New Reactors Programme

GDA close-out for the AP1000 reactor

GDA Issue GI-AP1000-SI-05: Compliance of AP1000 Main Structural Components with ASME III Design Rules.

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EXECUTIVE SUMMARY

Westinghouse is the design company for the **AP1000** reactor. Westinghouse completed Generic Design Assessment (GDA) Step 4 in 2011 and paused the regulatory process. It achieved an Interim Design Acceptance Confirmation (IDAC) to which 51 GDA Issues were attached. These issues require resolution prior to award of a Design Acceptance Confirmation (DAC) and before any nuclear safety related construction can begin on site. Westinghouse re-entered GDA in 2014 to close the 51 issues.

This report is the Office for Nuclear Regulation's (ONR's) assessment of the Westinghouse **AP1000** reactor design in the area of structural integrity. Specifically this report addresses GDA Issue GI-AP1000-SI-05 - Compliance of AP1000 Main Structural Components with American Society of Mechanical Engineers (ASME) III Design Rules. This GDA Issue arose in Step 4 due to:

Action GI-AP1000-SI.05.A1

- At Step 4 of GDA ONR's review of Westinghouse's ASME III stress analysis report for the Reactor Pressure Vessel (RPV) identified a number of areas where it was unclear why specific assumptions and approximations had been made.
- At Step 4 of GDA ONR's review of Westinghouse's ASME III stress analysis report for the Pressuriser (PRZ) identified errors in some calculations. A revision of this report was in preparation during ONR's review.
- The response to ONR comments on the RPV report and the revision of the PRZ report arrived too late for ONR to undertake full assessment within GDA step 4.

Action GI-AP1000-SI.05.A2

- At Step 4 of GDA ONR identified errors on a sample review of the PRZ stress analysis report. The report was verified and issued by Westinghouse, but not fully approved for formal issue. In this circumstance the formal issue of the report corrected the errors identified by ONR. Nonetheless, ONR judged that evidence is required to demonstrate that the process in issuing design reports is sufficiently robust.

The Westinghouse GDA Issue Resolution Plan stated that its approach to closing the issues was to provide:

- adequate responses to questions arising from ONR assessment of documents submitted during GDA Step 4 or in response to action GI-AP1000-SI.05.A1.
- evidence that the process for verifying documents is sufficiently robust in response to action GI-AP1000-SI.05.A2.
- adequate responses to any questions arising from assessment by ONR of the response to action GI-AP1000-SI.05.A2.

My assessment conclusions are:

- Westinghouse has adequately demonstrated compliance with the rules of Section III of the ASME Code for the set of components sampled in this assessment.
- Westinghouse has demonstrated that shortfalls in organisational performance in 2011 are understood and that action has been taken to prevent recurrence.
- Westinghouse has demonstrated that verification and approvals processes are robust and consistently applied in accordance with its internal arrangements.
- Westinghouse verification is not proportionately enhanced for highest reliability components.

My judgement is based upon the following factors:

- The satisfactory outcome of my detailed assessment of a sample of submissions by Westinghouse as evidence of compliance with the rules of Section III of the ASME Code for the Reactor Pressure Vessel (RPV), Pressuriser (PRZ), Steam Generator (SG) and Passive Residual Heat Removal Heat Exchanger (PRHR HX).

- ONR assessment of Westinghouse investigation, learning and improvement processes (ACA and RCA processes).

- ONR assessment of effectiveness of corrective actions, including improvements made to nuclear safety culture.

- ONR assessment of Westinghouse verification and approval processes, including sampling of verification report outcomes.

- Westinghouse enhanced inspection of its verification and approval processes covering all technical disciplines and subsequent improvement action.

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

- Review of developments in design and material selection during licensing for the RPV Control Rod Drive Mechanism penetrations and vent pipe sleeves.

- Provide detailed evidence that ASME III analysis methods adopted for HSS and Class 1 components provide conservative stresses.

- If the loadings on any HSS or ASME III Class 1 vessel are revised during licensing, demonstrate that the ASME III design analysis remains valid and conservative.

- Justify the corrosion allowances for the PRHR HX materials.

- Demonstrate that License Condition 17 “management systems” arrangements provide a robust technical governance framework and a graded verification and approvals approach with the highest standard of that graded approach being applied to Highest Safety Significant and High Integrity components.

In summary I am satisfied that GDA Issue GI-AP1000-SI-05 can be closed.

