 2009 /106628	Primary Circuit Vessel Nozzle to Safe End Welds
RO-AP1000-24 2009 /106640	Information on Reactor Internals
RO-AP1000-25 2009 /106655	ASME Design Specifications and Design Reports - Current Status and As-Built Status
RO-AP1000-26 2009 /106678	Fatigue Crack Initiation - Conservatism in ASME III Appendix I S-N Curves for Stainless Steel Material
RO-AP1000-27 2009 /106702	Reactor Internals - Testing and Inspection Programme of First AP1000 as "Prototype"
RO-AP1000-28 2009 /106736	Pressuriser Surge Line Stratification Evaluation - AP-1000 First Plant Only Test
RO-AP1000-29 2009 /106752	Reactor Pressure Vessel and Primary Circuit Pressure - Temperature Limits and Low Temperature Overpressure Protection
RO-AP1000-30 2009 /106762	Containment Pressure Shell ASME SA738 Grade B

Table 2

Main Parts of UK AP1000 PCSR Relevant to Structural Integrity Assessment

PCSR Section	PCSR Section Title
Chapter 1. Introduction	
1.4	Nuclear Safety Case Claims of the AP1000
Chapter 3. Management of Safety	
3.3.1	Design
3.3.3	Commissioning
3.3.6	Quality Assurance
Chapter 5. Safety Assessment Process	
5.3	Safety Objectives and Scope
5.4	Hazard Identification Process
5.4.2.1	Hazard Identification for Design Basis Assessment
5.4.2.3	Hazard Identification for Internal and External Hazards
5.4.3	Consideration of US Evidence to Demonstrate UK Safety Assessment Process Outputs Provided
5.4.3.1	Plant and Process Hazards
5.4.3.2	Internal and External Hazards
5.5	Design Basis Assessment
5.5.3	Design Substantiation
5.5.5	Special Considerations Quote (text the same in 2008 and 2009 versions of PCSR): This section of the PCSR considers DBA safety arguments that have been developed using an approach outside the standard DBA process. There is one such specific area, concerning the safety argument demonstrating the structural integrity of primary systems pipe work to be acceptable. The approach specified in the DCD diverges from UK relevant good practice, as represented in Reference 42. (Ref. 42 being: International Journal of Pressure Vessels and Piping, The Demonstration of Incredibility of Failure in Structural Integrity Safety Cases, R Bullough, F M Burdekin, O J V Chapman, V R Green, D P G Lidbury, J N Swingler and R Wilson, 2001, pages 539-552.) To further align this approach with UK relevant good practice, consideration will be given to supplementing the established NRC basis for the integrity of the AP1000 piping systems using fracture analyses of selected welds based upon credible fabrication manufacturing defects and the capability of the inspections applied during manufacture and pre-service. For the Leak before Break and the Break Exclusion Zone Categories, consideration will be given to presenting the overall arguments in terms of quality of design and manufacture, functional testing of the pipe work, quantification of the defect tolerance

PCSR Section	PCSR Section Title
	of the pipe work and indicators of forewarning of failure. The use of such an argument structure will expose the existing piping quality and integrity arguments in a way that is readily appreciable by a UK audience, and will allow judgements of the overall integrity of the systems to be made.
Chapter 6. Description of the Plant Systems and their Conformance with the Design Requirements	
6.1.2	Sources of Information
6.2	Reactor
6.2.1	System Duty Functions
6.2.1.1	System Safety Functions
6.2.3.2	Reactor Materials (only core support components)
6.3	Reactor Coolant System and Associated Systems
6.3.1	System Duty Functions
6.3.1.1	System Safety Functions
6.3.2	System Description
6.3.2.3	System Classification
6.3.3	Conformance with Design Requirements
6.3.3.2	Integrity of Reactor Coolant Pressure Boundary
6.3.3.3	Integrity of the Reactor Pressure Vessel
6.3.3.4	Reactor Coolant Pumps
6.3.3.5	Steam Generators
6.3.3.6	Reactor Coolant System Piping
6.3.3.7	Pressuriser
6.3.3.11	Reactor Coolant System Pressure Relief Devices
6.3.3.13	Core Make-Up Tank
6.3.4	Conformance with Safety Requirements
6.4.1	Containment System (only for the containment vessel which is a free standing cylindrical steel vessel with ellipsoidal upper and lower heads)
6.8	Power Conversion Systems (the main steam lines, feedwater lines and also feedwater quality)
Chapter 9. ALARP Assessment of the Design of the AP1000	
9.5.4	ALARP Review of the Principal Decisions during the Development of the AP1000 Design
9.5.7.1	Low Leakage Steel Containment
9.5.9	Primary System Design (does not address pressure boundary components, e.g. selection of reactor pressure vessel diameter)

PCSR Section	PCSR Section Title
Chapter 11. Operational Aspects	
11.4	Operating Instructions Quote: This section will describe the arrangements for the production and control of the operating instructions…This will be completed in the site licence application phase.
11.7	Maintenance, Surveillance, Inspection and Testing Quote: This section will describe….This is detailed in Chapters 14 and 16 of reference 1. Reference 1 in the quote is reference 6 here.
11.9	Management of Ageing Quote: This section will summarise….This is detailed in reference 1 Chapter 3 Appendix D. Reference 1 in the quote is reference 6 here

Table 3

Main Parts of AP1000 DCD Rev 17 Relevant to Structural Integrity Assessment

DCD Section	Comments
Chapter 3. Design of Structures, Components, Equipment and Systems	
3.1	Conformance with Nuclear Regulatory Commission General Design Criteria (Criteria as found in 10 CFR 50 Appendix A) **Criterion 4 - Environmental and Missile Design Bases** Quote: Structures, systems, and components important to safety shall be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accidents. These structures, systems, and components shall be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures and from events and conditions outside the nuclear power unit. However, dynamic effects associated with postulated pipe ruptures in nuclear power units may be excluded from the design basis when analyses reviewed and approved by the Commission demonstrate that the probability of fluid system piping rupture is extremely low under conditions consistent with the design basis for the piping. **Criterion 14 - Reactor Coolant Pressure Boundary** Quote: The reactor coolant pressure boundary shall be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture. **Criterion 15 Reactor Coolant System Design** Quote: The reactor coolant system and associated auxiliary, control, and protection systems shall be designed with sufficient margin to assure that the design conditions of the reactor coolant pressure boundary are not exceeded during normal operation, including anticipated operational occurrences. (The reactor coolant system stress analysis and the leak-before-break analyses are described in Appendices 3B and 3C. See Section 5.3 for additional information.) **Criterion 16 - Containment Design** Quote: The reactor containment and associated systems shall be provided to establish an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment and to assure that the containment design conditions important to safety are not exceeded

DCD Section	Comments
	for as long as postulated accident conditions require. **Criterion 30 - Quality of Reactor Coolant Pressure Boundary** Quote: Components which are part of the reactor coolant pressure boundary shall be designed, fabricated, erected and tested to the highest quality standards practical. Means shall be provided for detecting and, to the extent practical, identifying the location of the source of reactor coolant leakage. **Criterion 31 - Fracture Prevention of Reactor Coolant Pressure Boundary** Quote: The reactor coolant pressure boundary shall be designed with sufficient margin to assure that when stressed under operating, maintenance, testing, and postulated accident conditions (1) the boundary behaves in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized. The design shall reflect consideration of service temperatures and other conditions of the boundary material under operating, maintenance, testing, and postulated accident conditions and the uncertainties in determining (1) material properties, (2) the effects of irradiation on material properties, (3) residual, steady state, and transient stresses, and (4) size of flaws. **Criterion 32 - Inspection of Reactor Coolant Pressure Boundary** Quote: Components which are part of the reactor coolant pressure boundary shall be designed to permit (1) periodic inspection and testing of important areas and features to assess their structural and leak-tight integrity and (2) an appropriate material surveillance program for the reactor pressure vessel. **Criterion 51 - Fracture Prevention of Containment Pressure Boundary** Quote: The reactor containment boundary shall be designed with sufficient margin to assure that under operating, maintenance, testing, and postulated accident conditions (1) its ferritic materials behave in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized. The design shall reflect consideration of service temperatures and other conditions of the containment boundary material during operation, maintenance, testing, and postulated accident conditions, and the uncertainties in determining (1) material properties, (2) residual, steady-state, and transient stresses, and (3) size of flaws.
3.2	Classification of Structures, Components and Systems (especially Table 3.2-1 and Table 3.2-3)
3.6	Protection Against the dynamic Effects Associated with the Postulated Rupture of Piping
3.9	Mechanical Systems and Components

DCD Section	Comments
3.9.1	Special Topics for Mechanical Components
3.9.1.1	Design Transients
3.9.3.1	Loading Combinations, Design Transients and Stress Limits
APPENDIX 3B	Leak-Before-Break Evaluation of the AP1000 Piping (mainly Table 3B-1)
APPENDIX 3E	High-Energy Piping in the Nuclear Island
Chapter 4. Reactor	
4.5	Reactor Materials
4.5.2	Reactor internals and Core Support Materials
Chapter 5. Reactor Coolant System and Connected Systems	
5.1	Summary Description
5.2	Integrity of Reactor Coolant Pressure Boundary
5.3	Reactor Vessel
5.4	Component and Subsystem Design
Chapter 6. Engineered Safety Features	
6.1	Engineered Safety Features Materials
6.3	Passive Core Cooling System
6.3.2.2.2	Accumulators (including Table 6.3-2)
Chapter 10. Steam and Power Conversion	
10.3	Main Steam Supply System
10.4	Other Features of Steam and Power Conversion System
10.4.1	Main Condensers
10.4.1.1.1	Safety Design Basis Quote: The main condenser serves no safety-related function and therefore has no nuclear safety design basis. (comment: Other than keeping the ultimate heat sink fluid from mixing with the condensate / feedwater!)
Chapter 16. Technical Specifications	
16.1	Technical Specifications 3.4 Reactor Coolant System 3.4.7 RCS Operational Leakage LCO 3.4.7(a) no pressure boundary leakage

Table 4

Claims in UK AP1000 PCSR Section 1.4

PCSR Section	Claim
1.4.1	Nuclear Safety Analysis has been performed that proves that the AP1000 can be operated safely
1.4.2	The AP1000 has been designed so that all Risk is ALARP
1.4.3	The AP1000 Design can be built in accordance with the Design Intent
1.4.4	The AP1000 can be properly commissioned without Contravening the Safety case
1.4.5	The AP1000 can be Operated and Maintained in accordance with the Design Intent
1.4.6	The Design of the AP1000 includes Facilities and Equipment to Facilitate the Management of Emergencies
1.4.7	The Environmental Impact of the AP1000 is within the Authorised Limits and Operational Targets
1.4.8	The AP1000 Design Allows for the Proper Management of radioactive Waste and Spent Fuel
1.4.9	The AP1000 can be decommissioned at an Acceptable Level of Safety

Table 5

Recasting US NRC General Design Criteria in a 'Claims' Format

Criterion	Recasting
Criterion 4 Structures, systems, and components important to safety shall be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accidents. These structures, systems, and components shall be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures and from events and conditions outside the nuclear power unit. However, dynamic effects associated with postulated pipe ruptures in nuclear power units may be excluded from the design basis when analyses reviewed and approved by the Commission demonstrate that the probability of fluid system piping rupture is extremely low under conditions consistent with the design basis for the piping.	**Claim:** The probability of fluid system piping rupture is claimed extremely low for specified piping under conditions consistent with the design of the piping. Hence the dynamic effects associated with postulated pipe rupture are excluded from the design basis. This claim is supported by arguments, in terms relating to the structural integrity of the specified piping. The arguments are supported by evidence. Criterion 4 has this specific potential exclusion of pipework. Presumably Criterion 14 does not by itself apply to pipework, even if the pipework is part of the reactor coolant pressure boundary. There may be other components not explicitly addressed by either Criterion 4 or 14, for instance the accumulators. There are also the Core Makeup tanks, but they might fall under Criterion 14.
Criterion 14 The reactor coolant pressure boundary shall be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture.	**Claim:** The reactor coolant pressure boundary has an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture. The likelihood of gross rupture is so low it can be discounted.
Criterion 16 The reactor containment and associated systems shall be provided to establish an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment and to assure that the containment design conditions important to safety are not exceeded for as long as postulated accident conditions require.	**Claim:** The reactor containment and associated systems provide an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment for as long as postulated accident conditions require.

Table 6

Recasting US NRC General Design Criteria in an 'Arguments' Format

Criterion	Recasting
Criterion 15 The reactor coolant system and associated auxiliary, control, and protection systems shall be designed with sufficient margin to assure that the design conditions of the reactor coolant pressure boundary are not exceeded during normal operation, including anticipated operational occurrences. (The reactor coolant system stress analysis and the leak-before-break analyses are described in Appendices 3B and 3C. See Section 5.3 for additional information.)	Starting point for arguments that support the Claims associated with Criteria 4 and 14
Criterion 30 Components which are part of the reactor coolant pressure boundary shall be designed, fabricated, erected and tested to the highest quality standards practical. Means shall be provided for detecting and, to the extent practical, identifying the location of the source of reactor coolant leakage.	Starting point for arguments that support Claims associated with Criteria 4 and 14
Criterion 31 The reactor coolant pressure boundary shall be designed with sufficient margin to assure that when stressed under operating, maintenance, testing, and postulated accident conditions (1) the boundary behaves in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized. The design shall reflect consideration of service temperatures and other conditions of the boundary material under operating, maintenance, testing, and postulated accident conditions and the uncertainties in determining (1) material properties, (2) the effects of irradiation on material properties, (3) residual, steady state, and transient stresses, and (4) size of flaws.	Starting point for arguments that support the Claims associated with Criteria 4 and 14

Criterion	Recasting
Criterion 32 Components which are part of the reactor coolant pressure boundary shall be designed to permit (1) periodic inspection and testing of important areas and features to assess their structural and leak-tight integrity and (2) an appropriate material surveillance program for the reactor pressure vessel.	Starting point for arguments that support the Claims associated with Criteria 4 and 14
Criterion 51 The reactor containment boundary shall be designed with sufficient margin to assure that under operating, maintenance, testing, and postulated accident conditions (1) its ferritic materials behave in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized. The design shall reflect consideration of service temperatures and other conditions of the containment boundary material during operation, maintenance, testing, and postulated accident conditions, and the uncertainties in determining (1) material properties, (2) residual, steady-state, and transient stresses, and (3) size of flaws.	Starting point for arguments that support the Claims associated with Criteria 16

Table 7

GDA Step 2 Questions and Responses

Question	Response
1. Pressure Boundary Components	
QUESTION: For which pressure boundary components inside containment is the claim made that the likelihood of gross failure is so low that it can be discounted?	The following pressure boundary components inside containment are designed such that the likelihood of gross component failure is low enough that gross failures of these components are not considered design basis events: 1. Reactor vessel a. Below the top of the core b. Above the top of the core (including the RV head) 2. Pressurizer 3. Reactor Coolant Pump Casing 4. Steam Generator a. Channel Head b. Tube Sheet c. Multiple Steam Generator Tubes (Single tube analyzed in DCD Section 15.6.3) d. Secondary Shell 5. Core Makeup Tank 6. Accumulator 7. PRHR a. Channel head b. Tube sheet 8. Valve Bodies
4. Steam Generators	
QUESTION: Are the Steam Generator secondary shells designed, manufactured, inspected and tested in full compliance with ASME III Class 1 (i.e. Subsection NB of the Code)? If so, does this mean the scope and extent of pre- and in-service inspections using ASME XI would be as for other ASME III Class 1 vessels (e.g. the RPV)?	Yes, the Steam Generator secondary shells designed, manufactured, inspected and tested in full compliance with ASME III Class 1 (i.e. Subsection NB of the Code). The scope and extent of pre- and in-service inspections using ASME XI, however, would be as for other ASME III Class 2 vessels since ASME XI allows for this reclassification.
QUESTION: For the weld joining the Reactor Coolant Pump bowl to the Steam Generator channel head, do the geometry, accessibility and materials of construction, allow volumetric examination (including ultrasonic examination) from both sides of the weld; during manufacture and during service?	The steam generator channel head to reactor coolant pump casing weld is considered a Category B-F weld under ASME Code Section XI (1998 Edition, up to and including 2000 Addenda). Volumetric (UT) and surface (PT) examinations are required every 10 years. The examination volume for the UT examinations is defined as the inner 1/3 of the thickness plus 13mm on each side of the weld. The approach is to perform the UT examinations from the inside surface due to

Question	Response
	limited outside surface access for personnel and inspection equipment, the component thickness, the defined examination volume, and the materials of construction (particularly the cast stainless steel pump casing). The same region will also be examined during manufacturing as a baseline examination. The geometry of the weld is such that the UT examination can be made from both sides of the weld on the ID surface with no limitations, and current ID surface-applied UT processes have been demonstrated capable of detecting ID-initiated degradation in these materials of construction. Some development will be required for the remote scanning device which will be introduced through the steam generator channel head manway. The UT examinations must be performed per ASME Section XI, Appendix VIII Supplement 10. These requirements necessitate procedures and personnel that must undergo blind performance demonstrations that mandate compliance with defined flaw detection and sizing criteria. Existing Appendix VIII-demonstrated UT procedures are not sufficient for this weld configuration because the existing Appendix VIII Supplement 10 qualification program (PDI) does not currently have samples representing this configuration. Also, the cast stainless steel casing side of the weld is not currently addressed in Appendix VIII. A program to develop industry acceptable procedures and testing samples for this configuration/material will be initiated by Westinghouse.
QUESTION: Is gross failure of a channel head divider plate part of the design basis?	A complete structural analysis of the divider plate is performed to demonstrate compliance with all applicable ASME Code criteria. This analysis evaluates all limiting locations on the divider plate for all limiting conditions and therefore, protection against gross failure of the divider plate is inherent in the analysis. The analysis of the generator for a complete failure of the divider plate is not part of the design basis. The divider plate is not a structural member that supports the tubesheet, therefore, structural failure of the divider plate will not impact the overall integrity of the steam generator.
FOLLOW-ON QUESTION: For the purposes of system thermal-hydraulic response	A full or partial failure of the divider plate between the hot leg and cold leg side of the

Question	Response
analyses, is partial or complete failure postulated of the Divider Plate in the Channel Head of a Steam Generator?	steam generator channel head would result in flow bypassing the steam generator tubes, and a loss of heat removal capability from the affected steam generator. The cold leg flow in the failed loop would return to the reactor vessel at a higher temperature than the intact loop. This results in an S-signal which will trip the reactor, trip the RCPs, and isolate the steam generators. In addition, the isolation valves will be opened on the passive core cooling system, specifically the core make up tanks (CMTs) and the passive residual heat removal heat exchanger (PRHR-HX) discharge lines. The CMTs inject cold, borated water to the reactor vessel, while the PRHR-HX helps to remove the reactor decay heat. These two components are safety-grade equipment and are designed to operate with single failures. The PRHR-HX is the safety-grade decay heat removal system for AP1000 for non-LOCA faults. This event is enveloped by several AP1000 safety analyses including loss of feedwater, loss of AC power, and main steam line break. The passive core cooling system is designed to provide for decay heat removal separately from the steam generators. This is a significant change from plants with active systems that can only rely on the steam generators for safety-grade decay heat removal.
5. Pressuriser	
QUESTION: What material is used for the Pressuriser heater wells and instrument nozzles?	The current specified material for the pressurizer heater wells/sleeves is SA-182, Type 316 stainless steel.
6. Reactor Coolant Pumps	
QUESTION: What are the materials of construction of the Reactor Coolant Pump pressure boundary, including the pump bowl, the stator shell, closures and bolting systems?	The materials of construction of the main Reactor Coolant Pump pressure boundary components are as follows: Pump Casing ASME SA-351, CF8A Stator Housing Shell ASME SA-508, Grade 1 Stator Housing Upper Flange ASME SA-508, Grade 1 Stator Housing Lower Flange ASME SA-336. Grade F304 Main Closure Studs and Nuts ASME SA-540, Grade B23 or B24, Class 4 Stator Closure ASME SA-182 Grade F304

Question	Response
	Stator Closure Ring ASME SA-182 Grade F304 Stator End Cap ASME SA-182 Grade F304 Stator End Cap Closure Bolts ASME SA-540, Grade B23 or B24, Class 2 Note that the stator housing shell and the stator housing upper flange are not normally in contact with the reactor coolant. They are kept dry by the stator can. Materials for the external heat exchanger and its connecting piping, and for smaller components such as canopy seals and sensor wells, are not included in the above listing. Final selection of material for these items has not been completed.
11. Main Steam Lines	
QUESTION: What ASME III Code Classes are used for design of the Main Steam Lines from the Steam Generator to the pipe restraint at the auxiliary / turbine building wall?	The definition of the AP1000 Equipment Classification System is contained in UKP-GW-GL-700 Section 3.2.2, "AP1000 Classification System." Specific to the regulator inquiry, Sections 3.2.2.4 and 3.2.2.5 contain the definitions of Equipment Class B and Equipment Class C, respectively. As documented in UKP-GW-GL-700, Class B does designate the application of ASME Section III Class 2 (Subsection NC) design requirements. Class C equipment is designed in accordance with ASME Section III Class 3 (Subsection ND) requirements.
QUESTION: What materials / fabrication route is used to manufacture the Main Steam Line where the Main Steam Safety Valve branch connections are made?	The AP1000 Main Steam Safety Valve (MSSVs) inlet piping is designed in accordance with ASME Section III: - The individual MSSV inlets divert flow from the main steam line using a 38x8 Sweepolet fitting. The AP1000 main steam line is fabricated from SA-335 Grade P11 alloy steel. The Sweepolet fitting is manufactured from a compatible alloy steel material (SA-182 Grade F11, for example) and welded to the main steam line. A Sweepolet is a smooth transition fitting designed to reduce inlet entrance losses and improve weld quality by improving access to the weld location. - The MSSV inlets are manufactured from a long welding neck fitting (1500 pound pressure class). The long welding neck is manufactured from alloy steel compatible with the