LIST OF ABBREVIATIONS

ACA	Apparent Cause Analysis
ALARP	As Low As Reasonably Practicable
ASME	American Society of Mechanical Engineers
BMS	Business Management System
CAPAL	Corrective Action Prevention And Learning
CRDM	Control Rod Drive Mechanism
DAC	Design Acceptance Confirmation
EASL	Engineering Analysis Services Limited
FEA	Finite Element Analyses
FW	Feedwater
GDA	Generic Design Assessment
HSS	Highest Safety Significance
IDAC	Interim Design Acceptance Confirmation
IRWST	In-Containment Refuelling Water Storage Tank
MDEP	Multi-Disciplinary Regulatory Evaluation Panel
MFW	Main Feedwater
MSQA	Management for Safety and Quality assurance
NNSA	National Nuclear Safety Administration
NSCEP	Nuclear Safety Culture Excellence Plan
ONR	Office for Nuclear Regulation
PCSR	Pre-Construction Safety Report
PRHR HX	Passive Residual Heat Removal Heat Exchanger
PRZ	Pressuriser
PWSCC	Primary Water Stress Corrosion Cracking
RCA	Root Cause Analysis
RGP	Relevant Good Practice
RPV	Reactor Pressure Vessel
SAPs	Safety Assessment Principles
SG	Steam Generator
TAG	Technical Assessment Guide
TSC	Technical Support Contractor
US NRC	United States Nuclear Regulatory Commission

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INTRODUCTION

1.1 Background

1. Westinghouse completed Generic Design Assessment (GDA) Step 4 in 2011 and paused the regulatory process. It achieved an Interim Design Acceptance Confirmation (IDAC) to which 51 GDA Issues were attached. These issues require resolution prior to award of a Design Acceptance Confirmation (DAC) can be awarded and before any nuclear safety related construction can begin on site. Westinghouse resumed GDA in 2014 to close the 51 issues.

2. This report is the Office for Nuclear Regulation's (ONR's) assessment of the Westinghouse **AP1000** reactor design in the area of structural integrity. Specifically this report addresses GDA Issue GI-AP1000-SI-05 - Compliance of AP1000 Main Structural Components with American Society of Mechanical Engineers (ASME) III Design Rules.

3. The GDA Step 4 structural integrity assessment of the Westinghouse **AP1000** reactor (Ref. 1) is published on our website (Ref. 2) and describes the origin of the GDA Issue. General information on the GDA process is also available on our website (Ref. 3).

4. GI-AP1000-SI-05 was raised in Ref. 1 and required Westinghouse to provide evidence that the design of the **AP1000** reactor main structural vessels complies with Section III of the ASME Boiler and Pressure Vessel Code (ASME Code).

1.2 Scope

5. The scope is described in my assessment plan (Ref. 4) and includes a review of Westinghouse submissions related to this issue. My assessment concentrated on evidence of compliance of **AP1000** main structural components with design rules of Section III of the ASME Code, and evidence of verification by Westinghouse of its analyses to demonstrate compliance.

6. This GDA Issue is captured in two actions in the Resolution Plan (Ref. 5) as follows:

- **GI-AP1000-SI.05.A1:** Support the assessment of Westinghouse's response to ONR's findings on the **AP1000®** Stress Analysis. The review of the reactor pressure vessel report identified a number of areas where it was unclear why specific assumptions and approximations had been made. In its response to this review Westinghouse justified these. The review of the pressuriser report identified errors in the calculations for the safety relief nozzle however a revision of this report was in preparation during ONR's review; this corrected all the main errors. The response to the comments on the reactor pressure vessel report and the revision of the pressuriser report were both supplied too late for ONR to undertake a full assessment of these documents within GDA step 4. Westinghouse should provide adequate responses to questions arising from ONR assessment of documents submitted during GDA Step 4 or in response to this action.

- **GI-AP1000-SI- 05.A2**: Provide evidence that there will not be similar errors elsewhere in the design support documentation. ONR has identified errors on a sample review of the design calculations. The calculations were verified and issued, and referred to within the GDA submissions, but not approved as the formal issue (Rev 0) of the report. In this circumstance the formal issue of the report corrected the errors in the calculational route of 'design by rule', and in this case, even if error had not been detected, the design was still secure because the design route 'design by analysis" had also been followed. Nevertheless, since a sample review identified significant errors in a verified document, evidence is required to demonstrate that the process in raising design reports to Rev 0 is sufficiently robust to ensure that errors missed by the author and verifier of the earlier revisions will be reliably detected. Activities by Westinghouse should comprise:

(i) Provide evidence that the process for raising verified documents to Revision 0 is sufficiently robust.

(ii) Provide adequate responses to any questions arising from assessment by ONR of the response

7. The scope of assessment is appropriate for GDA because, in the United Kingdom (UK), there is an expectation that the safety case for a nuclear facility should demonstrate that the facility conforms to relevant good practice (RGP), such as by design against a set of deterministic engineering rules.

8. The scope of my assessment does not include matters already found by ONR to be satisfactory, as reported in Reference 1.