Question	Response
	Sweepolet. The inner diameter of the long welding neck is 8 inches (20.3 cm), in accordance with ASME Section III, Article NC-7000, “Overpressure Protection.” AP1000 MSSV design and installation shall be in accordance with ASME Section III, Article NC-7000.
12. Overpressure Protection	
QUESTION: Does overpressure protection of the Main Steam System depend on the reactor protection system, or is the steam relief flow capacity sufficient to avoid exceeding 110% of design pressure without the aid of a reactor trip?	The AP1000 Main Steam Safety Valves (MSSVs) have been sized in accordance with ASME Section III, Article NC-7000, “Overpressure Protection.” As required by the ASME code, the MSSV have been sized to prevent the Steam Generator from exceeding 110% of the design pressure as a result of both expected and unexpected transients, independent of operation of the reactor trip system. As documented in Table 10.3.2-2, the Steam Generator MSSV relieving capacity at 110% design pressure is 8,240,000 lbm/hr (3,740,000 kg/hr), or 110% of the 7,490,000 lbm/hr (3,400,000 kg/hr) design flow listed in Table 10.3.2-1. The system description in Subsection 10.3.2.2.2 states: Main steam safety valves with sufficient rated capacity are provided to prevent the steam pressure from exceeding 110 percent of the main steam system design pressure: • Following a turbine trip without a reactor trip and with main feedwater flow maintained • Following a turbine trip with a delayed reactor trip and with the loss of main feedwater flow These operating conditions are identified because the Westinghouse Steam Systems Design Manual and prior ASME Overpressure Protection Reports for secondary systems prepared by Westinghouse have identified a “Turbine trip

Question	Response
	without reactor trip" event represents the maximum MSSV demand and bounds expected and unexpected system transient response of the MSSVs.
14. Load Combinations and Stress Limits	
QUESTION: Are the stress limits in UKP-GW-GL-700 Tables 3.9-6 and 3.9-7 for seismic anchor motion consistent with not using Articles NB–3200, NB–3600, NC–3600, and ND–3600 of the ASME III code later than the 1993 Addenda?	The ASME Section III piping meets the ASME Code up to and including the 1989 Addenda for Articles NB/NC/ND-3600 not the 1994 Addenda. The stress limits in Tables 3.9-6 and 3.9-7 for seismic anchor motions are not the same as the stress limit in the ASME Code, 2000 Addenda. The stress limits in Tables 3.9-6 and 3.9-7 are intended to supplement the requirements of the ASME Code, 1989 Addenda. They provide similar protection to the stress limits in the 2000 Addenda.
QUESTION: Is there other nonsafety-related piping treated in the same manner as the CVS piping inside containment?	These equation 9 stress limits are also used for other ASME B31.1 piping as follows. The B31.1 piping that is connected to the ASME Section III piping meets these limits. The B31.1 piping that could adversely affect adjacent safety related components also meets these limits.

Table 8

Amended Version of Table 1 from Project initiation Document (PID) (Ref. 14)

Topic	GDA Step 3 Detailed Scope
1. NOT USED*	
2. Components and Systems to be Considered	**PWR** Reactor Pressure Vessel Core support structures Pressuriser Steam Generators - Primary and Secondary Side and Tubing Reactor Coolant Pumps pressure boundary and flywheel Primary Coolant Loop Pipework Pipework connecting auxiliary systems to the primary circuit Steam pipework from the Steam Generators Feedwater pipework to the Steam Generators Free-standing metal containment pressure boundary *(interface with Civil Engineering assessment)*
3. Level of Integrity Required for Nuclear Safety Claim	Identification of components where likelihood of gross failure is so low it can be discounted. This must be completed within the Step 3 period.
4. Safety Classification and Standards - Including Quality Assurance	Safety classification must be completed within Step 3 and assessment conclusion reached. Standards to be used for design, manufacture and installation must be identified and assessment conclusion reached on overall acceptability of standards proposed. Framework of quality assurance must be declared in Step 3 and assessment conclusion reached. Principle and outline of third party inspection agent for in-manufacture inspection should be agreed. General review by ND of standards for design, manufacture and installation.
5. Potential Failure Modes	No assessment needed. This topic is part of the explanation of the assessment approach.
6. Potential In-Service Degradation Modes (liked with 17. below)	Evidence of knowledge of and mitigation measures applied for known potential in-service degradation mechanisms. Assessment conclusion for treatment of potential in-service degradation mechanisms.
7. Analysis - Design Analysis, Fracture Mechanics Analyses	Evidence that general, top level analysis is available for sizing pressure boundary and other structural integrity components. Assessment of this top level analysis. Evidence of capability to perform fracture mechanics analyses or manage procurement of such analyses for determination of Validation Factors and other purposes.

Topic	GDA Step 3 Detailed Scope
	Evidence supported by examples. Agreement on principle and outline of use of fracture mechanics analyses along with examination qualification and material supply specification to include minimum fracture toughness.
8. Loadings	List of normal, expected operating transients and fault condition loads with definition in overall parameters such as pressure, fluid temperatures and mechanical loads. Indication of how specific parameters for individual components will be determined, e.g. through-wall temperature variation.
9. Materials - Choice and Specifications	Materials for all components for review in Step 3 (see 2. above) defined. Assess materials choices. Specifications for materials for all components for review in Step 3 defined. Assess materials specifications
10. Fabrication Design and Processes	Fabrication design (e.g. plate, forging, location of welds) proposals available for assessment for those components listed for review in Step 3. Assess taking account of what is likely to be possible by the time of manufacture of specific plant components and contribution to integrity of fabrication design options. Approach to qualifying manufacturing processes (e.g. welding) available. Assess general arrangements for qualifying manufacturing processes.
11. In-Manufacture Examinations - Scope, Extent. Qualification of Procedures, Equipment and Personnel	Scope and extent of in-manufacture examinations available. Assessment of overall proposals for qualification of procedures equipment and personnel. Assessment looking for proposals consistent with ENIQ approach and where examination is for planar, crack-like defects, consistent with determination of Validation Factors.
12. Procedural Control of Design, Manufacture and Installation	Top tier organisation arrangements for control of design, manufacture and installation available, and how top tier organisations define requirements for lower tier organisations. Assessment of procedural control arrangements. Related to quality management systems and quality assurance (see 2 above).
13. In-Manufacture Inspection	Agree concept of third party inspection agent and role of operator/licensee in procuring third party inspection agent.

Topic	GDA Step 3 Detailed Scope
14. Pressure System - Discharge and Flow Aspects	Primary and secondary system over-pressure protection system concept available and type of pressure relief valves defined. Assess over-pressure protection system concept. Review reliability of reactor systems (e.g. trip) to over-pressure protection system.
15. Pre-Service Examination - Scope, Extent. Qualification of Procedures, Equipment and Personnel	Scope and extent of pre-service examinations available. Related to in-service examinations. Assessment of overall proposals for qualification of procedures equipment and personnel. Assessment looking for proposals consistent with ENIQ approach and where examination is for planar, crack-like defects, consistent with determination of Validation Factors.
16. Definition of Operating Envelope	Evidence of process for defining an operating envelope.
17. Establish In-Service Monitoring, Examination and Testing Requirements (linked with 6 above)	Evidence of process for defining requirements or advice for in-service monitoring, examination and testing requirements.
* Numbering of topics used section numbering of report, section 1 is the Introduction	

Table 9

Amended Version of Table 1 from Project initiation Document (PID)
(Ref. 14) - How Topics Dealt With

Topic	How Dealt in GDA Step 3
1. NOT USED*	
2. Components and Systems to be Considered	RO-AP1000-18, RO-AP1000-20, RO-AP1000-21
3. Level of Integrity Required for Nuclear Safety Claim	RO-AP1000-18
4. Safety Classification and Standards - Including Quality Assurance	RO-AP1000-18, DCD Chapter 3.2
5. Potential Failure Modes	No assessment needed. This topic is part of the explanation of the assessment approach.
6. Potential In-Service Degradation Modes (liked with 17. below)	RO-AP1000-22, RO-AP1000-24
7. Analysis - Design Analysis, Fracture Mechanics Analyses	Design analysis - assessment of ASME code and RO-AP1000-25. Fracture mechanics RO-AP1000-19
8. Loadings	Assessment of DCD Chapter 3.9
9. Materials - Choice and Specifications	RO-AP1000-21, RO-AP1000-22, RO-AP1000-30
10. Fabrication Design and Processes	RO-AP1000-20, RO-AP1000-22, RO-AP1000-23, RO-AP1000-24, RO-AP1000-30
11. In-Manufacture Examinations - Scope, Extent. Qualification of Procedures, Equipment and Personnel	RO-AP1000-19
12. Procedural Control of Design, Manufacture and Installation	Assessment of ASME code
13. In-Manufacture Inspection	Assessment of ASME code
14. Pressure System - Discharge and Flow Aspects	Assessment of ASME code
15. Pre-Service Examination - Scope, Extent. Qualification of Procedures, Equipment and Personnel	Deferred to Step 4, along with in-service examination.
16. Definition of Operating Envelope	RO-AP1000-29
17. Establish In-Service Monitoring, Examination and Testing Requirements (linked with 6 above)	RO-AP1000-22
* Numbering of topics used section numbering of report, section 1 is the Introduction	

Table 10

Predictions of RT_{NDT} shift in AP1000 RPV (°C) from Ref. 19

		RG 1.99 Rev 2	EONY	RM-9
Not a low-leakage core				
	Weld 2	57	42	52
	Lower Shell	31	55	56
Low-leakage core				
	Weld 2	48	32	42
	Lower Shell	28	42	45

Table 11

Predicted End of Life RT_{NDT} for AP1000 RPV (°C)
using US NRC Reg Guide 1.99 Rev 2
RT_{NDT} shifts from Ref. 19, Start of Life RT_{NDT} from DCD Table 5.3-3

	RG 1.99 Rev 2 Shift ΔRT_{NDT}	Margin* M	SOL RT_{NDT}	EOL RT_{NDT}
Not a low leakage core				
Weld 2	57	31.1	-28.9	59.2
Lower Shell	31	18.8	-23.3	26.5

The method of determining EOL RT_{NDT} in regulatory Guide 1.99 Revision 2 and 10 CFR 50.61 are the same.

* Margin M as defined in USNRC Regulatory Guide 1.99 Rev 2 (Ref. 32):

$M = \sqrt{\sigma_I^2 + \sigma_\Delta^2}$ with σ_I the standard deviation of the initial, (measured in this case) RT_{NDT} and σ_Δ the standard deviation of the shift ΔRT_{NDT}. When the initial RT_{NDT} is measured, the standard deviation is estimated from the precision of the test; with no information here this has been taken as zero. Reg Guide 1.99 Rev 2 for σ_Δ gives 28°F for welds and 17°F for base metal.
Here then $M = 2\sigma_\Delta$ and converting to °C (recalling this is a change in temperature) M is 31.1°C for weld and 18.8°C for base metal.

Note: the shift and margins ΔRT_{NDT}, M are temperature changes and converting to °F adjustment for zero offset should NOT be used, i.e. just use 9/5 going from °C to °F. SOL and EOL RT_{NDT} are actual temperatures and converting to °F the adjustment for zero offset should be used, i.e. use 9/5 then add 32 going from °C to °F. This is an obvious point, but is a common error that has been found in several documents related to this assessment.

Table 12

Summary of US NRC Requirements for end of life RT_{NDT}

USNRC Regulatory Guide 1.99 Revision 2 - Section C, Regulatory Position, sub-section 3 - Requirement for New Plants:
"For beltline materials in the reactor vessel for a new plant, the content of residual elements such as copper, phosphorus, sulfur, and vanadium should be controlled to low levels. The copper content should be such that the calculated adjusted reference temperature at the 1/4T position in the vessel wall at end of life is less than 200^{o}F (93.3^{o}C). In selecting the optimum amount of nickel to be used, its deleterious effect on radiation embrittlement should be balanced against its beneficial metallurgical effects and its tendency to lower the initial RT_{NDT}." Note: the adjusted reference temperature at end of life is end of life RT_{NDT} The adjusted reference temperature at the inner surface of the RPV will be higher than at the 1/4T position (i.e. a quarter the way through the wall from the inner surface).
10 CFR 50.61 Fracture toughness requirements for protection against pressurized thermal shock events:
(b)(2) pressurised thermal shock screening criterion: 270^{o}F (132^{o}C) for plates, forgings, and axial weld materials, and 300^{o}F (149^{o}C) for circumferential weld materials In the determination of the shift in RT_{NDT} in 10 CFR 50.61, (c)(1)(iv)(B) states the fluence to be used is the best estimate neutron fluence in n/cm^2 (E>1Mev) at the clad-base metal interface on the inside surface of the vessel…..

Table 13

Main Features of Methodology in ASME XI Appendix G for Determination of Pressure - Temperature Limits Curves

(i) postulated defects are surface defects with a depth 0.25 the wall thickness and a defect length along the surface of 6 times the defect depth (for the relevant wall thickness in the beltline of the UK AP-1000); (ii) defects are postulated on both the inside and outside surfaces (to account for thermal stress profiles through the wall for both start-up and shutdown conditions); (iii) axial defects are postulated for plates, forgings and axial welds; circumferential defects are postulated for circumferential welds; (iv) applied stresses are those arising from Level A and B Service Limit loadings; (iv) the requirement to be satisfied and from which an allowable pressure can be determined for any assumed rate of temperature changes is: $2K_{Im} + K_{It} < K_{Ic}$ (K_{Ic} used by virtue of item 2 above) for locations away from nozzles, flanges and shell regions near geometric discontinuities; (v) for locations involving nozzles, flanges and shell regions near geometric discontinuities, primary bending stress and secondary membrane and bending stresses are required to be considered in addition to primary membrane and thermal stresses. (vi) K_{Ic} values ‘at the locations of interest’ are determined using a reference curve (for unirradiated material) provided in ASME. The definition of ‘locations of interest’ includes the crack tip at its deepest point, and takes account of the attenuation of neutron flux through the vessel wall (1/4T or 3/4T position).
Not part of ASME but: the shift in reference temperature is determined using US NRC Regulatory Guide 1.99 (Revision 2).

Figure 1

AP1000 RPV side elevation with various parts and welds identified. (Dimensions in inches)

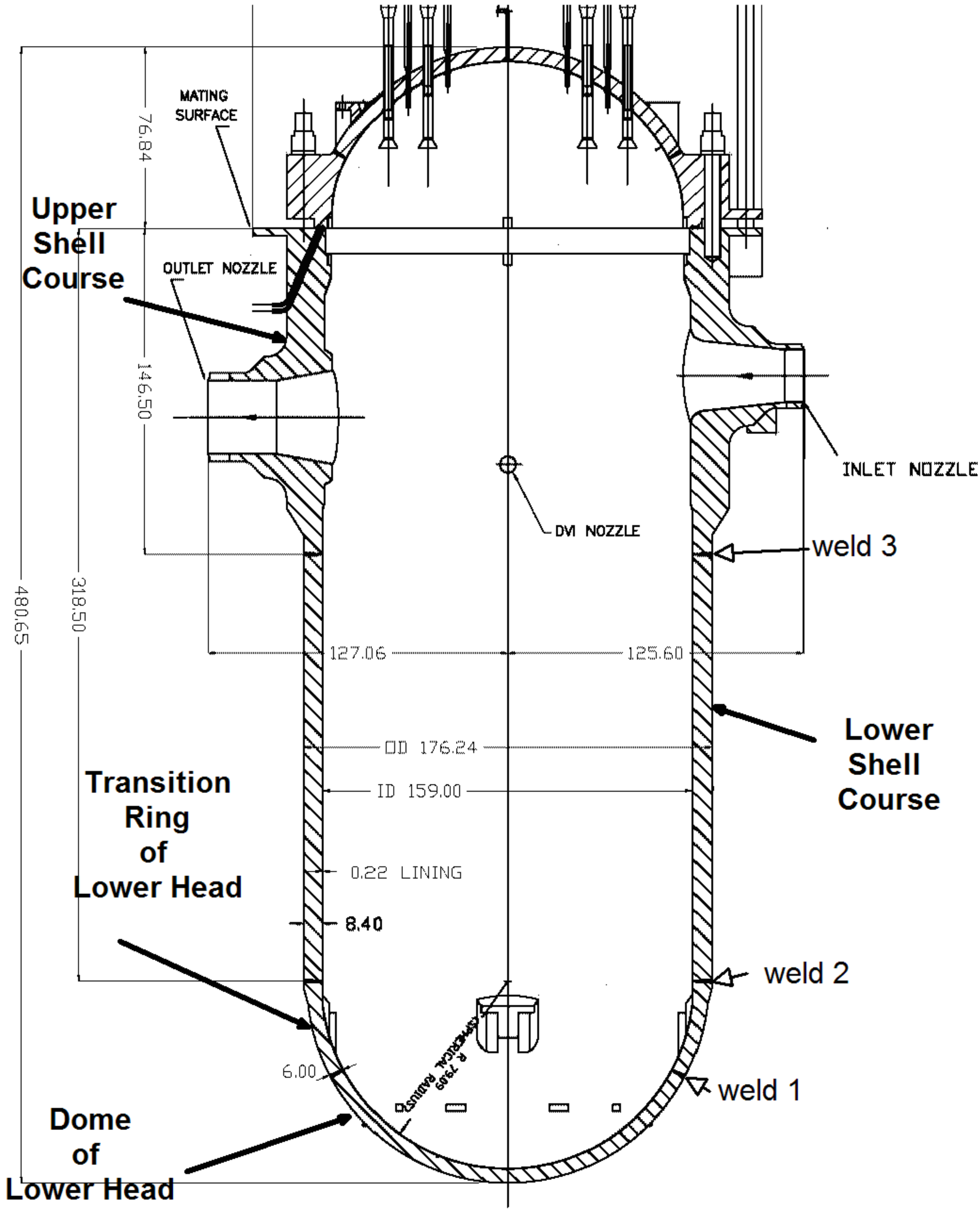


Figure 2.

Example of fracture toughness temperature transition curve indexed to RT_{NDT}. This example is for K_{Ic}, other measures of toughness exist.