1.3 Method

9. This assessment complies with ONR guidance on the mechanics of assessment (Ref. 6) and with the requirements of the ONR Business Management System (BMS) document “Purpose and Scope of Permissioning” (Ref.7) which defines the process of assessment within ONR.

1.3.1 Sampling strategy

10. It is rarely possible or necessary to assess an entire safety submission, therefore ONR adopts an assessment strategy of sampling. Reference 7 explains the process for sampling safety case documents.

11. The sampling strategy for this assessment focused on the method and application of rules of the ASME Code for design of **AP1000** reactor components, identified in Reference 1 as requiring further evidence to establish compliance with UK expectations of RGP.

2 ASSESSMENT STRATEGY

2.1 Pre-Construction Safety Report (PCSR)

12. ONR's GDA Guidance to Requesting Parties (Ref. 8) states that the information required for GDA may be in the form of a PCSR, and Technical Assessment Guide (TAG) 051 (Ref. 9) sets out regulatory expectations for a PCSR.

13. At the end of Step 4, ONR and the Environment Agency raised GDA Issue GI-AP1000-CC-02 (Ref. 10) requiring that Westinghouse submit a consolidated PCSR and associated references to provide the claims, arguments and evidence to substantiate the adequacy of the **AP1000** plant design reference point.

14. A separate regulatory assessment report is provided to consider the adequacy of the PCSR and closure of GDA Issue GI-AP1000-CC-02, and therefore this report does not discuss the structural integrity aspects of the PCSR. This assessment focused on the supporting documents and evidence specific to GDA Issue GI-AP1000-SI-05.

2.2 Standards and Criteria

15. The standards and criteria adopted within this assessment are principally the Safety Assessment Principles (SAPs) (Ref. 11), internal TAGs, relevant standards and RGP informed by existing UK practice and international standards.

2.2.1 Safety Assessment Principles

16. The key SAPs that have informed this assessment are listed in

17. Table **1**.

2.2.2 Technical Assessment Guides

18. The key TAGs that have informed this assessment are listed in Table 2.

2.2.3 National and International Standards And Guidance

19. Standards and guidance that have informed this assessment are listed in Table 3.

2.3 Use of Technical Support Contractors (TSCs)

20. A Technical Support Contractor (TSC) was engaged to support closure of GDA Issue GI-AP1000-SI-05. The TSC, Frazer-Nash Consultancy Limited (Frazer-Nash), provided independent expert review of Westinghouse's application of the ASME Code for design of **AP1000** reactor components.

2.4 Integration with Other Assessment Topics

21. GDA requires the submission of an adequate, coherent and holistic generic safety case. Regulatory assessment cannot therefore be carried out in isolation as there are often safety issues of a multi-topic or cross-cutting nature. This assessment has considered information from Management for Safety and Quality assurance (MSQA) specialists in ONR.

2.5 Out of Scope Items

22. This report does not consider structural integrity aspects of the PCSR, which is covered by a separate ONR cross discipline assessment.

3 REQUESTING PARTY'S SAFETY CASE

23. Nuclear pressure vessels and piping are designed to internationally accepted design codes and Westinghouse has designed the **AP1000** plant against the American Society of Mechanical Engineers (ASME) nuclear design code. Section III of the ASME Code provides rules for calculating the required dimensions of pressure-containing components, taking into account operating pressures, operating temperatures, materials of construction, thermal effects, plant faults and accident conditions. The code provides protection against the likely failure modes of such components, i.e. plastic collapse, plastic/thermal ratcheting, buckling and fatigue.

24. The subject of this assessment is the compliance of **AP1000** reactor main structural components with ASME III Design Rules. Section III of the ASME Code provides methods of design, either by rule or by analysis, to safely determine component sizes and geometry.

25. The 'Design by Rule' method uses simple mathematical formulae to determine the required thicknesses of the major parts of a pressure vessel, whereas the 'Design by Analysis' method determines the stresses in a pressure vessel in detail (typically using Finite Element Analyses (FEA)) and compares these to allowable limits to demonstrate compliance with the code. A vessel can be designed by either method, but for safety significant vessels 'Design by Rule' is commonly used in the initial design which may subsequently inform the modelling assumptions for 'Design by Analysis'. The 'Design by Rule' method is sometimes used as a scoping method to obtain a starting geometry prior to undertaking 'Design by Analysis'.

26. During this assessment, Westinghouse has submitted evidence of compliance with ASME III design rules for the following components:

- Reactor Pressure Vessel (RPV) (Refs. 12 to 14)
- Pressuriser (PRZ) (Refs. 15 to 23)
- Steam Generator (SG) (Refs. 24 to 32)
- Passive Residual Heat Removal Heat Exchanger (PRHR HX) (Refs. 33 to 37)

4 ONR ASSESSMENT OF GDA ISSUE GI-AP1000-SI-05

27. This assessment has been carried out in accordance with the ONR BMS document "Purpose and Scope of Permissioning" (Ref.7).