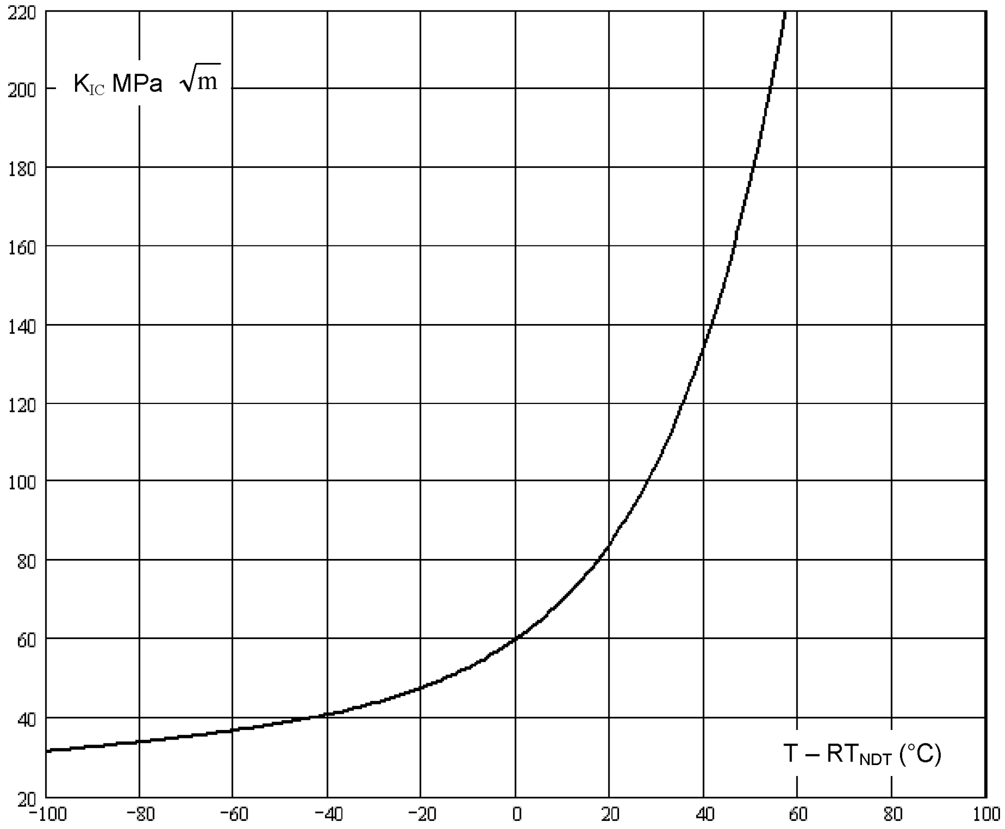
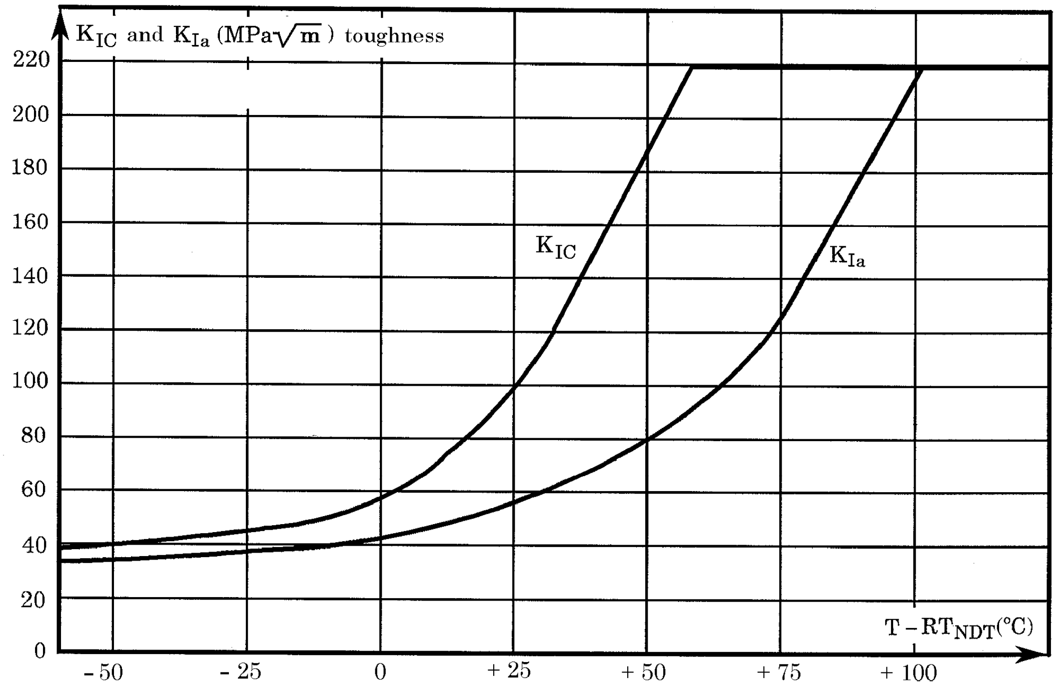


Figure 3

K_{Ic} and K_{Ia} fracture toughness temperature transition curves indexed to RT_{NDT}



Annex 1 – AP1000 Structural Integrity – Status of Regulatory Issues and Observations

RI / RO Identifier	Date Raised	Title	Status	Required timescale (GDA Step 4 / Phase 2)
Regulatory Issues				
None				
Regulatory Observations				
RO-AP1000-18	draft 31/10/08 final 12/3/09	Categorisation of Safety Function, Classification of Structures, Systems and Components	A programme of work has been offered to address the matters raised in this RO (see section 5.5 of report). Programme of work will need to be monitored and output assessed by HSE / ND.	Programme of work GDA Step 4
RO-AP1000-19	draft 31/10/08 final 12/3/09	Avoidance of Fracture - Margins Based on Size of Crack-Like Defects Integration of Material Toughness Properties, Non-Destructive Examinations During Manufacture and Analyses for Limiting Sizes of Crack-Like Defects	A programme of work has been offered to address the matters raised in this RO (see section 5.6 of report). Programme of work will need to be monitored and output assessed by HSE / ND. Full implementation may extend beyond GDA Step 4 and into Phase 2.	Programme of work GDA Step 4 and possibly Phase 2
RO-AP1000-20	draft 31/10/08 final 12/3/09	Manufacturing Method for Reactor Coolant Pump Casings	Substantial progress has been made with important parts of this RO. Remaining matters are associated with integrity of any large repair welds in castings. Propose RO-AP1000-20 be closed and new, more focussed RO opened to deal with remaining matters. (See Section 5.7 of report)	Residual matters in GDA Step 4. Propose via new RO
RO-AP1000-21	draft 31/10/08 final 12/3/09	Materials Specifications and Selection of Material Grade - Reactor Pressure Vessel, Pressuriser, Steam Generator Shells	Some progress has been made on matters raised in this RO. As a result of advice received, there are a number of details of material selection still to consider (see section 5.8 of report). Propose RO-AP1000-21 be closed and a new, more focussed RO be opened to deal with the remaining matters.	Residual matters in GDA Step 4. Propose via new RO

RI / RO Identifier	Date Raised	Title	Status	Required timescale (GDA Step 4 / Phase 2)
RO-AP1000-22	draft 31/10/08 final 12/3/09	Reactor Pressure Vessel Body Cylindrical Shell Forging Material and Associated Circumferential Welds Effects of Irradiation	Remaining matters for this RO mainly concern whether neutron fluence to Reactor Pressure Vessel has been reduced as low as reasonably practicable, given options still available for reduction. Propose RO-AP1000-22 be closed and new, more focussed RO be opened to deal with remaining matters. One option for dose reduction involves core management strategy which might be a licensee issue rather than a vendor issue. So this could extend into Phase 2. (See section 5.9 of report).	Residual matters in GDA Step 4 and possibly into Phase 2
RO-AP1000-23	draft 31/10/08 final 12/3/09	Primary Circuit Vessel Nozzle to Safe End Welds	Matters raised under this RO have been dealt with. RO-AP1000-23 can be closed. There may be HSE / ND assessment activity on this matter in GDA Step 4 but this activity might well be done without raising an RO. (See Section 5.10 of report).	
RO-AP1000-24	draft 31/10/08 final 12/3/09	Information on Reactor Internals	Matters raised under this RO have been dealt with. RO-AP1000-24 can be closed. There might be consideration of some internals under the programmes of work mentioned against RO-AP1000-18 and RO-AP1000-19, but that remains to be determined. (See Section 5.11 of report).	
RO-AP1000-25	draft 31/10/08 final 12/3/09	ASME Design Specifications and Design Reports - Current Status and As-Built Status	Matters raised under this RO have been dealt with. RO-AP1000-25 can be closed. This RO has provided HSE / ND with a better understanding of document structure for relevant components. This will provide a basis for sampling selection of documents in GDA Step 4. This may not require a new RO. (See Section 5.12 of report).	

RI / RO Identifier	Date Raised	Title	Status	Required timescale (GDA Step 4 / Phase 2)
RO-AP1000-26	draft 31/10/08 final 12/3/09	Fatigue Crack Initiation - Conservatism in ASME III Appendix I S-N Curves for Stainless Steel Material	Part of the matters raised under this RO have been dealt with. One specific aspect remains open - fatigue usage factors for the Pressuriser surge line. RO-AP1000-26 remains open until this last matter is dealt with. (See Section 5.13 of report).	GDA Step 4
RO-AP1000-27	draft 31/10/08 final 12/3/09	Reactor Internals - Testing and Inspection Programme of First AP1000 as "Prototype"	The matters raised under this RO have been dealt with. RO-AP1000-27 can be closed. (See Section 5.14 of report).	
RO-AP1000-28	draft 31/10/08 final 12/3/09	Pressuriser Surge Line Stratification Evaluation - AP-1000 First Plant Only Test	The matters raised under this RO have been dealt with. RO-AP1000-28 can be closed. (See Section 5.14 of report).	
RO-AP1000-29	draft 31/10/08 final 12/3/09	Reactor Pressure Vessel and Primary Circuit Pressure - Temperature Limits and Low Temperature Overpressure Protection	From this RO the basis for the Pressure-Temperature limits is clear. Whether the P-T limits provide an 'as low as reasonably practicable' basis is still an open matter. The way to take matters forward here might be for HSE / ND to consider its position, in light of information available. Such consideration would probably not need an RO, though it might eventually lead to a new RO. RO-AP1000-29 can be closed, on the understanding that HSE / ND might wish to consider this topic further, and that might result in a new RO. (See Section 5.15 of report).	
RO-AP1000-30	draft 5/3/09 final 12/3/09	Containment Pressure Shell ASME SA738 Grade B	The matters raised under this RO have been dealt with. However as a result of assessment of the response there are a number of matters to take forward. It might be that the follow-on matters can best be dealt with by establishing a new RO. (See Section 5.16 of report).	

ANNEX 2

Regulatory Observations

UK AP-1000

Generic Design Assessment - Step 3

Structural Integrity - Metal Components and Structures

NOTE

Due to page layout requirements for this Assessment Report the text of the Regulatory Observations in general is re-paginated compared to the original. However, the content of the Regulatory Observations is unchanged from the originals.

REGULATORY OBSERVATION

RO-AP1000-18

Categorisation of Safety Function, Classification of Structures, Systems and Components

Originated by / Organisation:	Approved by / Organisation:	Assessment Area	Date:
NII	NII	SI	12 March 2009

INTRODUCTION

Safety Function Categorisation and Classification of Structures, Systems and Components is an important, fundamental foundation in the development of a deterministic safety case. Together they are the basis for determining applicable codes and standards and other requirements applicable to Structures, Systems and Components.

SAPs ECS.1 and ECS.2 address categorisation of safety function and classification of structures, systems and components; SAP EKP.4 deals with identification of Safety Functions. For highest reliability metal components and structures (gross failure claimed so low it can be discounted), SAPs EMC.1 to EMC.3 apply, and SAPs ECS.3 and EMC.4 to EMC.34 are applicable with maximum stringency.

ASPECTS OF UK AP-1000, Safety, Security and Environmental Report (SSER) - UKP-GW-GL-700 Rev 2

The NII SAPs base safety classification of structures, systems and components on the prior categorisation of safety functions.

UK AP-1000 SSER, UKP-GW-GL-700 Rev 2 in Chapter 3.2 deals with classification of structures, systems and components. UK AP1000 SSER, Section 3.2.1 deals with seismic classification and section 3.2.2 deals with the general AP-1000 classification system.

The UK AP-1000 SSER does not appear to deal explicitly with safety function categorisation.

Three fundamental safety functions are defined in IAEA NS-R-1 (ref 1) Section 4.6:

- Control of reactivity
- Removal of residual heat
- Containment of radioactive fission products

The Annex to IAEA NS-R-1 lists 19 more specific safety functions (detailed sub-division of the three fundamental safety functions). For metal components and structures, IAEA NS-R-1 Annex safety functions 11 and 19 are particularly relevant (integrity of reactor coolant pressure boundary (11) and prevent failure or limit consequences of failure where the failure could cause impairment of a safety function (19)). So in general, a particular structure, system component could contribute to more than one safety function; and the classification of a SSC should take account of all applicable safety functions.

IAEA NS-R-1 Annex safety function 11 (integrity of reactor coolant pressure boundary) might be narrowly interpreted to mean avoidance of loss of coolant. However IAEA NS-R-1 safety function 19 (prevent failure or limit consequences of failure where the failure could cause impairment to a safety function) is more general. For a pressure boundary (or certain other components) all consequences of failure of the component should be considered for their potential to impair one or more of the three fundamental safety functions. In the context of pressure boundary components, IAEA NS-R-1 Annex safety function 19 is in effect an 'internal hazard' safety function.

For pressure boundary components, the internal hazard presented by gross failure go beyond 'loss of coolant' (i.e. direct potential reduction in capability to cool the fuel) and includes missile generation, blast, jet force, flood.

UK AP-1000 SSER in section 3.5.1.2.1.1 lists "missiles not considered to be credible". The arguments for missiles not being credible fall into either:

1. insufficient energy;

2. due to the design there is a negligible probability of a failure producing a missile.

A particular 'internal hazard' related to gross failure of a set of pressure boundary components is gross rupture of pipework. UK AP-1000 Sub-Chapter 3.6 deals with protection against the dynamic effects associated with the postulated rupture of piping. UK AP-1000 Sub-Chapter 3.6 states that the following is eliminated from the design basis:

> the dynamic effects of postulated breaks in the reactor coolant loop, main steam lines inside containment, other primary piping inside containment with nominal pipe size (NPS) equal to or greater than 6-inch (ND 150mm);

The basis for this elimination is "mechanistic pipe break (leak before break) considerations".

UK AP-1000 SSER in Sub-Chapter 3.6 notes that requirements for emergency core cooling and environmental qualification of equipment are not changed by the elimination of the dynamic effects of some pipe breaks.

In addition, UK AP-1000 SSER section 3.6.2.1.1 states that breaks are not postulated in piping in the vicinity of containment penetrations. Portions of piping within the vicinity of containment penetrations where breaks are not postulated are described as in the 'break exclusion area'. UK AP-1000 SSER section 3.6.2.1.1.4 sets out the requirements for piping in 'break exclusion area' and summarises the piping that falls within 'break exclusion areas'. It is noted that the requirements for piping to qualify as being in a 'break exclusion area' are not the same as the requirements for elimination of the dynamic effects of postulated breaks dealt with above (leak before break).

UK AP-1000 SSER section 5.4.1.1 states that for the main coolant pump flywheels:

> “The reactor coolant pump pressure boundary shields the balance of the reactor coolant pressure boundary from theoretical worst-case flywheel failures.”

And UK AP-1000 section 5.4.1.3.6.3 states:

> “The flywheel assemblies are located within and surrounded by the heavy walls of the stator closure, stator main flange, casing, thermal barrier, or lower stator flange. In the event of a postulated worst-case flywheel assembly failure, the surrounding structure can, by a large margin, contain the energy of the fragments without causing a rupture of the pressure boundary.”

The implication of this statement is that a flywheel could break into pieces, and the reactor coolant pump pressure boundary would contain the pieces.

The presumption for this Regulatory Observation is the UK AP-1000 SSER claims for the reactor coolant pump flywheel apply to both the upper and lower flywheel assemblies (UK AP-1000 SSER, Figure 5.4-1).

UK AP-1000 SSER section 3.9.5 deals with the design of reactor vessel internals. Notable functions of the reactor internals are

1. to provide protection, support and alignment of the core and control rods

2. to direct the coolant flow to and from the fuel assemblies.

Clearly these are closely related to the first and second fundamental safety functions of IAEA NS-R-1 i.e. control of reactivity and removal of residual heat. The question then arises whether for the core internals (particularly the lower internals), gross failure is postulated or not. UK AP-1000 SSER in section 3.9.5.1.1 states:

> “In the event of an abnormal downward vertical displacement of the internals following a hypothetical failure, energy-absorbing devices limit the dynamic force imposed on the reactor vessel. The energy absorbing device is the secondary core support. In addition, the secondary core support also transmits the vertical load of the core uniformly to the reactor vessel, limits the displacement to prevent withdrawal of the control rods from the core, and limits the displacement to prevent loss of alignment of the core with the upper core support to allow the control rods to be inserted into the reactor.”

The above implies that so far as control of reactivity is concerned, the consequences of gross failure of the lower internals (e.g. complete circumferential failure) are dealt with by limiting the downward displacement. There appears to be no explicit statement on the ability to direct coolant flow through the fuel assemblies to remove residual heat.

DISCUSSION

Collecting together information from different parts of the UK AP-1000 SSER, the following components, one way or another, have a claim of such high integrity against gross failure, that gross failure can be discounted from the safety case:

1. Reactor Pressure Vessel

2. Main Coolant Loop Pipework

3. Reactor Coolant Pump Bowl Casings

4. Pressuriser

5. Steam Generator Channel Head Shell, Tubesheet and Secondary Shell Pressure Boundary

6. Main Steam Lines Inside Containment

7. Accumulator Tanks

8. Core Makeup Tanks

9. Pipework in the vicinity of containment penetrations - pipework in ‘break exclusion areas’ (main steam lines and feedwater lines outside containment, Chemical and Volume Control System lines inside and outside containment).

It is not clear whether the reactor core internals should be included in this list or not; one potential consequence of gross failure (reactivity control) may be dealt with by design; it is not clear whether the other potential main potential consequence, inability to direct coolant flow through the fuel assemblies to remove residual heat, has been considered in the design.

It may be this list could have been constructed more directly, if the 19 safety functions in the Annex to IAEA NS-R-1 (in particular 11 and 19) had been used as a starting point. The topics of Categorisation of Safety Functions (identification of safety functions being the subject of SAP EKP.4) and Safety Classification of Structures, Systems and Components may be the subject of a separate, more general Regulatory Observation.

Is the above list of 9 components an accurate summary of the UK AP-1000 components where in effect gross failure is discounted on the basis of the integrity claimed for the components?

For the immediate purpose of the assessment of the 9 components listed above it is proposed to:

1. Apply a coherent approach to the integrity of those components where the claim is the likelihood of gross failure is so low it can be discounted - the components being the 9 items listed above (note ‘coherent’ does not imply exactly the same approach for every component);

2. If a claim is made that integrity of a component is such that the likelihood of gross failure is so low it can be discounted for some consequences but for other consequences the plant is designed to cope, then the assessment of the structural

integrity arguments will be on the same basis as if gross failure is discounted for all consequences of gross failure;

3. The assessment will be on the basis of testing against what might be termed a ‘no break’ criterion;

4. The assessment of structural integrity claims and arguments would then be within the framework of SAPs EMC.1 to EMC.3 with associated paragraphs 243 to 253 (and SAPs EMC.4 to EMC.34 applied within the framework set by EMC.1 to EMC.3).

5. Consider the classification of components and whether the corresponding design codes and other requirements are consistent with the apparent functional requirements.

In the case of pipework this question is put with the understanding of the range of consequences of failure that are evaluated for the pipework covered by “leak before break”. However as the range of pipe failure consequences considered in the SSER is not the full range for a gross guillotine failure, the assessment will be on the basis of the avoidance of the initiating event, not ameliorated by consideration of some consequences.

Example of Accumulator Tanks vs. Reactor Coolant Pump Flywheels

Accumulator tanks may be isolated from the primary circuit during normal operation. Gross failure of an accumulator tank would then not result directly in a loss of primary coolant. However there are other potential consequences of gross failure of an accumulator tank. These other potential consequences might be termed internal hazards, inside containment.

If, in effect, these other potential consequences of gross failure of an accumulator tank are discounted, the overall justification is based on the structural integrity of the accumulator tank pressure boundary. The assessment of the accumulator tank integrity claims and arguments would then be within the framework of SAPs EMC.1 to EMC.3 with associated paragraphs 243 to 253 (and SAPs EMC.4 to EMC.34 applied within the framework set by EMC.1 to EMC.3).

By comparison, UK AP-1000 SSER section 5.4.1.1 implies that gross disassembly of a reactor coolant pump flywheel would have limited consequences because the high energy fragments would be contained within the reactor coolant pump pressure boundary. If so, the overall safety claim and argument for the integrity of reactor coolant pump flywheels is a combination of the integrity of the flywheels and the ability of the pump casing to contain the fragments. In this situation, the assessment of the structural integrity of the flywheel can be within the framework of SAPs paragraphs 254 to 257, with SAPs EMC.4 to EMC.34 applied within that framework. The presumption for this Regulatory Observation is the UK AP-1000 SSER claims for the reactor coolant pump flywheel apply to both the upper and lower flywheel assemblies (UK AP-1000 SSER, Figure 5.4-1).

REFERENCES

1. *IAEA, Safety of Nuclear Power Plants: Design: Safety Requirements.* IAEA Safety Standards Series. NS-R-1. IAEA, Vienna (2000). ISBN 92-0-101900-9.

REGULATORY OBSERVATION

RO-AP1000-19

Avoidance of Fracture - Margins Based on Size of Crack-Like Defects

Integration of Material Toughness Properties, Non-Destructive Examinations During Manufacture and Analyses for Limiting Sizes of Crack-Like Defects

Originated by / Organisation:	Approved by / Organisation:	Assessment Area	Date:
NII	NII	SI	12 March 2009

Regulatory Observation RO-AP1000-18 lists 9 components where the likelihood of gross failure is in one way or another claimed to be so low it can be discounted. All components operate at temperatures sufficiently low for creep deformation not to be relevant. For the materials and components in question, there are two basic failure modes due to tensile stress:

1. plastic deformation, when the applied load exceeds the combination of material strength and wall thickness / shape, either by single load application or repeated loading causing incremental distortion;

2. propagation of a pre-existing crack-like defect in either a ‘brittle’ or ‘ductile’ mode.

Failure mode 1 above is well controlled by the traditional, long-established requirements of design codes.

Failure mode 2 above is unlikely but arguably is not as well controlled as mode 1 by design codes.

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

1. absence of crack-like defects at the end of the manufacturing process - confirmed by examinations during manufacture;

2. material toughness offering good resistance to propagation of crack-like defects - underpinned by minimum material toughness requirements in Equipment Specifications;

3. absence of in-service sub-critical crack growth mechanisms that could lead to the increase in the size of pre-existing defects; or in the extreme, nucleation and growth of defects from an essentially defect-free initial condition.

Usually the main locations of concern are welds, but some base material areas may also be relevant. The dominant in-service, sub-critical defect growth mechanism for the relevant PWR components is expected to be some form of fatigue.

A measure of the ‘margin’ implied by the above ‘defence-in-depth’ approach, and one based directly on defect size is:

DSM = ELLDS/(QEDS + LFCG)

where:

DSM - Defect Size Margin

ELLDS - End of Life Limiting Defect Size, is the size of defect which is calculated to give a fracture driving force equal to an end-of-life fracture toughness criterion. The fracture toughness criterion is intended to be a ‘lower bound’ to the true fracture toughness. Hence the term ‘limiting defect size’ is used rather than ‘critical defect size’, the latter implying actual failure;

QEDS - Qualified Examination Defect Size, is the defect size that can be detected, sized and characterised with high confidence. The claim for defect size would be supported by qualification of the examination. The extent of qualification depends on the difficulty and novelty of the examination;

LFCG - Lifetime Fatigue Crack Growth, is the calculated fatigue crack growth over the lifetime of the component, starting with an initial crack size equal to the Qualified Examination Defect Size (QEDS).

The examinations referred to above are those conducted during manufacture. The role of in-service examination is not considered here.

The basis for the DSM, is that if a defect of the QEDS size was in a component on entering service and grew by the LFCG amount by the end of life, the resulting defect would still not be capable of precipitating failure.

The approach in the UK has been to seek a target DSM of 2.

A margin based on defect size is preferred over, for instance, one based on load margin. Fracture of a component is caused by the presence of a crack.

This defect size margin approach requires manufacturing examinations capable of detecting and sizing crack-like defects of concern. 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; the margin being expressed as the Defect Size Margin (DSM).

In practice the dependence on manufacturing examinations usually means use of ultrasonic techniques. This approach may require ultrasonic examinations during manufacture that are not required by the applicable design/fabrication code or standard.

Crack-like defects are usually characterised by a depth (component through wall direction) and a length (along the component wall direction). For this deterministic approach, a representative crack shape aspect ratio is required. To cover a range of likely possibilities, 1:10 and 1:2 depth to length ratios might be chosen. In some locations, only 1:2 ratio defects might be plausible (e.g. cracks transverse to welds or at nozzle corners).

For this approach, there are some fundamental supporting requirements:

Materials Toughness: There needs to be a basis for a conservative (lower bound) value of fracture toughness for end of life conditions. In some cases (e.g. shells of Reactor Pressure Vessel, 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;

Qualification of Manufacturing Examinations: Ultrasonic examination is the predominant means of examination for crack-like defects. The European Network on Inspection Qualification (ENIQ) provides a framework for such qualification.

> As input to the qualification, a definition is required for the nature and size of defects to be found with high confidence. Usually, the qualification requirement will not be set at the theoretical smallest defect the technique can find. Instead the requirement is to set the qualification defect size less than the limiting defect size, by some margin.
>
> There should be consistency between defect aspect ratios included in the qualification, and those used in the fracture mechanics analyses for limiting defect sizes;

Limiting Defect Size Analyses: All relevant materials are ductile thus the analyses need to make use of elastic-plastic fracture mechanics methods.

All design basis load conditions need to be considered, from normal operation to fault (loads for which ASME III Service Levels A, B, C and D apply).

For analyses of loads for which Level A and B Limits apply, initiation fracture toughness is expected to be used. For analyses of loads for which Levels C and D Limits apply, fracture toughness based on a limited amount of stable tearing would be acceptable, so long as the level of toughness and stable tearing is supported by test data. This load/toughness combination balances likelihood of occurrence of the load with the margin on toughness to actual failure.

Whatever measure of fracture toughness is used, it should be representative of end of life conditions.

The fracture analyses should include primary and secondary stresses, including weld residual stresses.

All potential locations for crack-like defects should be included in the fracture mechanics analyses.

It is reasonable to use bounding analyses to limit the volume of analysis work. However care is needed in selecting bounding conditions. For example, an analysis for a load for which Level D limits apply that used stable tearing would not bound an analysis for a load for which Level A limits applied and where initiation toughness was used (the Level D load would bound the level A load, but the tearing toughness would exceed the initiation toughness and so would not be bounding).

Defect aspect ratios used should be consistent with those used for the qualification of examination. Surface breaking defects at either the inner or outer surfaces of components will usually give the highest crack driving force for a given set of conditions. In determining limiting defect conditions, the analyses should consider the crack front at the deepest through-wall position and at the surface points.

To implement the approach outlined above requires a number of enablers, as described below.

(1) Including a requirement in the Equipment Specification for material toughness using parameters directly usable in fracture mechanics analyses. Or some requirement based on other parameters that can be shown to support some claim of minimum toughness. An example of the sort of fracture toughness requirement in an Equipment Specification for items made from low alloy ferritic steel (e.g. Reactor pressure vessel, Steam Generator shells, Pressuriser) is given in Appendix 1 of this RO.

(2) Including in the manufacture of components a suitably redundant and diverse range of manufacturing examinations, most likely including ultrasonic examinations. The examinations would require qualification (see below). An example of the sort of manufacturing examination schedule for a component like a Reactor Pressure Vessel is given in Appendix 2. Much of this examination schedule is additional to standard code requirements. The additional steps would have to be specified in the Equipment Specification, including the qualification of examinations carried out by the manufacturer. Ultrasonic examinations during manufacture that are not Code requirements would need acceptance criteria to be defined by the Customer and included in the Equipment Specification. The acceptance criteria would be expected to be associated with defects smaller than the qualification defect sizes (e.g. see table 2 of ref 1, compared with qualification sizes below). Qualification defect sizes are detectable with high confidence. This means the techniques will be capable of detecting smaller defects, but with reduced likelihood at smaller defect sizes.

(3) Qualification of manufacturing ultrasonic examinations - procedures, equipment and personnel - using a framework such as that of the European Network on Inspection qualification (ENIQ). This would include Technical Justifications and, as appropriate, practical trials. Examples of qualification defect sizes that would be expected to be practically achievable are given in Appendix 3. In some cases, specific measures are required to facilitate examination. For example, for austenitic stainless steel, the use of sufficiently worked forged components. The latter produces a suitable grain structure for transmission of ultrasound.

The examination procedures for the required level of qualification may need to be specifically designed to meet the requirements, rather than taken straight from a code. Examples of manual ultrasonic examination procedures for ferritic and austenitic stainless steel pipe welds are given in Appendix 4.

Some austenitic stainless steel components may be difficult to examine using ultrasonic techniques, e.g. thick, cast components. Ultrasonic examination of the component walls within some distance below the surfaces could still be a contribution to integrity. However it is recognised that practically, the time required to examine large surface areas needs to be considered against the contribution to integrity.

(4) Fracture mechanics analyses to determine end of life limiting crack sizes. Some aspects of such analyses are covered above. From experience it is known that such analyses have produced end of life limiting defect sizes generally meeting the Defect Size Margin target of 2, when used with the sort of Qualified Examination Defect Sizes as summarised in Appendix 3. Obviously the value of fracture toughness used in such analyses has an important effect on the calculated limiting defect sizes. Some representative examples of fracture toughness are summarised in Appendix 5.

Ref 1. Whittle M J., Collier J G., The Design and Validation of Reactor Vessel Inspections. In Proceedings of the Second Birmingham Seminar. The Pressurised Water Reactor and the United Kingdom. 22-23 April 1985. Editors D R Weaver, J Walker. University of Birmingham.

APPENDIX 1

Example of Material Fracture Toughness Requirement in the Equipment Specification - Component Made from Low Alloy Ferritic Steel

Tests shall be carried out on specimens taken from forging material and from weld metal of each weld procedure qualification test.

Fracture toughness J-tests shall be carried out using standard compact tension (CT) J-specimens side-grooved to a depth of 10% each side. The test specimens shall be at least 25mm thick.

Test standard to be defined. Blunting line and exclusion lines to be defined.

Individual J-Δa data shall be reported.

Test shall be carried out at two temperatures:

T1 = (max RT_{NDT} + ΔT) (maxRT_{NDT} and ΔT to be defined)

T2 = Normal Operating Temperature

The following parameters shall be evaluated for each test:

J_{1c} - the value of J at the intersection of the blunting line and the linear regression line of data provided validity criteria are met (this is initiation toughness J)

K_{Jc} - the value of K computed from J_{1c} (initiation toughness K)

$J_{\Delta a2}$ - the value of j computed from the intersection of the data regression line at Δa=2mm

The materials shall meet the following minimum toughness requirements:

Test Temp >	**T1**		**T2**	
	Forging	Weld	Forging	Weld
J_{1c} (kJ/m^2)	160	160	140	140
K_{Jc} (MPa√m)	190	190	170	170
$J_{\Delta a2}$ (kJ/m^2)	700	500	400	250

At temperature T1, no specimen shall show cleavage instability at a K_J value less than 300 MPa√m.

Appendix 2

Schedule of Manufacturing Examinations
Example of Item such as
Reactor Pressure Vessel

NDT technique abbreviations:

MT = Magnetic Particle
PT = Dye Penetrant
RT = Radiography
UT = Ultrasonics

NDT Technique	Notes
UT	UT of forged parts prior to welding
	Ferritic butt welds completed. Initial heat treatment completed
MT	Inner and outer surfaces (standard code requirement)
RT	(standard code requirement)
UT	Qualified manual UT by manufacturer (not required by code)
	Back clad strip at main butt welds **Post weld heat treatment** **Surface preparation of clad**
UT for UCC	Qualified manual UT from inner surface, to detect underclad cracking. No-standard examination. Might be omitted on basis of arguments that fabrication route optimised to avoid underclad cracking.
PT	Test of back cladding strip. (standard code requirement)
UT	Qualified manual UT of ferritic welds through cladding, form inner surface only. Examination by manufacturer. Arguably only to limit commercial risk. Might be omitted. (not required by code)
UT	Manual UT examination of cladding to check for bonding at clad to ferritic base interface.
	Final Stress Relief
UT	Qualified manual UT from both surfaces. Examination by manufacturer (not required by code)
	Hydrotest
PT	All clad internal surfaces (not required by code)
MT	Examination of all external surfaces (not required by code0
Automated UT	Qualified automatic UT of welds conducted in fabrication shop. Examination from both surfaces. Examinations conducted by Agent of Customer. Intent is diversity from manufacturer’s examinations (not required by code)
PSI	Base-line qualified UT examination representative of in-service examination (code requirement)

Appendix 3

Ultrasonic Examination Defect Sizes
(High Confidence of Detection and Sizing)

Component / Location	Defect Size (mm)* Through Wall Extent x Length
RPV - low alloy ferritic steel	
Shell weld	P=25x250 T=25x50
Main Nozzle Bore	10x100
Main Nozzle Inner Corner	10x20
Nozzle Weld	P=25x250 T=25x50
Steam Generator - low alloy ferritic steel	
Shell weld	P=15x150 T=15x30
Nozzle Inner Radius	10x20
Main and Auxiliary Feedwater Nozzle welds	15x30
Main Feedwater Nozzle Bore	10x100
Pressuriser - low alloy ferritic steel	
Shell weld	P=15x150 T=15x30
Nozzle Inner Surface	10x20
Primary Coolant Loop Pipework -	
forged austenitic stainless steel (int. dia. circa 750mm, wall thickness circa 80mm)	
Narrow gap TIG welds	P=10x100 T=10x20 Ultrasound path through forged material to inspect narrow gap TIG welds. Similar capability for safe end welds
Flywheel - plate ferritic steel	
plate	10 x full depth of flywheel plate
Pressure Vessel - ferritic steel	
dia. circa 4300mm, wall thickness circa 75mm	
Shell weld	P=15x150 T=15x30
Nozzle inner radius	10x20
Main Steam Line - ferritic steel	
int. dia circa 660mm, wall thickness circa 45mm	
Pipe circumferential welds	P=10x100 T=10x20
Nozzle inner radius (nom. dia. 300mm)	10x20

* crack orientation
P= parallel to weld
T= transverse to weld

Appendix 4

Examples of Manual Ultrasonic Examinations
1. Ferritic Pipe Circumferential Welds

Diameter / Wall thickness (mm) and conditions	Qualification Crack Size Depth x length (mm)	Summary of Ultrasonic Examination Procedure - For Guidance Only
Pipe dia. >80mm wall thickness 12-100mm Extent of surface preparation for butt welds, 4T either side of weld along whole length of weld (T=wall thickness)	<u>Back wall and mid-wall:</u> about 3x6mm whether weld prepared or not <u>Near surface:</u> (i.e. probe surface) *prepared weld:* 3x6mm *As-welded:* not detectable Note: Back wall 3mm depth takes account of weld root ultrasonic response	Examination from outside surface (the 'near surface'). Scan for longitudinal and transverse defects. UT probes 10mm dia., 38^o, 45^o, 60^o 4/5MHz* single shear wave, 70^o single and double shear wave. * where attenuation losses exceed 6dB, 2/2.5MHz probes can be used. <u>Longitudinal Defects:</u> Scans using at least 2 (up to 40mm thick) or 3 (over 40mm thick) probe angles from: 45^o, 60^o, 70^o. One angle selected so beam strikes main fusion face as near to normal incidence as possible. <u>Transverse Defects:</u> Scan using at least 2 probe angles from 30^o, 45^o, 60^o, 70^o. Sensitivity (using 3mm dia. side-drilled holes): Longitudinal defect scans: DAC + 14dB + ΔdB Transverse defect scans: DAC + 20dB + ΔdB DAC = Distance Amplitude Correction ΔdB = Correction for attenuation Sizing using methods in BS3923 Part 1

2. Austenitic Stainless Steel Pipe Circumferential Welds

Diameter / Wall thickness (mm) and conditions	Qualification Crack Size Depth x length (mm)	Summary of Ultrasonic Examination Procedure - For Guidance Only
Normal attenuation material Forged austenitic stainless steel joined by narrow gap TIG welds	10x100	Longitudinal Defects: Twin angled compression probes (2MHz 60^o and 70^o) and a 2MHz 60^o shear wave probe scanned axially, in both directions. Transverse Defects: Twin angled compression probes (2MHz 45^o and 60^o) and 2MHz 50^o skewed shear wave probes scanned circumferentially in both directions. Sensitivity compensated for material attenuation Recording thresholds 20% to 30% DAC using 3mm dia. side drilled holes. Sizing using methods in BS3923 Part 1.
highly attenuating material		As above, for normal attenuation material except: Longitudinal Defects: Twin angled compression probes (1.5MHz 60^o and 70^o) and a 1.5MHz 60^o shear wave probe scanned axially, in both directions. Transverse Defects: Twin angled compression probes (2MHz 45^o and 60^o) and 1.5MHz 50^o skewed shear wave probes scanned circumferentially in both directions.

Appendix 5

Examples of Fracture Toughness Levels used in Fracture Mechanics Analyses

Low alloy steel vessel weld (e.g. RPV, SG, Pressuriser)

Upper shelf toughness at normal operating temperature:

initiation toughness: 160 MPa√m

toughness after 2mm stable tearing: 220MPa√m

Other ferritic steel components, base material and welds

Upper shelf toughness:

initiation toughness - generic lower bound: 110MPa√m

toughness after 2mm stable tearing - generic bound: 170 MPa√m

Austenitic stainless steel pipe welds

MMA (including safe end welds)

initiation toughness 80MPa√m

toughness after 2mm stable tearing 160 MPa√m

TIG

initiation toughness 120MPa√m

toughness after 2mm stable tearing 275MPa√m

REGULATORY OBSERVATION

RO-AP1000-20

Manufacturing Method for Reactor Coolant Pump Casings

Originated by / Organisation:	Approved by / Organisation:	Assessment Area	Date:
NII	NII	SI	12 March 2009

UK AP-1000 SSER (UKP-GW-GL-700 Rev 2) in Table 5.2-1 (Sheet 3 of 6) indicates the Reactor Coolant Pump (RCP) Casings are to be manufactured as single castings using austenitic-ferritic stainless steel - ASME SA-351 CF8A. This was confirmed in the answer to GDA Step 2 Technical Query AP1000-000024 (raised 6 May 2008).

We note the molybdenum level of CF8A (0.5% maximum) compared with CF8M (2.0-3.0%).

We note that SA 351 CF3A has similar chemical content to CF8A, except for a lower carbon content; CF3A and CF8A have the same mechanical strength requirements (UKP-GW-GL-710 Section F Chapter 5 (rev 2), section 5.2.3.2 refers to CF3A). Why has CF8A been chosen rather than CF3A?

ASME SA 351/SA-351M references A 703/A 703M. SA 351 and SA 703 contain a number of potential supplementary requirements, notably:

S2 (SA 351 & SA 703) - destruction tests (representative castings selected from a heat and sectioned for examination for internal defects);

S5 (SA 351 & SA 703) - radiographic inspection

S6 (SA 351 & SA 703) - liquid penetrant inspection;

S7 (SA 703 only) - ultrasonic inspection;

S8 (A 703 only) - Charpy Impact Test;

S11 (SA 351 only) - post weld heat treatment (castings subjected to weld repairs given a post weld solution heat treatment);

S12 (SA 703 only) - prior approval of major weld repairs;

S20 (SA 703 only) - weld repair charts;

S24 (SA 703 only) - specified ferrite content range.

SA 703 Supplementary Requirement S 24 requires the minimum specified ferrite content to be 10%, but no lower than required to achieve the minimum mechanical properties. We note the necessity for the ferrite content to achieve the minimum mechanical properties, but the maximum ferrite content should not exceed 20%.

From ASME III Subsection NB, NB-2570 and Table NB-2571-1, volumetric examination of a thick-wall cast stainless steel pump casing is likely to only be by radiography.

With regard to ‘severity levels’ applied to radiographic examinations, the Steel Founder’s Society of America web site contains the following comment with regard to ASTM inspection standard usage:

> “It should be borne in mind at all times that the severity rating is strictly arbitrary and based on little more than opinion. None of the reference radiographs are based on any kind of test data, and the severity levels are not graded to any basis of acceptability as to service performance. They only serve as a reference point in communicating the purchasers' requirements.”
>
> [http://www.sfsa.org/sfsa/buyrord3.html#spf8.4]

Large castings such as those for the RCP casings may exhibit concentrations of defects at about the mid-point of the wall thickness. This implies at least some repair welds with through thickness extent around half wall thickness. Repair welds are likely to be in the as-welded condition. Thus any potential crack-like defects within the volume of a repair weld would be subjected to as-welded residual stresses, in addition to applied loading stresses. And the fracture toughness of the as-deposited weld is likely to be lower than that of the parent casting material.

Experience of analysing the integrity of postulated crack-like defects in large repair welds in cast austenitic stainless steel RCP casings is illustrated in ref 1. The analyses were based on surface breaking defects being of most concern, and qualified surface and ultrasonic examination procedures applied to the pump casings before and after repair. The ultrasonic examinations were only intended to detect defects within 25mm of the component surfaces (ref 2).

Ultrasonic examination of the Reactor Coolant Pump Casings production parts is not a requirement of ASME III. This is possibly because:

1. the microstructure of the casting material means that ultrasound will not penetrate far into the component from either inside or outside surfaces;

2. ultrasonic examination from outside and inside surfaces would require additional surface preparation;

3. ultrasonic examination from outside and inside surfaces would take a considerable length of time.

Item 1 above is a fundamental limitation, the issues are covered in ref 3; items 2 and 3 are mainly economic factors.

Casting as a manufacturing route for RCP pump casings is almost universal. However, in the past an integral forged pump casing design has been used, though in a low alloy ferritic material, rather than austenitic-ferritic stainless steel (ref 4). It is noted a large initial ingot is required, with the finished forged casing being only about 25% of the mass of the initial ingot.

DISCUSSION

From the above, the following questions arise:

1. What is the approximate mass of each pump casing?

2. Has a study been conducted of manufacturing options for the Reactor Coolant Pump Casings, including assessment of the relative strengths and weaknesses of forging versus casting? If such a study was done some time ago, has there been any subsequent review, taking account of developments in technology?

3. Assuming an austenitic-ferritic stainless steel casting:

   Why has CF8A been specified rather than another grade, for example CF3A?

   How is the solution heat treatment controlled in order to give a material condition with suitable ferrite content?

   What Supplementary Requirements in SA 351 and SA 703 are included in the Equipment Specification? Justify exclusion of any of the following Supplementary Requirements: S2, S8, S12, S20, S24 (see list above).

4. What practice is used for initiating a manufacturing programme? For example is the first item produced subjected to destructive examination?

5. For the cast austenitic-ferritic RCP casings, has the radiographic examination procedure been qualified for its capability to detect crack-like defects lying perpendicular to main stress components in the wall of the casing? Has the radiographic examination procedure been checked against predicted and actual defects using destructive examination on components with representative material and wall thickness?

6. Ultrasonic examination of austenitic stainless steel castings can be difficult due to the effects on ultrasound transmission. But in some circumstances it has been shown to be possible, at least to depths up to 25 to 50mm below the surface (refs 2, 3). Would ultrasonic examination at least of near surface regions provide an additional element to the evidence for integrity of the component? If scanning the complete outer and inner surfaces was considered not economic for the benefit, would at least ultrasonic examination of large repair welds be reasonable?

7. Is there an Equipment Specification for the Reactor Coolant Pump Casings and does this include extra and / or more specific requirements compared with ASME III?

References

1. Bouchard P J., Goldthorpe M R., Prottey P., J-Integral and Local Damage Fracture Analyses for a Pump Casing Containing Large Weld Repairs. International Journal of Pressure Vessels and Piping Vol 78 pp295-305 (2001).

2. Hook D E., Booler R V., Validated Ultrasonic Techniques for Austenitic Welds and Castings. Proc. 10th Int. Conf. on NDE in the Nuclear and Pressure Vessel Industries 1990.

3. Boveyron C., Villard D., Boudot R., Ultrasonic Testing of cast Stainless Steel Components. pp 403-408, Proc. 11th Int. Conf. on NDE in the Nuclear and Pressure Vessel Industries. Albuquerque (30 April-2 may 1992).

4. Austel W., Körbe H., Integral Forged Pump Casing for the Primary Coolant Circuit of a Nuclear Reactor: Development in Design, Forging Technology, and Material. pp 285-397. Steel Forgings. A Symposium. ASTM STP 903. Edited by E G Nisbett, A S Melilli. Williamsburg (28-30 November 1994).

REGULATORY OBSERVATION

RO-AP1000-21

Materials Specifications and Selection of Material Grade -

Reactor Pressure Vessel, Pressuriser, Steam Generator Shells

Originated by / Organisation:	Approved by / Organisation:	Assessment Area	Date:
NII	NII	SI	12 March 2009

This Regulatory Observation addresses materials specifications and selection of materials for the following major pressure vessels (materials defined in UK AP-1000 SSER (UK-GW-GL-700, rev 2), Table 5.2-1):

Reactor Pressure Vessel (forgings SA 508 Grade 3 Class 1)

Pressuriser (forgings SA 508 Grade 3 Class 2)

Steam Generator Shells (primary and secondary circuit sides) (forgings SA 508 Class 1A or Grade 3 Class 2);

Steam Generator Channel Head (forgings SA508 Grade 3 Class 2).

In line with international practice for PWRs, the above vessels in the UK AP-1000 are specified to be made using a quenched and tempered low-alloy ferritic steel. The AP-1000 specifies forgings as the material of construction.

[*As an aside, UKP-GW-Gl-700 (page 5.4-10) states the primary and secondary sides of the steam generator pressure boundaries are designed to ASME III NB (Class 1) even though the secondary side could be designed to ASME III NC (Class 2). But for ASME XI in-service inspection, the steam generator secondary side shells are treated as ASME III Class 2 components. This was confirmed in the reply to TQ AP1000-000020).]*

Appendix 1 to this Regulatory Observation makes comparisons between the following forging materials:

Reactor Pressure Vessel (Tables 1 and 3, Table 2 covers welds)

ASME A508 Grade 3 Class1 (previously called A508 Class 3);

UK specification of material, based on A508 Class 3;

Steam Generators and Pressuriser (Tables 4 and 5)

ASME SA508 Grade 3 Class 2 and Class 1A;

UK specification of material also used A508 Class 3 for these components;

For the RPV, Table 1 compares chemical compositions and Table 3 compares heat treatment, mechanical tensile properties, design stress and fracture properties requirements.

For the Steam Generators and the Pressuriser, Table 4 compares chemical compositions and Table 5 compares heat treatment, mechanical tensile properties, design stress and fracture properties requirements.

COMPARISONS FOR REACTOR PRESSURE VESSEL

The following are notable highlights of the comparison:

Chemical Composition

1. UK usage of ASME A508 Class 3 has additional and in some cases more restrictive chemical composition requirements compared to ASME A508 Grade 3 Class 1;

2. UKP-GW-GL-700 Rev 2 for AP-1000 has limits on Nickel, Sulphur and Phosphorus similar to UK usage of SA 508 Class 3 (now Grade 3 Class 1). The AP-1000 limit for Copper is slightly lower than the UK usage specification (assuming 0.06% max for AP-1000). The Vanadium limit for UK usage is somewhat lower than the AP-1000 limit. In practice actual materials for UK usage construction were within these limits, for instance it is understood Sulphur content for RPV forgings was approximately 0.003%.

3. ASME A508 Grade 3 Class 1 chemical composition includes limits on Boron, Columbium (Niobium), Calcium and Titanium. These have been introduced since the ASME code version for UK usage.

Charpy Impact Energy

1. UK usage of ASME A508 Class 3 is the same as ASME but with additional ‘upper shelf temperature’ requirement. The latter is similar to the AP-1000 requirement (which is derived from 10 CFR 50 Appendix G).

RT_{NDT}

1. ASME III NB-2300 defines the method for determining RT_{NDT} but does not specify required values;

2. UK usage of ASME A508 Class 3 specifies start of life RT_{NDT} of less than -12°C for all RPV forgings except the nozzles and -22°C for the nozzles. AP-1000 specifies beltline forging and weld RT_{NDT} as -23°C and -29°C respectively, with +12°C elsewhere;

3. Apart from shift in RT_{NDT} through life due to neutron irradiation, it has been UK precedent to include the potential shift in RT_{NDT} due to thermal and strain ageing. A representative, claimed conservative, value of RT_{NDT} shift due to thermal and strain ageing is 30°C (covering weld and base metal).

COMPARISONS FOR STEAM GENERATORS AND PRESSURISER

Chemical Composition

1. The UK usage has limits on Antimony, Arsenic, and Tin (to reduce the potential for temper embrittlement). Also the UK usage specification has a lower limit on Chromium (0.15% max vs. 0.25% max). The UK usage limit on Carbon (0.2% or 0.22%) is somewhat lower than the standard specification (0.25% max). The UK usage specification does not include a limit on Aluminium (it does for the RPV material specification);

Tensile Properties & Design Stress

1. The UK usage of ASME A508 Class 3 means the tensile strength requirements are those of ASME A508 Grade 3 Class 1;

2. Related to the lower tensile properties values, the UK usage of A508 Class 3 material has a lower design stress (S_m = 184MPa) than for the AP-1000 material specification.

Charpy Impact Energy

1. UK usage of A508 Class 3 includes a target for ‘upper shelf temperature’ Charpy impact.

RT_{NDT}

1. The UK usage of A508 Class 3 specifies RT_{NDT} less than -12^{o}C;

2. For components not affected by neutron irradiation, UK precedent has been to include the potential of thermal and strain ageing to increase the RT_{NDT} over the life of components. A claimed conservative assumption is to take the shift in RT_{NDT} (i.e. ΔRT_{NDT}) due to thermal and strain ageing to be 30^{o}C (covering weld and base metal).

FRACTURE MECHANICS BASED MATERIAL SUPPLY REQUIREMENTS

The UK usage of A508 Class 3 for the Reactor Pressure Vessel included in the Equipment Specification the requirement to show material properties meeting minimum values in terms of ‘J-resistance curve’ fracture toughness parameter (see Appendix 2).

The UK usage of A508 Class 3 for the Steam Generators and Pressuriser also included a requirement based on the ‘J-resistance curve’ fracture toughness parameter, at their respective design temperatures of 300^{o}C and 345^{o}C. This included a requirement to achieve a minimum upper shelf initiation toughness of 165MPa$\sqrt{}$m.

SUMMARY OF REASONS FOR ADDITIONAL CHEMICAL COMPOSITION RESTRICTIONS FOR UK USAGE OF ASME SA508 CLASS 3

The lower limit on carbon is to improve weldability and give increased ductility.

Specification of low Copper, Phosphorus and Vanadium content is to control deterioration of properties due to irradiation. In addition the maximum level of Nickel is reduced for the same reason.

The limits on impurity elements are to achieve:

steel cleanliness giving better weldability;

general improvement in toughness;

reduced tendency to weld reheat cracking;

avoiding thermal ageing of the Heat Affected Zone of welds at around 300^{o}C;

avoiding temper embrittlement and minimising strain ageing.

Limitation on Sulphur content is useful in minimising susceptibility to Environmentally Assisted Cracking (EAC). The UK usage of SA508 specifies a maximum Sulphur content of 0.008% (actual achieved for RPV circa 0.003%), while the AP-1000 requirement includes a maximum limit on Sulphur of 0.01%. ASME XI Code Case 643-2 (ref 1) states that material with low Sulphur content (<0.004%) is not susceptible to EAC. For material meeting this Sulphur content limit, the fatigue crack growth law is of simple form. This applies to PWR primary water environments. Even with low Sulphur content, SA508 and similar materials can exhibit EAC and enhanced fatigue crack growth rates in secondary-side water conditions (ref 2).

DISCUSSION

From the foregoing the following questions arise.

1. Why does the UK AP-1000 SSER Table 5.2-1 for the steam generators specify for forging material both SA508 Class 1A and Grade 3 Class 2? Are these two materials alternatives to be chosen in any particular design, or are they used in different locations in the steam generators?

2. UKP-GW-GL-700 Rev 2 Chapter 5 Table 5.3-1 indicates that for the RPV beltline forging, the limit on Copper content is 0.06% max. However UKP-GW-GL-710 rev 2 on page 5-50 indicates the beltline forging and weld metal will have a limit on Copper of 0.03% max. Which limit is correct for Copper?

3. Is there any substantive reason why the Equipment Specification for the UK AP-1000 Reactor Pressure Vessel materials could not include limits on the chemical composition as in the UK usage of ASME SA508 Class 3 (now Grade 3 Class 1) (notably Carbon, Chromium, Arsenic, Antimony, Tin and Hydrogen)? Also could the Sulphur limit be specified as <0.004%?

4. UK precedent has been to use the UK modified specification of ASME SA508 Class 3 (now Grade 3 Class 1) for the Reactor Pressure Vessel, Pressuriser and Steam Generator (primary and secondary side) shells. For AP-1000, could all these pressure vessel shells be specified to be constructed from ASME Grade 3 Class 1?

5. AP-1000 documentation refers to the beltline forging and the ‘beltline weld’ or lower girth weld (e.g. UKP-GW-GL-700 Rev 2 Table 5.3-3 Note 2 and UKP-GW-GL-710 Rev 2 pages

5-48, 5-50). Does reference to this weld mean the weld between the bottom of the shell course and the top of the lower head? And specifically, is there no circumferential weld at the beltline mid-height (see Figure 1)?

REFERENCES

1. Case N-642-2. Fatigue Crack Growth rate Curves for Ferritic Steels in PWR Water Environment. Section XI Division 1. Cases of ASME Boiler and Pressure Vessel Code, 2007 Edition.

2. Eason E D., Nelson E E., Heys G B., Fatigue Crack Growth Rate of Medium and Low Sulfur Ferritic Steels in Pressurized Water Reactor Primary Water Environments. Trans ASME, Journal of Pressure vessel Technology, Vol 125 pp 385-392 (November 2003).

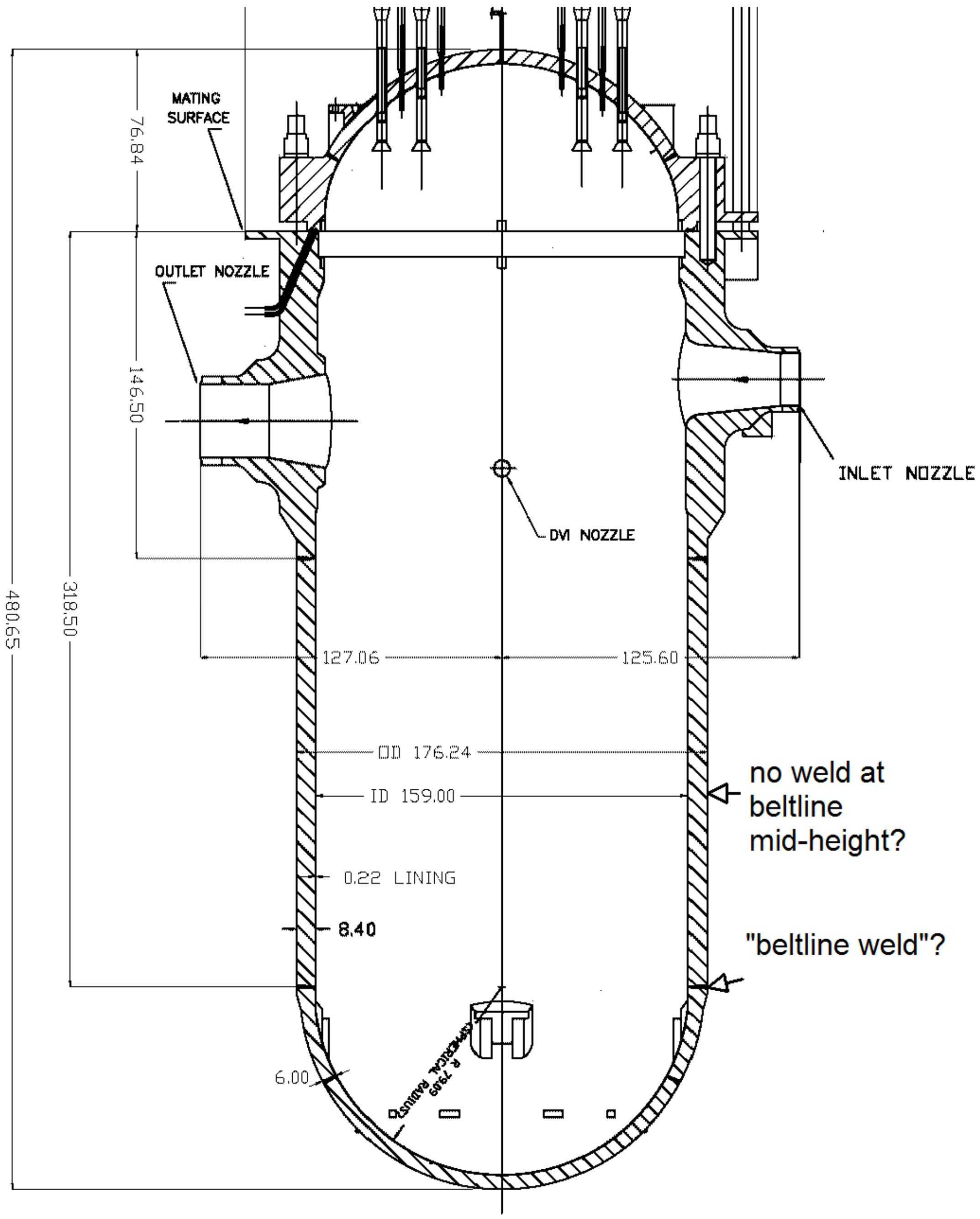


Figure 1 AP-1000 RPV side elevation

APPENDIX 1

Materials - Comparison Tables

TABLE 1 Reactor Pressure Vessel Materials

	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
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%	0.85% max
Sulphur	0.025% max	0.008% max	0.01% max
Phosphorus	0.025% max	0.008% max	0.01% max
Silicon [3]	0.4% max	0.3% max	
Chromium	0.25% max	0.15% max	
Copper	0.2% max	0.08% max	0.06% max[5]
Vanadium	0.05% max	0.01% max	0.05% max
Antimony	-	0.008% max	
Arsenic	-	0.015% max	
Cobalt	-	0.02% max	
Tin	-	0.01% max	
Aluminium	0.025% max [2]	0.045% max	
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 and electric furnace and vacuum-degassed.

Notes to Table 1

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.

5. UKP-GW-GL-710 Rev 2 on page 5-50 states for beltline forging material, limit on copper will be 0.03% max.

TABLE 2 For information, the corresponding weld metal chemical composition is given in the table below.

	UK Usage Weld Metal (as deposited)	**UK AP-1000 SSER Chapter 5 Section 5.3 Table 5.3-1 Weld Metal (as deposited)[1]**
Carbon	0.15% max	
Manganese	0.8 to 1.8%	
Molybdenum	0.35 to 0.65%	
Nickel	0.85% max	0.85% max
Sulphur	0.01% max	0.01% max
Phosphorus	0.01% max	0.01% max
Silicon	0.15 to 0.6%	
Chromium	0.15% max	
Copper	0.07% max	0.06% max[2]
Vanadium	0.01% max	0.05% max
Antimony	0.008% max	
Arsenic	0.015% max	
Cobalt	0.02% max	
Tin	0.01% max	
Aluminium		
Hydrogen		

Notes to Table 2

1. Element composition assumed the same as ASME standard composition unless specific different limit stated.

2. UKP-GW-GL-710 Rev 2 on page 5-50 states for beltline forging material, limit on copper will be 0.03% max.

TABLE 3 Reactor Pressure Vessel - Important Material Parameters

	ASME standard composition SA508 Grade 3 Class 1 (formerly SA508 Class 3)	**UK Usage of SA508 Class 3 (now called SA508 Grade 3 Class 1) RPV[5] additions/changes**
Austenitising Temperature	"to produce an austenitic structure"[6]	
Tempering Temperature	min 650^{o}C (4.2.2) min 635^{o}C (S13)	
Simulated Stress Relief Heat Treatment	less than 620^{o}C (S13)	
Tensile Properties		
Room Temp	min yield S_y = 345Mpa UTS S_u = 550-725Mpa min A% = 18%	

Design Stress		
Room Temp	184Mpa	
300°C	184Mpa	
350°C	184Mpa	
Charpy Impact Energy		
4.4°C	min average 41J min individual 34J	

"Upper shelf temperature"	UK AP-1000 SSER Chap 5 Sec. 5.3.2.5[4] Start of Life: 101J for beltline (10 CFR 50 App G)	Start of Life: 101J in active core region and weld between nozzle course and core shell course 'target' of 88J for nozzle course, nozzles and nozzle welds
80°C		all pressure retaining material: min individual of 3 tests 100J min average of 3 tests 150J
RT_{NDT} [3]	Method for determination set out in ASME III NB-2300, but no criteria for values UK AP-1000 SSER Chap 5 Table 5.3-3: Start of Life: beltline forging -23°C beltline weld -29°C elsewhere +12°C UKP-GW-GL-710 rev 2 Section F Chap 5 page 5-48 & 5-50: end of life (54 efpy) maximum neutron fluence $9.7x10^{19}$ n/cm^2 (E>1MeV) for beltline forging), $2.8x10^{19}$	Start of Life: All forgings except nozzles, less than -12°C. Nozzle forgings, less than -22°C. Indicative end of life (32efpy) bounding neutron fluence (beltline forging) $3x10^{19}$ n/cm^2 (E>1MeV). Weld between top of cylinder and bottom of nozzle course circa $8x10^{18}$ n/cm^2. Weld between bottom of cylinder course and top of transition ring less than $3x10^{17}$ n/cm^2. End of Life (32 efpy) indicative RT_{NDT} = 38°C on inside surface i.e. shift of 50°C (irradiation and thermal ageing combined) for most irradiated

	n/cm^2 for 'lower girth weld'. Preliminary end of life (54 efpy) RT_{NDT} (same as RT_{PTS}) for beltline forging and weld 34^{o}C, 64^{o}C respectively (UKP-GW-GL-700 rev 2 Table 5.3-3, Note 2), presumably at the 1/4T location.	location in beltline forging.

A% = Uniform Elongation

Notes to Table 3

1. UK precedent for the Reactor Pressure Vessel is to require fracture toughness tests based on the 'J-integral' fracture parameter, see Appendix 2

2. ASME SA-508 specification requires quenching "in a suitable liquid medium by spraying or immersion". . No specific grain size requirement for A508 Grade 3 Class 1.

3. In ASME III, method to determine RT_{NDT} set out in ASME III Subsection NB, NB-2300 for Class 1 components. Method based on combination of drop weight test results and Charpy impact energy test results. Requires T_{NDT} to be determined using Pellini Drop Weight test. ASME specifies ASTM E208 as the standard for drop weight testing. ASME allows specimen types P1, P2 or P3 to be used. ASME III NB-2300 define RT_{NDT} as:

> (i) T_{NDT} if at T_{NDT} + 33^{o}C, Charpy tests give at least 0.9mm (0.89mm ASME) lateral contraction and not less than 68J absorbed energy;
>
> (ii) If (i) not satisfied, determine temperature T_{Cv} at which Charpy test requirements in (i) are met and then $RT_{NDT}=T_{Cv}$ - 33^{o}C.

TABLE 4 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				
Tin			0.015% max	0.015% max
Aluminium	0.025% max	0.025% max		
Hydrogen				

Note: 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 4

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

TABLE 5 Steam Generator & Pressuriser Materials - Important Material Parameters

	ASME SA508 Grade 3 Class 2 2007 Edition (formerly SA 508 Class 3A)	**ASME SA508 Class 1A 2007 Edition**	**UK Usage of SA508 Class 3 (now called SA508 Grade 3 Class 1) and welds for Steam Generators base materials and welds [1]**	**UK Usage of SA508 Class 3 (now called SA508 Grade 3 Class 1) and welds for Pressuriser base material and welds[2]**
Austenitising Temperature			"to produce an austenitic structure"	"to produce an austenitic structure"
Tempering Temperature	min 620^{o}C (4.2.2)	min 620^{o}C (4.2.2)	min 650^{o}C (4.2.2) min 635^{o}C (S13)	min 650^{o}C (4.2.2) min 635^{o}C (S13)
Simulated Stress Relief Heat Treatment			less than 620^{o}C (S13)	less than 620^{o}C (S13)

Tensile Properties				
Room Temp	min yield S_y =450MPa UTS S_u = 620-795Mpa min A% = 16%	min yield S_y =250MPa UTS S_u =485-655Mpa min A% = 20%	min yield S_y = 345MPa UTS S_u = 550-725MPa min A% = 18%	min yield S_y = 345MPa UTS S_u = 550-725MPa min A% = 18%
Design Stress				
Room Temp	206MPa	161MPa	184MPa	184MPa
300°C	206MPa	127MPa	184MPa	184MPa
350°C	206MPa	123MPa	184MPa	184MPa
Charpy Impact Energy				
4.4°C	min average of 3= 48J min single value= 41J	min average of 3= 20J min single value= 14J	min average of 3= 41J min single value= 34J	min average of 3= 41J min single value= 34J
"Upper shelf temperature"			'target' of 88J for all shell materials	'target' of 88J for all shell materials
RT_{NDT} [3]			Less than -12°C	Less than -12°C

A% = Uniform elongation

Notes to Table 5:

1. For UK Steam Generator forgings and welds J-resistance curve fracture toughness parameter determined at 300°C. Equipment Specification included requirement for forgings to show upper shelf initiation toughness of at least 165MPa√m.

2. For UK Pressuriser forgings and welds J-resistance curve fracture toughness parameter determined at 345°C. Equipment Specification included requirement for forgings to show upper shelf initiation toughness of at least 165MPa√m.

3. In ASME III, method to determine RT_{NDT} set out in ASME III Subsection NB, NB-2300 for Class 1 components. Method based on combination of drop weight test results and Charpy impact energy test results. Requires T_{NDT} to be determined using Pellini Drop Weight test. ASME both specifies ASTM E208 as the standard for drop weight testing. ASME allows drop weight specimen types P1, P2 or P3 to be used. ASME III NB-2300 defines RT_{NDT} as:

(i) T_{NDT} if at T_{NDT} + 33^{o}C, Charpy tests give at least 0.9mm (0.89mm ASME) lateral contraction and not less than 68J absorbed energy;

(ii) If (i) not satisfied, determine temperature T_{Cv} at which Charpy test requirements in (i) are met and then $RT_{NDT}=T_{Cv}$ - 33^{o}C.

NOTE: ASME III Subsection NC and ND for Class 2 and 3 components respectively does not include requirement for determination of RT_{NDT}. **ASME III Subsection NC for Class 2 components:** unless one of 9 exemptions applies, Pellini drop weight tests (T_{NDT}) or Charpy impact tests are used (64mm maximum thickness), or combination of both (over 64mm thickness). Criterion for T_{NDT} is margin to Lowest Service Temperature (LST) - margin set out in ASME III Appendix R (Non-Mandatory Appendix). **ASME III Subsection ND for Class 3 components:** unless one of 9 exemptions applies, Charpy impact tests are used, test at or below Lowest Service Temperature (LST) with criteria for minimum lateral expansion and absorbed energy.

APPENDIX 2

Example Material Fracture Toughness Requirement in the Equipment Specification - Component Made from Low Alloy Ferritic Steel

Tests shall be carried out on specimens taken from forging material and from weld metal of each weld procedure qualification test.

Fracture toughness J-tests shall be carried out using standard compact tension (CT) J-specimens side-grooved to a depth of 10% each side. The test specimens shall be at least 25mm thick.

Test standard to be defined. Blunting line and exclusion lines to be defined.

Individual J-Δa data shall be reported.

Test shall be carried out at two temperatures:

T1 = (max RT_{NDT} + ΔT) (maxRT_{NDT} and ΔT to be defined)

T2 = Normal Operating Temperature

The following parameters shall be evaluated for each test:

J_{1c} - the value of J at the intersection of the blunting line and the linear regression line of data provided validity criteria are met (this is initiation toughness J)

K_{Jc} - the value of K computed from J_{1c} (initiation toughness K)

$J_{\Delta a2}$ - the value of j computed from the intersection of the data regression line at Δa=2mm

The materials shall meet the following minimum toughness requirements:

Test Temp >	**T1**		**T2**	
	Forging	Weld	Forging	Weld
J_{1c} (kJ/m^2)	160	160	140	140
K_{Jc} (MPa√m)	190	190	170	170
$J_{\Delta a2}$ (kJ/m^2)	700	500	400	250

At temperature T1, no specimen shall show cleavage instability at a K_J value less than 300 MPa√m.

REGULATORY OBSERVATION

RO-AP1000-22

Reactor Pressure Vessel Body Cylindrical Shell Forging Material and Associated Circumferential Welds

Effects of Irradiation

Originated by / Organisation:	Approved by / Organisation:	Assessment Area	Date:
NII	NII	SI	12 March 2009

NOTE: This RO is mainly concerned with the effects of neutron irradiation on the materials of the Reactor Pressure Vessel adjacent to the core. However it also mentions strain ageing and thermal ageing.

UK AP-1000 SSER, UKP-GW-GL-700 Rev 2, Chapter 5, Figure 5.3-6 shows a side elevation of the AP-1000 Reactor Pressure vessel. The vessel body apparently consists of a lower head, a transition ring and a cylindrical section and a nozzle shell course; all these elements are forgings. It is our understanding there is no weld in the cylindrical region of the RPV body. Our understanding of the location of the circumferential welds is shown in Figure 1.

The forged sections are joined by circumferential welds as follows:

between bottom head dome and transition ring (Figure 1, weld 1);

between transition ring and bottom of the lower cylindrical ring (Figure 1, weld 2);

between top of upper cylindrical ring and bottom of nozzle shell / flange course (Figure 1, weld 3).

According to UK AP-1000 SSER,Table 5.3-3, the start of life RT_{NDT} for the beltline forging material is less than or equal to -23^{o}C. UK AP-1000 SSER Table 5.3-3 also refers to a ‘beltline weld’ and this has a start of life RT_{NDT} of less than -29^{o}C.

UK AP-1000 SSER section 5.3.2.2 states that the reactor vessel parts are joined by welding, using single or multiple wire submerged arc and shielded metal arc processes. It is assumed the main circumferential welds (Figure 1) will be made using the submerged arc welding process. The UK AP-1000 SSER does not state whether these welds will have ‘narrow gap’ preparations.

UK AP-1000 SSER section 5.3.2.2 states that minimum preheat requirements have been established for pressure boundary welds using low-alloy material. The preheat is maintained until either a low temperature (204°C – 260°C) post weld heat treatment, an intermediate postweld heat treatment or a full postweld heat treatment is performed.

UK AP-1000 SSER Chapter 4 Table 4.3-6 indicates that the typical neutron flux at full power (E>1MeV) for the pressure vessel peak azimuthal position is 4.71×10^{10} $n/cm^2/s$. The UK AP-1000 SSER does not contain any statements about total neutron fluence to end of life for the RPV. Assuming a life of 54 Effective Full Power Years (efpy) as in UK AP-1000 Table 5.3-3 (60 year design life and 90% availability), gives 1.698×10^9 effective full power seconds of operation. The RPV peak azimuthal neutron flux and 54 efpy give 8×10^{19} n/cm^2 as the end of life neutron fluence at the peak location for the RPV.

UKP-GW-GL-710 rev 2 Section F Chapter 5, page 5-48 quotes the following end of life (60 year, assumed to be 54 efpy) neutron fluences:

9.76×10^{19} n/cm^2 for the forging

2.85×10^{19} n/cm^2 for the “lower girth weld”

UK AP-1000 SSER Table 5.3-3 indicates in Note 2 to the table, that preliminary RT_{NDT} (same as RT_{PTS}) values at end of life (54 efpy) are 34°C for the beltline forging and 64°C for the “beltline weld”; it is assumed these values are at the 1/4T position through the RPV wall. These values are included in Table 3 here, note the indicative end of life RT_{NDT} value for UK usage in Table 2 is for 32 efpy and the location is the inside surface of the RPV.

Tables 1 and 2 compare the chemical composition of relevant Reactor Pressure Vessel base materials and weld consumables respectively.

UK AP-1000 SSER Chapter 5 describes the proposed arrangements for materials irradiation monitoring. Highlights noted are:

The surveillance programme is based on pre-irradiation testing of Charpy V-notch and tensile specimens and post-irradiation testing of Charpy V-notch, tensile and ½-T compact tension (CT) fracture mechanics test specimens;

The surveillance programme includes eight specimen capsules of which 5 are intended for use and 3 are spare;

The capsules contain reactor vessel weld metal, base metal and heat-affected zone metal specimens;

The fast neutron exposure of the specimens occurs at a faster rate than that experienced by the RPV wall. All other things being equal, the effect on material properties of the surveillance specimens at any point in time should be representative of the vessel at a later time;

The following is the recommended withdrawal schedule for surveillance capsules, based on neutron fluence at the capsule locations:

1. neutron fluence equals 5×10^{18} n/cm^2

2. neutron fluence is approximately equal to the end of life fluence at the reactor vessel wall 1/4T location (i.e. 25% in from the inside surface);

3. neutron fluence is approximately equal to the end of life fluence at the reactor vessel inner wall location;

4. neutron fluence between 1x and 2x the peak fluence for the vessel wall at end of life;

5. 60 years (end of life)

Capsules 6 to 8 are shown as ‘standby’. UK-GW-GL-700 Rev 2 Figure 5.3-4 appears to show 3 surveillance capsule locations, with an indication of 3 capsules at two of the locations and 2 capsules at the third location.

UK AP-1000 SSER Table 5.3-4 shows the specimen plan for the eight surveillance capsules. In each capsule there are:

15 Charpy specimens for each of forging (tangential and axial), weld metal and HAZ (60 specimens per capsule);

3 tensile specimens for each of forging (tangential and axial) and weld metal (9 specimens per capsule);

2 1/2T CT specimens for each of forging (tangential and axial) and weld metal (6 specimens per capsule).

DISUSSION

Issue 1

The end of life peak neutron fluence (54 efpy) to the beltline forging is about 9.7×10^{19} n/cm^2 (E>1MeV). By comparison, UK expectation for existing plant is about 3×10^{19} n/cm^2 (E>1MeV) (32 efpy). Part of the difference can be accounted for by the different lifetimes (factor of 1.7). However the AP-1000 peak fluence is still about a factor of 2 higher even accounting for the difference in life.

We note the AP-1000 RPV cylinder section has an internal diameter of about 4040mm, compared with UK existing plant of 4395mm. The UK AP-1000 SSER Figures 3.9-5 and 3.9-8 are the only diagrams we can find of the lower reactor internals and their relationship to the RPV. It is not clear whether features of the reactor vessel lower internals have an effect on the end of life neutron fluence.

Is the factor of 2 difference between peak end of life fluence of AP-1000 and adjusted value from expected end of life value for existing UK plant, due to physical differences in core or lower internals geometry, or is this difference within expected uncertainty in fluence calculations or input data?

Are the AP1000 fluence estimates based on a ‘low leakage’ core?

Issue 2

It appears that the RPV girth weld that receives the highest neutron fluence is the 'lower girth weld', weld 2 in Figure 1, also referred to in the UK AP-1000 documentation as the 'beltline weld'. The end of life (54 efpy) neutron fluence to this weld is about $2.85x10^{19}$ n/cm^2 (E>1MeV). This differs from UK expectation for UK existing plant, where the weld subjected to the highest neutron fluence is the weld at the top of the cylinder, weld 3 in Figure 1. UK expectation for existing UK plant is the weld equivalent to weld 2 in Figure 2 will be receive a relatively low neutron dose by end of life.

What is the explanation for the AP-1000 weld 2 in Figure 1 being subjected to a higher neutron does than weld 3? We note the length of the AP-1000 fuel assemblies (and therefore the core) is the same as a typical 'XL' fuel assembly (UK-GW-GL-700 Rev 2 Table 4.1-1 Sheet 4 of 4). Does this increased length for XL fuel compared with non-XL fuel mean the active region of the core extends lower down in the RPV, and so weld 2 is not shielded to the same extent with XL length fuel compared with shorter fuel?

<u>Issue 3</u>

For AP-1000, what is the anticipated end of life neutron fluence to the weld at the top of the cylindrical region of the RPV cylinder region, i.e. weld 3 in Figure 1?

<u>Issue 4</u>

The end of life neutron fluence to the inside surface of the cylindrical region of the RPV body clearly varies both axially and circumferentially. Can a diagram be provided showing contours of end of life neutron fluence for the cylindrical region of the RPV body? For example, using Figure 1 nomenclature, a developed view of the RPV inside surface with contours of neutron fluence, from weld 1 to weld 3 axially and around the complete circumference. On such a diagram it would be of assistance if the location of the following could be indicated:

safety injection nozzles;

neutron panels on the core barrel;

surveillance specimen carriers on the core barrel.

<u>Issue 5</u>

In the UK, fracture mechanics analyses for postulated defects in RPV locations consider both the deepest point and the surface breaking point of hypothetical semi-elliptical surface defects. These fracture mechanic analyses require materials toughness data at the RPV inner surface as well as at the deepest point.

What is the anticipated end of life RT_{NDT} on the inside surface of the RPV at the following locations:

peak neutron fluence (presumably in the cylinder forging of the RPV body cylinder);

weld 2 of Figure 1;

weld 3 of Figure 1.

Issue 6

It is understood that use of Mixed Oxide Fuel (MOX) might increase the maximum neutron dose to the RPV. In addition, use of MOX fuel could change the neutron energy spectrum compared with UO_2 fuel. Does the AP-1000 design include the possibility of use of MOX fuel? If so, how would the use of MOX fuel affect the shift in RT_{NDT} for the Reactor Pressure Vessel core shell base and weld materials?

Issue 7

As a sensitivity study, what would be the effect on neutron fluence and shift in RT_{NDT} of assuming life extension to, say, 80 years?

Issue 8

For AP-1000, the method of determining the change in RPV material properties appears to be based on neutron fluence expressed in terms of n/cm^2 (E>1MeV). We presume the overall method is effectively correlated with a body of existing test data. For the AP-1000, is the neutron energy spectrum for the neutron flux incident on the RPV the same as typical historical neutron energy spectra relevant to the database of test data on which the dose-damage correlation is based?

Issue 9

For the surveillance programme it appears pre-irradiation tests will be confined to Charpy specimens. In practice we have found data from the 1/2T Compact Tension specimens to be at least as useful. The number of CT specimens in each capsule seems rather low. UK practice has been to generally include a total of more than 30 1/2T CT specimens in each surveillance capsule. It has proved useful for continued operation to have test results from this sort of population of surveillance programme CT specimens. In addition, to be able to make meaningful inference of change from unirradiated to irradiated material properties, the heats of material should be the same for manufacture of the specimens which are tested in the unirradiated and irradiated conditions.

Issue 10

Is it the intention to load all eight surveillance capsules into the reactor at start of life, or will those identified as 'standby' be stored outside the reactor for possible future use?

Issue 11

What are the factors of acceleration between dose to the surveillance specimens in the capsules and the inner parts of the Reactor Pressure Vessel wall, for both base metal and welds? In this regard, it is noted the welds are away from the highest neutron fluence, but the specimens appear to be located in the vicinity of the peak in neutron fluence. Is there an intent to 'manage' the range of acceleration factors for neutron dose?

Issue 12

There are a number of sources of water for injection to the RPV (Core Makeup Tanks, Accumulators, In-Containment Refuelling Water Storage Tank). What are the temperatures of the stored water for injection to the RPV? If the storage temperatures can vary, is it possible to give expected and minimum temperatures?

For the various events which might cause actuation of injection to the RPV (including spurious actuation) from the Core Makeup Tanks, Accumulators or IRWST, is it possible to provide a summary of the RPV internal pressure versus the stored temperature of the sources of water for injection?

Issue 13

For events involving injection of water to the RPV from the Core Makeup Tanks, Accumulators and IRWST, have analyses been completed for RPV wall temperature variation and RPV internal pressure versus time?

Issue 14

UK AP-1000 SSER (UKP-GW-GL-700 Rev 2) in Chapter 5.3 indicates that to estimate the effects on RPV material RT_{NDT} of neutron irradiation, USNRC Regulatory Guide 1.99 has been used. It is assumed this refers to Regulatory Guide 1.99, Revision 2 (May 1988) (ref 1). As of 14 October 2008, Revision 2 of Regulatory Guide 1.99 was the version available on the USNRC web site. The key references in USNRC Regulatory Guide 1.99 Revision 2 are its refs 2 and 3 and these are dated 1984. USNRC Reg. Guide 1.99 Rev 2 is included in a survey of national regulatory requirements (ref 2) and reviewed in ref 3.

Figure 1 of USNRC Reg. Guide 1.99 Rev. 2 shows fluence factor versus neutron fluence (n/cm^2, E>1MeV). The maximum neutron fluence in this figure is $1x10^{20}$ n/cm^2. The maximum neutron fluence to the AP-1000 RPV (forging material) is more or less equal to the maximum value in USNRC Reg. Guide 1.99 Rev. 2 Figure 1. Figure 2 of USNRC Reg. Guide 1.99 Rev. 2, shows decrease in (upper) shelf energy versus neutron fluence and here the maximum fluence is $6x10^{19}$ n/cm^2.

For neutron fluence above about $1x10^{19}$ n/cm^2, USNRC Reg. Guide 1.99 Rev.2 Figure 1 shows a ‘saturation’ effect, that is decreasing change in RT_{NDT} with increasing fluence.

From a recent Symposium (ref 4), there appear to be questions over trend curves at high neutron fluence (e.g. above about $3x10^{19}$ n/cm^2, E>1Mev). In particular there seems to be some doubt about whether saturation of the embrittling effect occurs at high doses. Also high flux could have a role in some observed high fluence effects.

Given the UK AP-1000 range of parameters for:

Copper and Nickel content;

neutron flux;

end of life neutron fluence to regions of the RPV body;

how much of the data that was used to establish the neutron dose - damage relationship is relevant to AP-1000 (e.g. change in RT_{NDT} with neutron fluence)?

Are uncertainties in changes in RT_{NDT} and upper shelf Charpy Impact Energy at high neutron fluence adequately accounted for in the trend curves used (noting that the neutron fluence values are best-estimate calculations)?

Is this an area where changes can be expected in the foreseeable future? For example publication of USNRC Regulatory Guide 1.99 Revision 3?

Issue 15

How is the potential for thermal and strain ageing accounted for in prediction of changes in fracture properties of RPV pressure boundary materials? In this regard it is noted there may be differences in variation with time between these potential ageing mechanisms and the effects of neutron irradiation.

References

1. USNRC, Radiation Embrittlement of Reactor Vessel Materials. Regulatory Guide 1.99 revision 2 (May 1988).

2. Gerard R., Survey of National Requirements. AMES Report No4. EUR16305N (June 1995).

3. Petrequin P., A Review of Formulas for Predicting Irradiation Embrittlement of Reactor Vessel Materials. AMES Report No6. EUR16455 (December 1996).

4. 24th Symposium on Effects of Radiation on Nuclear materials and the Nuclear Fuel Cycle. Denver, Colorado (24-26 June 2008).

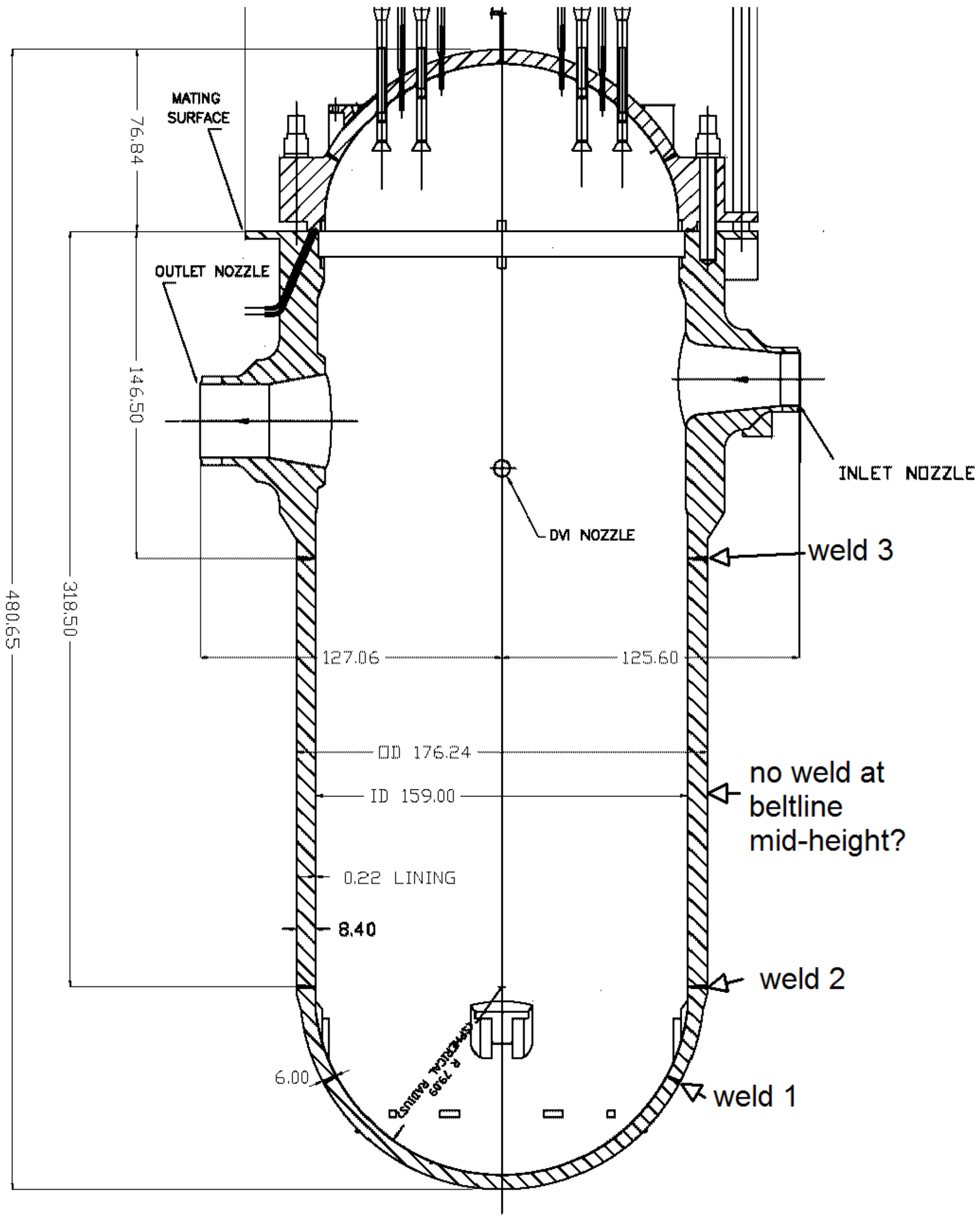


FIGURE 1 AP-1000 RPV side elevation with vessel body welds highlighted

TABLE 1 Reactor Pressure Vessel Materials

	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**
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%	0.85% max
Sulphur	0.025% max	0.008% max	0.01% max
Phosphorus	0.025% max	0.008% max	0.01% max
Silicon [3]	0.4% max	0.3% max	
Chromium	0.25% max	0.15% max	
Copper	0.2% max	0.08% max	0.06% max[5]
Vanadium	0.05% max	0.01% max	0.05% max
Antimony	-	0.008% max	
Arsenic	-	0.015% max	
Cobalt	-	0.02% max	
Tin	-	0.01% max	
Aluminium	0.025% max [2]	0.045% max	
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 and electric furnace and vacuum-degassed.

Notes to Table 1

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.

5. UKP-GW-GL-710 Rev 2 on page 5-50 states for beltline forging material, limit on copper will be 0.03% max.

TABLE 2 For information, the corresponding weld metal chemical composition is given in the table below.

	UK Usage Weld Metal (as deposited)	**UK AP-1000 SSER Chapter 5 Section 5.3 Table 5.3-1 Weld Metal (as deposited)[1]**
Carbon	0.15% max	
Manganese	0.8 to 1.8%	
Molybdenum	0.35 to 0.65%	
Nickel	0.85% max	0.85% max
Sulphur	0.01% max	0.01% max
Phosphorus	0.01% max	0.01% max
Silicon	0.15 to 0.6%	
Chromium	0.15% max	
Copper	0.07% max	0.06% max[2]
Vanadium	0.01% max	0.05% max
Antimony	0.008% max	
Arsenic	0.015% max	
Cobalt	0.02% max	
Tin	0.01% max	
Aluminium		
Hydrogen		

Notes to Table 2

1. Element composition assumed the same as ASME standard composition unless specific different limit stated.

2. UKP-GW-GL-710 Rev 2 on page 5-50 states for beltline forging material, limit on copper will be 0.03% max.

TABLE 3 Reactor Pressure Vessel - Important Material Parameters

	ASME standard composition SA508 Grade 3 Class 1 (formerly SA508 Class 3)	**UK Usage of SA508 Class 3 (now called SA508 Grade 3 Class 1) RPV[5] additions/changes**
Austenitising Temperature	"to produce an austenitic structure"[6]	
Tempering Temperature	min 650°C (4.2.2) min 635°C (S13)	
Simulated Stress Relief Heat Treatment	less than 620°C (S13)	
Tensile Properties		
Room Temp	min yield S_y = 345Mpa UTS S_u = 550-725Mpa min A% = 18%	

Design Stress		
Room Temp	184Mpa	
300°C	184Mpa	
350°C	184Mpa	
Charpy Impact Energy		
4.4°C	min average 41J min individual 34J	

"Upper shelf temperature"	UK AP-1000 SSER Chap 5 Sec. 5.3.2.5[4] Start of Life: 101J for beltline (10 CFR 50 App G)	Start of Life: 101J in active core region and weld between nozzle course and core shell course 'target' of 88J for nozzle course, nozzles and nozzle welds
80°C		all pressure retaining material: min individual of 3 tests 100J min average of 3 tests 150J
RT_{NDT} [3]	Method for determination set out in ASME III NB-2300, but no criteria for values UK AP-1000 SSER Chap 5 Table 5.3-3: Start of Life: beltline forging -23°C beltline weld -29°C elsewhere +12°C UKP-GW-GL-710 rev 2 Section F Chap 5 page 5-48 & 5-50: end of life (54 efpy) maximum neutron fluence 9.7×10^{19} n/cm^2 (E>1MeV) for beltline forging), 2.8×10^{19}	Start of Life: All forgings except nozzles, less than -12°C. Nozzle forgings, less than -22°C. Indicative end of life (32efpy) bounding neutron fluence (beltline forging) 3×10^{19} n/cm^2 (E>1MeV). Weld between top of cylinder and bottom of nozzle course circa 8×10^{18} n/cm^2. Weld between bottom of cylinder course and top of transition ring less than 3×10^{17} n/cm^2. End of Life (32 efpy) indicative RT_{NDT} = 38°C on inside surface i.e. shift of 50°C (irradiation and thermal ageing combined) for most irradiated

	n/cm^2 for 'lower girth weld'. Preliminary end of life (54 efpy) RT_{NDT} (same as RT_{PTS}) for beltline forging and weld 34^{o}C, 64^{o}C respectively (UKP-GW-GL-700 rev 2 Table 5.3-3, Note 2), presumably at the 1/4T location.	location in beltline forging.

A% = Uniform Elongation

Notes to Table 3

1. UK precedent for the Reactor Pressure Vessel is to require fracture toughness tests based on the 'J-integral' fracture parameter, see Appendix 2

2. ASME SA-508 specification requires quenching "in a suitable liquid medium by spraying or immersion". . No specific grain size requirement for A508 Grade 3 Class 1.

3. In ASME III, method to determine RT_{NDT} set out in ASME III Subsection NB, NB-2300 for Class 1 components. Method based on combination of drop weight test results and Charpy impact energy test results. Requires T_{NDT} to be determined using Pellini Drop Weight test. ASME specifies ASTM E208 as the standard for drop weight testing. ASME allows specimen types P1, P2 or P3 to be used. ASME III NB-2300 defines RT_{NDT} as:

(i) T_{NDT} if at T_{NDT} + 33^{o}C, Charpy tests give at least 0.9mm (0.89mm ASME) lateral contraction and not less than 68J absorbed energy;

(ii) If (i) not satisfied, determine temperature T_{Cv} at which Charpy test requirements in (i) are met and then $RT_{NDT}=T_{Cv}$ - 33^{o}C.

REGULATORY OBSERVATION

RO-AP1000-23

Primary Circuit Vessel Nozzle to Safe End Welds

Originated by / Organisation:	Approved by / Organisation:	Assessment Area	Date:
NII	NII	SI	12 March 2009

As usual with a PWR that uses stainless steel pipework, the connection between the pipework and the ferritic pressure vessels is made by means of stainless steel 'safe ends' attached to the ends of the vessel nozzles. The safe ends are welded to the vessel nozzles in the fabrication shop, the welds between the safe end and the pipework being made at site.

The detailed fabrication route for the attachment of safe ends is not given.

For the UK AP-1000 and the following vessel to primary loop piping connections:

Reactor Pressure Vessel;

Steam Generator;

Pressuriser;

will the bimetallic connection between vessel ferritic nozzles and the stainless steel safe ends be made directly (without buttering) or will buttering of the end of the ferritic nozzle be used? If used, what material will be used for buttering the end of the ferritic steel nozzle? What filler material will be used between the safe end and the vessel nozzle?

In addition there is a bimetallic weld required to connect the Reactor Coolant Pump bowls to the outlet nozzles of the Steam Generator Channel Heads. What is the intended method of manufacture of the pump casing to SG connections? For instance, are safe ends used for the pump casing to SG connections? If so the above questions for vessel to piping nozzles also apply.

Whatever form of bimetallic connection is made, what is the extent of experience with the chosen form of safe end to ferritic nozzle weld? How extensive are the weld procedure trials results for this weld type, over all applicable safe end diameter and thickness combinations? What is the experience in practice of ultrasonic examination of such welds?

Response to AP-1000 Technical Query AP-1000-000021, indicates that for in-service examination of the pump casing to Steam Generator Channel Head nozzle welds, the intention is to conduct the examination from the inside surface. Would such examination be by access via the Steam Generator Channel Head, or would it require removal of the canned pump assembly? If removal of the canned pump assembly is required for this examination, would this only be done for other maintenance reasons (i.e. examination alone would not lead to removal of the pump, for instance using Note 2 to ASME XI Table IWB-2500-1 B-L-2, pump casings item B12.20)?

The response in AP-1000 Technical Query AP1000-000021 regarding the need to develop a qualification programme for the examination of the Pump Casing to Steam Generator Channel head weld is noted.

REGULATORY OBSERVATION

RO-AP1000-24

Information on Reactor Internals

Originated by / Organisation:	Approved by / Organisation:	Assessment Area	Date:
NII	NII	SI	12 March 2009

UK AP-1000 SSER (UKP-GW-GL-700 Rev 2) Chapter 3.9, Figure 3.9-5 shows a general drawing of the Lower Reactor Internals, Figure 3.9-6 shows the Upper Core Support Structure and Figure 3.9-8 shows how the internals reside within the Reactor Pressure Vessel. UK AP-1000 SSER sections 3.9.5.1.1 and 3.9.5.1.2 describe in general respectively the Lower Reactor Internals and the Upper Core Support Structure.

UK AP-1000 SSER (UKP-GW-GL-700 Rev 2) Chapter 4.5 section 4.5.2 describes in general terms the materials of construction of the reactor internals and core support.

UK AP-1000 SSER section 3.9.2.3 (page 3.9-33) states:

> "The reactor vessel internals in the AP1000 are similar in size and overall configurations to the reactor vessel internals in previous Westinghouse-designed three-loop nuclear power plants.
>
> The original reference plant for Westinghouse three-loop plant reactor internals flow-induced vibration is H. B. Robinson. The results of vibrations testing at H. B. Robinson are reported in WCAP-7765-AR.…
>
> Successive design changes that have been incorporated into the AP1000 design since the reference plant tests have also been tested in preoperational plant vibration measurement programs, including the following:
>
> - Inverted hat upper internals and 17x17 guide tubes at DOEL 3 and Sequoyah 1
> - XL lower core support structure at DOEL 4
> - Core shroud at Yonggwang 4
> - Neutron panels at Trojan 1
>
> These tests confirmed that the internals behaved as expected and that the vibration levels were within allowable values…..
>
> AP1000 includes design features that differ from the design in plants in which the reactor internals have been tested as outlined previously. These design differences include the following:

• The design has four inlet nozzles and two outlet nozzles in a three-loop size reactor vessel with a three-loop size core barrel diameter;

• The AP1000 core barrel overall length is 11 inches (279.4 mm) longer than that of the standard 3XL design;

.

• The skirt of the internals support structure is 11 inches (279.4 mm) longer than the skirt of previous three-loop internals designs;

• The upper support plate has sixty-nine 9.78 inch (248.4 mm) diameter holes as compared to sixty-one 9.50 inch (241.3 mm) diameter holes in the previous three-loop design. The plate thickness is identical at 12 inches (304.8 mm) in both designs;

• The design has a new in-core instrumentation system;

• The structures below the lower core support plate and the height of the lower plenum have been changed. The core barrel restraint elevation is within the radius of the lower head;

• The reactor coolant is moved using a sealless pump instead of a shaft seal pump;

• A flow skirt is included in the reactor vessel lower head."

UK AP-1000 SSER in section 4.5.2.1 states:

"The estimated peak neutron fluence for the AP1000 reactor internals has been considered in the design. Susceptibility to irradiation-assisted stress corrosion cracking or void swelling in reactor internals identified in the current pressurized water reactor fleet are being addressed in reactor internals material reliability programs. The selection of materials for the AP1000 reactor internals considers information developed by these programs."

UKP-GW-GL-710 Rev 2 in Section F Chapter 4 states in section 4.5.2.2.5:

"…the estimated peak neutron fluence for the AP1000 reactor vessel internals is 9E21 n/cm^2. At this neutron fluence, neither IASCC nor void swelling is expected. In addition, the ongoing EPRI/MRP reactor internals program addresses these issues….. the … applicant should address the findings from the EPRI/MRP reactor internals program applicable to the AP1000 reactor internals design."

UK AP-1000 SSER in section 3.9.8.2 states:

"The consistency of the reactor vessel core support materials relative to known issues of irradiation-assisted stress corrosion cracking or void swelling has been evaluated and addressed in APP-GW-GLR-035 (Reference 21)."

Reference 21 is: APP-GW-GLR-035, "Consistency of Reactor Vessel Core Support Materials Relative to Known Issues of Irradiation-Assisted Stress Corrosion

Cracking (IASCC) and Void Swelling for the AP1000 Plant," Westinghouse Electric Company LLC.

Regarding the Core Shroud, UK AP-1000 SSER in section 3.9.5.1.1 states:

"The core shroud is located inside the core barrel and above the lower core support. This shroud forms the radial periphery of the core. Through the dimensional control of the cavity (the gap between the fuel assemblies and the shroud) and the shroud cooling flow inlets, the core shroud provides directional and metered control of the reactor coolant through the core. The core shroud serves to provide a transition from the round core barrel to the square fuel assemblies."

And UK AP-1000 SSER in section 3.9.5.1.3 states:

"The core shroud is between the lower core barrel and core, surrounding the core and forming the core cavity. The core shroud consists of formed vertical plates with fully welded vertical seams to prevent lateral flow from the fuel assemblies. This core shroud is a proven design that is currently utilized in operating plants."

DISCUSSION

Despite the information summarised above, from a structural integrity perspective, there is a shortage of useful information. For example the wall thickness of the cylindrical 'barrel' of the Lower Core Support Assembly is not mentioned and there is no indication of the location or nature of welds in the barrel. The attachment of the Lower Core Support Plate to the lower end of the Core Barrel is not explained. The extent to which bolting is used is not covered; it appears that the neutron panels are fixed to the core barrel using bolts or similar fixings. It may be that relevant information is contained in APP-GW-GLR-50 "Reactor Internals Design Specification and Design Reports Summary" (listed in UK AP-1000 SSER Table 3.9-19 (UKP-GW-GL-700 Rev 2)).

[Note the matter of summary reports for the design specifications and design reports is dealt with across components in RO-AP1000-25]

There is brief mention of the neutron fluence to the "reactor vessel internals" in UKP-GW-GL-710 Rev 2, but the exact location of the internals subject to this fluence is not stated. It is considered important to know the neutron fluence of those components forming the load path for the weight of the fuel, including the Core Barrel, the Lower Core Support Plate, the connection between the Core Support Plate and the Core Barrel and the flange region at the top of the Core Barrel.

There is no indication of the start of life and end of life fracture toughness of the Core Support Plate, Core Barrel base materials and associated welds.

No arguments or evidence are presented to support the claim that "…neither IASCC nor void swelling is expected…". But if this is so certain, what role does the "ongoing EPRI/MRP reactor internals program" have in addressing these issues? It may be that report APP-GW-GLR-035 includes such arguments and evidence.

It is stated the reactor lower internals includes a 'Core Shroud' and this is made from formed plates with vertical welds. The Core Shroud probably carries little load, but has an important role in maintaining the design coolant flow route through the fuel assemblies. Is there a means of detecting flow bypassing fuel assemblies in the event of leakage through the Core Shroud?

What neutron fluence will the welds of the Core Shroud receive over the design life of the plant? What is the anticipated fracture toughness of the Core Shroud welds at end of life? Do correlations of fracture toughness against neutron dose for stainless steel base metal and welds apply to the conditions of the Core Shroud (e.g. neutron energy spectrum and associated gamma dose)?

REGULATORY OBSERVATION

RO-AP1000-25

ASME Design Specifications and Design Reports - Current Status and As-Built Status

Originated by / Organisation:	Approved by / Organisation:	Assessment Area	Date:
NII	NII	SI	12 March 2009

UK AP-1000 SSER (UKP-GW-GL-700 Rev 2) section 3.9.3 (page 3.9-42) states:

> "The ASME Code, Section III requires that a design specification be prepared for ASME Class 1, 2, and 3 components. The specification conforms to and is certified to the requirements of ASME Code, Section III. The Code also requires a design report for safety-related components, to demonstrate that the as-built component meets the requirements of the relevant ASME Design Specification and the applicable ASME Code. The design specifications and design reports will be completed as discussed in subsection 3.9.8.2. Design specifications for ASME Class 1, 2, and 3 components and piping are prepared utilizing procedures that meet the ASME Code. The design
> report includes as-built reconciliation.
>
> The as-built reconciliation includes the evaluation of pipe break dynamic loads, changes in support locations, preoperational testing, construction deviations, and completion of the small bore piping analysis".

UK AP-1000 SSER section 3.9.8.2 (page 3.9-96) indicates that reconciliation of the as-built piping includes "…verification of the thermal cycling and stratification loadings considered in the stress analysis discussed in subsection 3.9.3.1.2"

UK AP-1000 SSER section 3.9.8.2 states:

> "The design specification and design reports for the major ASME Code, Section III components and piping are available … via the technical reports listed in Table 3.9-19. Design Specifications and selected design analysis information are also available for ASME Code, Section III valves and auxiliary components."

The content of Table 3.9-19 is reproduced below.

TECHNICAL REPORTS SUMMARIZING DESIGN SPECIFICATION AND DESIGN REPORTS FOR ASME SECTION III COMPONENTS AND PIPING	
Document Number	**Document Title**
APP-GW-GLR-013	Safety Class Piping Design Specifications and Design Reports Summary
APP-GW-GLR-048	Core Makeup Tank Design Specification and Design Report Summary
APP-GW-GLR-049	Accumulator Design Specification and Design Report Summary
APP-GW-GLR-050	Reactor Internals Design Specification and Design Reports Summary
APP-GW-GLR-051	Pressurizer Design Specification and Design Report Summary
APP-GW-GLR-052	Reactor Coolant Pump Design Specification and Design Report Summary
APP-GW-GLR-053	Passive RHR Heat Exchanger Design Specification and Reports Summary
APP-GW-GLR-054	In-Core Instrumentation Guide Tube Design Requirements and Design Report Summary
APP-GW-GLR-055	Reactor Vessel Design Specification and Design Report Summary
APP-GW-GLR-056	Steam Generator Design Specification and Design Report Summary
APP-GW-GLR-057	Control Rod Drive Mechanism Design Specification and Design Reports Summary

DISCUSSION

The Technical Reports listed in the above table (reports series APP-GW-GLR-xxx) apparently *summarise* the Design Specifications and Design Reports for important ASME III components and piping. For further NII assessment, we request the reports listed in the above table. Subsequently during GDA Step 3, NII may wish to examine a sample of existing design specifications and reports.

There is always a need to reconcile the design reports with the as-built condition. However this could at minimum be a relatively minor task, if the as-built condition conforms closely to the design intent. For AP-1000 various piping related loading conditions seem to be included in the final as-built reconciliation requirements. How substantial is the reconciliation expected to be in practice?

REGULATORY OBSERVATION

RO-AP1000-26

Fatigue Crack Initiation - Conservatism in ASME III Appendix I S-N Curves for Stainless Steel Material

Originated by / Organisation:	Approved by / Organisation:	Assessment Area	Date:
NII	NII	SI	12 March 2009

Ref 1 reviews margins in ASME Code fatigue design curves. One conclusion of the review is:

> "The results indicate that the current ASME Code requirements of a factor of 2 on stress and 20 on cycles are quite reasonable, but do not contain excess conservatism that can be assumed to account for the effects of LWR environments."

However table 9 of ref 1 also shows that for stainless steel in PWR primary water environment, the required factor on cycles can range from 19 to 31. The factor on strain or stress for all forms of material considered was 1.6 - 1.7.

There are various ways in which conservatism can be explicitly or implicitly included in a design fatigue endurance calculation. For example a conservative stress analysis could introduce an element of conservatism in addition to that in the design S-N curves.

However it would appear that for situations involving stainless steel and a high calculated usage factor, the actual level of conservatism could be lower than the expected factors of 2 on stress and 20 on cycles..

For stainless steel components where a design analysis for cyclic loading is required, how many component locations have a usage factor exceeding 0.75? For these locations, has a review been made of the levels of conservatism?

REFERENCES

1. Chopra O K., Shack W J., Review of Margins for ASME Code Fatigue Design Curve - Effects of Surface Roughness and Material Variability. USNRC document NUREG/CR-6815 (September 2003).

REGULATORY OBSERVATION

RO-AP1000-27

Reactor Internals - Testing and Inspection Programme for First AP1000 as "Prototype"

Originated by / Organisation:	Approved by / Organisation:	Assessment Area	Date:
NII	NII	SI	12 March 2009

UK AP-1000 SSER (UKP-GW-GL-700 Rev 2) Chapter 3.9, section 3.9.2.3 states:

> "The vibration assessment program for the AP1000 reactor internals determines, prior to testing of the first AP1000, that the internals are not expected to be subject to unacceptable flow-induced vibrations.....
>
> The AP1000 core barrel and core shroud will be instrumented during the pre-operational testing of the first plant to determine the shell mode and beam mode frequencies and amplitudes.

UK AP-1000 SSER (UKP-GW-GL-700 Rev 2) Chapter 3.9, section 3.9.2.4 states:

> The pre-operational vibration test program for the reactor internals of the AP1000 conducted on the first AP1000 is consistent with the guidelines of Regulatory Guide 1.20 for a comprehensive vibration assessment program.
>
> The program is directed toward confirming the long-term, steady-state vibration response of the reactor internals for operating conditions. The three aspects of this evaluation are the following: a prediction of the vibrations of the reactor internals, a preoperational vibration test program of the internals of the first plant, and a correlation of the analysis and test results.
>
> With respect to the reactor internals preoperational test program, the first AP1000 plant reactor vessel internals are classified as prototype as defined in Regulatory Guide 1.20. The AP1000 reactor vessel internals do not represent a first-of-a-kind or unique design based on the arrangement, design, size, or operating conditions……
>
> The pre-operational test program of the first AP1000 plant includes a limited vibration measurement program and a pre- and post-hot functional inspection program. This program satisfies the guidelines for a Regulatory Guide 1.20 Prototype Category plant. AP1000 plants
> subsequent to the first plant will also be subject to the pre- and post-hot functional inspection program. The program for plants subsequent to the first plant satisfies the guidelines for a Non-Prototype Category IV plant.
>
> During the hot functional test, the internals are subjected to a total operating time at greater than normal full-flow conditions of at least 240 hours. This provides a cyclic

> loading of greater than 10^6 cycles on the main structural elements of the internals. In addition, there is some operating time with one, two, or three pumps operating.
>
> Instrumentation is designed and installed to measure the vibration of the internals during hot functional testing...
>
> Since the most notable differences with previously tested designs are in the lower internals, the instrumentation is concentrated on the lower internals. In particular, instrumentation is provided to verify that the incorporation of a core shroud does not cause an unacceptable vibration and to confirm that the flow-induced vibration of the vortex suppression plate is acceptable.....
>
> The reactor internals flow-induced vibration measurement program will be conducted during preoperational tests of the first AP1000. The response of the reactor and the internals due to flow-induced vibration will be measured during the hot functional test. Data will be acquired at
> several temperatures from cold startup to hot standby conditions."

USNRC Regulatory Guide 1.20 Revision 3 (ref 1), defines "prototype" as:

> A "prototype" is a configuration of reactor internals that, because of its arrangement, design, size, or operating conditions, represents a first-of-a-kind or unique design for which no "valid prototype" exists.

Ref 1 also defines the term "valid prototype", along with several other terms that include the word "prototype".

The stated intent for AP-1000 plants after the first (UKP-GW-GL-700 Rev 2 Chapter 3, page 3.9-36):

> "The program for plants subsequent to the first plant satisfies the guidelines for a Non-Prototype Category IV plant."

However, UKP-GW-GL-700 Rev 2 Chapter 1 Appendix 1A, page 1A-7 states that:

> "Subsequent plants will be classified as Non-Prototype Category I based on the designation of the first AP1000 as a Valid Prototype."

UKP-GW-GL-710 Rev 2 in Chapter 3 section 3.9.2.3 states:

> "…the final predictive analysis phase of the RG 1.20 internals vibration assessment program will not be completed until the final design of the reactors internals is fully developed…"

DISCUSSION

It is noted the reactor internals of the first AP-1000 are designated "prototype", even if there does not appear to be alignment between Westinghouse and USNRC on the definition of the term "prototype".

At what point in time will the testing and inspection programme of the reactor internals of the first AP-1000 end? For instance, are inspection activities in the first re-fuelling outage included within the 'prototype' testing and inspection programme for the reactor internals? Assuming no adverse results, what defines the point in time that confirmation has been achieved for long-term response of the reactor internals for operating conditions?

Will the pre-operational vibration testing be done with simultaneous application of the following plant conditions (i) normal operating temperature and normal operating temperature differentials; (ii) full core fuel load?

If a 'dummy' fuel load is expected to be used, what will be the characteristics of the 'dummy' fuel load?

The first AP-1000 will be owned by a particular operator. What form of arrangement is envisaged to make available to succeeding AP-1000 plant licensees and relevant regulator, the detailed results from the 'prototype' testing and inspection of the reactor internals of the first AP-1000? On the face of it, such information would be relevant supporting evidence material to the Station Safety Report.

In different parts of the UK AP-1000 SSER, it is stated that the programme for plants subsequent to the first will satisfy the guidelines for a Non-Prototype Category IV plant or Category I (see above). Please confirm the planned designation for reactor internals for AP-1000 plants after the first. With reference to Figure 1 of ref 1, what route is claimed through the diagram from Prototype to the claimed state of reactor internals for AP-1000 plants after the first? If the route includes "extended satisfactory inservice operation", how will that be demonstrated between the first and subsequent AP-1000 plants?

REFERENCES

1. USNRC, Comprehensive Vibration Assessment Program for Reactor Internals During Preoperational and Initial Startup Testing. USNRC Regulatory Guide 1.20 Revision 3 (March 2007).

REGULATORY OBSERVATION

RO-AP1000-28 CORRECTED

Pressuriser Surge Line Stratification Evaluation - AP-1000 First Plant Only Test

Originated by / Organisation:	Approved by / Organisation:	Assessment Area	Date:
NII	NII	SI	5 June 2009

UK AP-1000 SSER (UKP-GW-GL-700 rev 2) Chapter 3.9 section 3.9.3.1.2 page 3.9-51, states:

> "A monitoring program will be implemented as discussed in subsection 3.9.8.5 at the first AP1000 to record temperature distributions and thermal displacements of the surge line piping, as well as pertinent plant parameters such as pressurizer temperature and level, hot leg temperature, and reactor coolant pump status. Monitoring will be performed during hot functional testing and during the first fuel cycle. The resulting monitoring data will be evaluated to show that it is within the bounds of the analytical temperature distributions and displacements."

section 3.9.8.5 states:

> "A monitoring program will be implemented by the Combined License holder at the first AP1000 to record temperature distributions and thermal displacements of the surge line piping as outlined in subsection 3.9.3.1.2."

UK AP-1000 SSER Chapter 14.2 section 14.2.5 includes a list of tests applicable to only the first AP-1000. This includes "Pressuriser Surge Line Stratification Evaluation"

and section 14.2.9.1.7 (page 14.2-30) states:

> "As described in subsection 3.9.3, temperature sensors are installed on the pressurizer surge line and pressurizer spray line for monitoring thermal stratification and thermal cycling during power operation. Testing is performed to verify proper operation of these sensors. **Note that this verification is required only for the first plant."**

DISCUSSION

It is noted monitoring will be performed during hot functional testing and during the first fuel cycle. Monitoring data will be evaluated to show that it is within the bounds on analysis results. Will monitoring continue to the end of the first fuel cycle of the first AP-1000?

The first AP-1000 will be owned by a particular operator. What form of arrangement is envisaged to make available to succeeding AP-1000 plant licensees and relevant regulator, the detailed results from the comparison of monitoring data versus analysis prediction? On the face of it, such information would be relevant supporting evidence material to the Station Safety Report.

REGULATORY OBSERVATION

RO-AP1000-29

Reactor Pressure Vessel and Primary Circuit Pressure - Temperature Limits and Low Temperature Overpressure Protection

Originated by / Organisation:	Approved by / Organisation:	Assessment Area	Date:
NII	NII	SI	12 March 2009

UK AP-1000 Safety, Security and Environmental Report (SSER) (UKP-GW-GL-700-Rev 2) in Chapter 5 section 5.3, subsection 5.3.3 and subsection 5.3.6.1 deals with Pressure - Temperature (P-T) Limits for the Reactor Pressure Vessel (RPV). Although the P-T limits are determined for the RPV, they are claimed to be applicable for the rest of the reactor coolant system.

The Pressure - Temperature limit curves are effectively claimed to be developed using, for the most part:

the methodology in ASME XI Appendix G;

for material degradation effects due to neutron irradiation, the methodology in USNRC Regulatory Guide 1.99.

According to UKP-GW-GL-710-Rev 2 Section F Chapter 5, Revision 2 of USNRC Regulatory Guide 1.99 was used.

Regions of the RPV considered in setting P-T limits include the cylindrical, beltline region adjacent to the core and the vessel body to head closure flange.

Regarding the beltline region, UK AP-1000 SSER subsection 5.3.3.1 states the following:

1. The fluence values used are calculated values, not best-estimate values;

2. The K_{Ic} critical stress intensities are used in place of the K_{Ia} critical stress intensities;

3. The 1996 version of Appendix G to Section XI is used rather than the 1989 version.

For item 2 above, the difference between the K_{Ic} and K_{Ia} can be seen in Figure A-4200-1 of ASME XI Appendix A. At any particular temperature, $K_{Ia} < K_{Ic}$. For item 3 above, the difference appears to be mainly the method of determining K_{Im}, and as a consequence also K_{Ib}.

Some of the main aspects of the methodology in ASME XI Appendix G are:

(i) postulated defects are surface defects with a depth 0.25 the wall thickness and a defect length along the surface of 6 times the defect depth (for the relevant wall thickness in the beltline of the UK AP-1000);

(ii) defects are postulated on both the inside and outside surfaces (to account for thermal stress profiles through the wall for both start-up and shutdown conditions);

(iii) axial defects are postulated for plates, forgings and axial welds; circumferential defects are postulated for circumferential welds;

(iv) applied stresses are those arising from Level A and B Service Limit loadings;

(iv) the requirement to be satisfied and from which an allowable pressure can be determined for any assumed rate of temperature changes is:

$2K_{Im} + K_{It} < K_{Ic}$ (K_{Ic} used by virtue of item 2 above)

for locations away from nozzles, flanges and shell regions near geometric discontinuities;

(v) for locations involving nozzles, flanges and shell regions near geometric discontinuities, primary bending stress and secondary membrane and bending stresses are required to be considered in addition to primary membrane and thermal stresses.

K_{Ic} values ‘at the locations of interest’ are determined using a reference curve (for unirradiated material) provided in ASME and a shift in reference temperature determined using USNRC Regulatory Guide 1.99 (Revision 2). The definition of ‘locations of interest’ includes the crack tip at its deepest point, and takes account of the attenuation of neutron flux through the vessel wall (1/4T or 3/4T position).

DISCUSSION

The overall methodology used to determine Pressure - Temperature limits for the RPV (and the rest of the primary circuit), are long-standing. For example USNRC Regulatory Guide 1.99 Revision 2 was issued in May 1988. And the requirements of ASME XI Appendix G have existed in substantially their current form (originally in ASME III Appendix G) for many years. For example much the same methodology is contained in ASME III Appendix G in the 1983 Edition of the Code.

The ASME XI Appendix G methodology might be characterised as a nominal, deterministic procedure to determine pressure - temperature limits. It is nominal in the sense that some aspects, e.g. choice of defect depth and aspect ratio are not based on any obvious criterion; other than it defines a large defect that is unlikely to occur in practice.

The procedure has a number of explicit margins applied and contains a number of implicit margins. It may also contain some negative margins, for example only the deepest point of the postulated defect on the inside surface is considered, along with the corresponding neutron fluence at that point. For the inside surface crack, the crack tip at the surface (along with the higher neutron dose on the inside surface) is not explicitly considered. On the other hand, for the postulated crack on the outside surface, taking the neutron fluence at the deepest point of the crack front bounds the neutron fluence at any point on its crack front.

The change from use of K_{Ia} to K_{Ic} as the criterion was first introduced with ASME Code Case N-640, appeared again in ASME Code Case N-641 and was incorporated in ASME XI Appendix G in Addenda to the 1998 Edition of the Code.

The motivation for older plants to avoid impact on operating windows at low temperature is understandable. However for new plant with, for example, restrictions on Copper based on current understanding of neutron irradiation embrittlement processes, the need to remove conservatism is less obvious.

Matters relating to the determination of shift in reference temperature are dealt with in another Regulatory Observation.

Issues

<u>Reactor Pressure Vessel "Beltline Region"</u>

From the perspective of

(i) "As Low as Reasonably Practicable (ALARP)", (e.g. paragraphs 9-17 and 134 of the NII Safety Assessment Principles, ref 1),

(ii) a UK AP-1000 with Reactor Pressure Vessel material chemical composition restricted as indicated in UK AP-1000 SSER (UK-GW-GL-700, rev 2, Table 5.3-1),

would it be reasonably practicable in determining Pressure - Temperature limit curves and other low temperature limits for the Reactor Pressure Vessel, to use K_{Ia} as the fracture toughness criterion rather than K_{Ic}?

Thinking about other combinations of assumptions for a nominal deterministic procedure to determine P-T limits, would be following be reasonably practicable:

1. For postulated defects on the inner surface of the RPV, and when considering the deepest point on the crack front, would it be possible to use a fracture toughness criterion value based on the reference temperature shift corresponding to the neutron fluence at the inner surface of the vessel, rather than at the 1/4T position?

2. For postulated defects on the inner surface of the RPV, compare the applied stress intensity in the vicinity of where the crack front meets the inner surface against the fracture toughness criterion adjusted for the effect of neutron irradiation at the inner surface of the RPV?

<u>Reactor Pressure Vessel Closure Flange and other Regions where Bending Stresses Need to be Included (ASME XI Appendix G G-2220)</u>

For locations where bending stress has to be included, is the only point of evaluation still the deepest point of the postulated crack front, or is the vicinity of the point at which the crack front intersects the surface also considered?

UK AP-1000 document UKP-GW-GL-710 rev 2, Section F Chapter 5, sub-section 5.3.3.1 states:

The operating curves are developed in accordance with …. Appendix G, with the exception that the flange requirement is consistent with WCAP-15315 "Reactor Vessel Closure Head/Vessel Flange Requirements Evaluation for Operating PWR and BWR … Plants"

However in section 5.3.3.2 of UKP-GW-GL-710 rev 2 Section F (page 5-47) it is stated that revised P/T curves were established on the basis of WCAP-15315. WCAP-15315 is not mentioned in the UK AP-1000 SSER (UKP-GW-GL-700 rev 2). WCAP-15315 is not included in the set of supporting documents.

For a UK AP-1000, is WCAP-15315 relevant to the determination of Press - Temperature Limits for the Reactor Pressure Vessel and primary circuit components? If so it should be provided as a supporting document.

Questions of Clarification

Separate questions of clarification follow.

The ASME XI Appendix G procedure regarding postulated defects states in G-2120 (2007 Edition, similar wording since 1998 Edition with Addenda):

> "The postulated defects used in this recommended procedure are sharp, surface defects oriented axially for plates, forgings, and axial welds, and circumferentially for circumferential welds…"

For the AP-1000, does this imply that, among those cracks postulated for determination of P-T limit curves, axial cracks in the beltline forging at the location(s) of highest neutron fluence, are included, as well as circumferential cracks in the circumferential welds and forging?

If a range of crack locations and orientations is postulated in both welds and forging material, which postulated cracks are controlling for P-T limits?

UKP-GW-GL-710 rev 2 Section F in 5.3.3.2 (page 5-46) mentions document WCAP-14040-NP-A "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves". To date, WCAP-14040-NP-A does not appear to have been provided as a supporting reference to the UK AP-1000 GDA process.

For a UK AP-1000, is WCAP-14040-NP-A relevant to the determination of Pressure - Temperature Limits for the Reactor Pressure Vessel and primary circuit components? If so it should be provided as a supporting document.

REFERENCES

1. HSE. Safety Assessment Principles for Nuclear Facilities. 2006 Edition Revision 1 (January 2008). (www.hse.gov.uk/nuclear/saps/saps2006.pdf)

REGULATORY OBSERVATION

RO-AP1000-30

Containment Pressure Shell
ASME SA738 Grade B

Originated by / Organisation:	Approved by / Organisation:	Assessment Area	Date:
NII	NII	SI	12 March 2009

Introduction

The UK AP1000 Safety, Security and Environmental Report (UKP-GW-GL-700-Rev 2) in Sub-Chapter 3.8 (Section 3.8.2.1.1) states that the material of construction for the containment shell is ASME SA738 Grade B. The wall thickness of most of the containment shell is 44.4mm (1.75in), with the lowest cylindrical course 47.6mm (1.875in). The axial welds of the lowest cylindrical course are post-weld heat treated, all other seam welds are in the as-welded condition (UKP-GW-GL-710-Rev 2, Section F, Chapter 3, page 3-106). The diameter of the containment shell is 39.6m and the height (crest to crest of domes) 65.6m; the cylindrical section of the containment shell is about 42.6m high.

A material specification for ASME SA738 has existed in the ASME Code since about 1980. The specification has evolved over the years, but fundamentally a material like the current SA738 Grade B could have been procured at any time since 1980. The specification for ASME SA738 Grade B is the same as ASTM A738 Grade B.

That is for SA 738 Grade B:

- The minimum UTS of Grade B (85ksi) is within the original specification range of 75-95 ksi;

- The steel shall be made to a killed fine grain practice. This requirement is in the original specification. The only change since has been to add a reference to Specification A 20 for the definition of ‘fine grain’;

- Grade B must be quenched and tempered. In the original specification this was an option up to 2.5 inch thickness and required over 2.5 inch thickness;

- The maximum Carbon limit for Grade B (0.2%) is within the maximum for the original specification (0.24%);

- Limits on Vanadium and Columbium (Niobium) (Product Analysis) are the same now for Grade B compared to producer / purchaser agreement option in the original specification;

- Limit on Copper content is the same for Grade B as the original specification;

- Limits on Nickel and Chromium are slightly higher for Grade B than the original specification (0.6% and 0.3% max for Grade B and 0.5% and 0.25% max for original specification (Heat Analysis));

- The most notable chemical composition difference is for Molybdenum. The maximum permitted levels of Molybdenum are higher for Grade B compared with the original specification (0.2% up to 1.5 inch thick and 0.3% over 1.5 inch thick for Grade B, 0.08% independent of thickness for original specification (Heat Analysis)).

Up to 2002, SA 738 was only permitted for use for ASME Section VIII Division 1 vessels. An important change occurred in 2002 when the ASME Code was amended to allow SA 738 Grade B to be used for ASME Section III vessels. This allowed SA 738 Grade B to be used in metal containment shells complying with ASME III Subsection NE (Class MC Components).

Another relevant change was made to the ASME Code in 2002, there was a change in maximum allowable stress limits. In ASME II Part D Subpart 1, Table 1A, the Maximum Allowable Stress defined for SA 738 Grade B was altered. The limit of 24.3ksi which had applied up to 200^{o}F was now extended to apply to up to 500^{o}F. The AP-1000 containment pressure shell Design Temperature is 300^{o}F.

For the AP-1000 containment pressure shell, the original ASME Code edition chosen by the designer was the 1998 Edition with Addenda up to and including the 2000 Addenda. Subsequently this was changed to the 2001 Edition including 2002 Addenda. This change brought SA 738 Grade B into the list of materials included in the ASME III Code for use in metal containment shells and altered the Allowable Maximum Stress at the Design Temperature of the AP-1000 containment pressure shell. This change in the Code of reference, and the changes to the Code itself appear to have occurred at about the time of USNRC evaluation of the AP-1000 DCD, Rev 15 (RAIs 220.001, 220.002, 220.003, letter dates between 19/9/02 and 22/4/03).
It is noted that ASME VIII Divisions 1 and 2 (ref 1) provides an exemption from post weld heat treatment for P-No 1 Group No 1 to 3 (SA738B P No 1 Group No 3) for welded joints less than 38.1mm (1.5 inch) (Table UCS-56 for ASME VIII Division 1 and Table 6.8 for ASME VIII Division 2).

It is noted from British Standards Published Document PD5500:2006 (ref 2), that a Group 1.3 material (the relevant Group for SA738B - Table 2.1-1 of ref 1), would not normally require post weld heat treatment if the nominal thickness is less than 35mm (Table 4.4-1 of ref 1), where vessels are designed to operate above 0^{o}C (4.4.3.1 ref 1). For ferritic steel vessels designed to operate below 0^{o}C, post weld heat treatment requirements are in accordance with Annex D of ref 1.

Discussion

In assessing the structural integrity of the containment shell, it is noted that the shell is unlikely ever to see its design loading; in practice it will only be subjected to test loads on a few occasions through plant life.

However, it is understood this material has not been used in practice for a nuclear application before and so practical experience will be from non-nuclear applications.

Potentially the most relevant non-nuclear experience would be for large diameter vessels or storage tanks.

The following questions define the scope of this Regulatory Observation:

What are typical non-nuclear uses of SA738 Grade B?

Do examples of non-nuclear use include large diameter vessels or storage tanks?

Do examples of non-nuclear use include on-site fabrication and assembly of vessels or tanks?

For the AP1000 containment shell:

What is the intended weld consumable?

Which welding processes are intended to be used? (e.g. Submerged Arc, Manual Metal Arc [Shielded Metal Arc], Electroslag)

What is the extent of information on mechanical strength and fracture property data for plate, as-welded and PWHT welds;

It appears likely there will be a number of external surface welds to the plates, to attach the hangers for the air baffle plates. What is the nature of these surface welds and what is the manufacturing inspection of such welds?

Does static strain ageing occur in this material following working the initially flat plates into curved plates for the shell courses?

Document UKP-GW-GL-700-Rev 2 in section 3.8.2.1.1 states the design temperature is 300^{o}F (148.89^{o}C). Section 3.8.2.6 states that the lowest service metal temperature requirement is -15^{o}F (-26.1^{o}C). This temperature is stated to be established from analysis for the portion of the vessel exposed to the environment when the minimum ambient temperature is -40^{o}F (-40^{o}C). From this these questions arise:

Is this lowest service metal temperature determined for the containment shell when the plant is in normal operation (i.e. not for a loading condition design basis accident is occurring)?

For the design basis accident loading condition what assumptions are made about initial environment conditions (ambient temperature) and how does the lowest service metal temperature vary with internal pressure of the design basis accident? How does the metal temperature vary over the containment shell for the design basis accident (including through-wall variation)?

References

1. American Society of Mechanical Engineers Boiler and Pressure Vessel Code. 2007 Edition with 2008a Addenda. ASME VIII Division 1 Rules for Construction of Pressure Vessels. Division 2 Alternative Rules.

2. Specification for Unfired Fusion Welded Pressure Vessels. Published Document PD5500:2006 (including Amendments 1 and 2). British Standards Institution (BSI), London (2007).