4.1 Scope of Assessment Undertaken

28. I sampled several Westinghouse documents related to the ASME III design and analysis of major vessels in the **AP1000** plant classified by Westinghouse as either HSS or Standard Class 1 (Ref.38). To determine the adequacy of the Westinghouse response to GDA Issue GI-AP-1000-SI-05 A1, I have undertaken the following assessment activities:

- Review of the Westinghouse documents.
- Multiple technical meetings with Westinghouse, where I:
  - Discussed my regulatory expectations, based on relevant good practice.
  - Discussed the associated technical and safety aspects of each of the submissions to ensure there was sufficient evidence to inform my regulatory judgement.
- Issuing of several detailed regulatory queries to progress the assessment of the Westinghouse generic documentation and ASME III design assessments.
- Inspection of the Westinghouse verification process.

29. To determine the adequacy of the Westinghouse response to GDA Issue GI-AP-1000-SI-05 A2, I have undertaken the following assessment activities:

- Review of the Westinghouse investigation into this issue.
- Review of Westinghouse procedures and records.
- Inspection of Westinghouse arrangements and evidence of implementation.
- Several L4 technical meetings with Westinghouse to gain clarity as to the meaning of its responses and provide feedback.
- Issuing of several Regulatory Queries to ensure I had sufficient evidence to substantiate my regulatory judgement and to provide feedback.

30. My overriding assessment objectives were to consider whether Westinghouse's safety case submissions:

- Adequately addressed the key points raised in ONR's Step 4 structural integrity assessment report.
- Adequately consider UK relevant good practice for ASME III design assessments.

31. These assessment objectives were intended to draw out conclusions as to whether there is adequate evidence to support the closure of GDA Issue SI-05.

4.2 Assessment

32. This part of the report is divided into three sections and which describe in turn the following aspects of my assessment:

- Assessment of GDA Issue SI-05 Action 1
- Assessment of GDA Issue SI-05 Action 2
- Key assessment considerations and regulatory judgements.

4.2.1 Assessment of GDA Issue SI-05 Action 1

33. It is ONR's expectation that Structures, Systems and Components important to safety are designed to internationally accepted design codes and Westinghouse has designed the **AP1000** reactor against the American Society of Mechanical Engineers nuclear design code, ASME III (see Table 3). The ASME III code provides detailed and comprehensive rules for calculating the required dimensions of pressure-containing components, taking into account operating pressures, operating temperatures, materials of construction, thermal effects, plant faults and accident conditions. The code provides protection against the likely failure modes of such components, e.g. plastic collapse, plastic/thermal ratcheting, buckling and fatigue.

34. ONR is familiar with the requirements of ASME III and judges these to be generally acceptable for nuclear pressure systems. A large part of the use of an appropriate design code is the correct and accurate interpretation and application of the code by the designers. The expectation is that the designer has suitably qualified and experienced staff and appropriate procedures to ensure that the design complies with the chosen design code. It would not be appropriate for a regulator to check systematically every calculation that is made, but the regulator can judge from a sampling review the quality of the calculations and the qualification and experience of the designers.

35. Two methods are often employed in pressure vessel design; the 'Design by Rule' method uses simple mathematical formulae to determine the required thicknesses of the major parts of a pressure vessel, whereas the 'Design by Analysis' method determines the stresses in a pressure vessel in detail (typically using Finite Element Analyses (FEA)) and compares these to allowable limits to demonstrate compliance with the code. A vessel can be designed by either method, but for safety significant vessels 'Design by Rule' is commonly used in the initial design which may subsequently inform the modelling assumptions for 'Design by Analysis'. The 'Design by Rule' method is sometimes used as a scoping method to obtain a starting geometry prior to undertaking 'Design by Analysis'. When this approach is adopted, it is important that the 'Design by Analysis' covers all aspects of the design requirements, especially if 'Design by Rule' is not carried out in its entirety, or is not entirely valid for the geometry under consideration. If different aspects of the design are undertaken partly using 'Design by Rule' and partly by 'Design by Analysis', there is a risk that certain aspects of the design code compliance may slip between the methods. It is also important that the output from 'Design by Rule' informs the assumptions used in the subsequent 'Design by Analysis'.

36. Given the importance of 'getting the design right', at GDA Step 4, ONR decided to check a sample of the design calculations for two of the most safety significant steel components; namely the reactor pressure vessel (RPV) and pressuriser (PRZ). The RPV and PRZ are part of the primary circuit of the **AP1000** plant. The RPV shell and removable head contain the reactor core and contains numerous penetrations for reactor coolant nozzles, control rods and other services. The PRZ, as its name implies, is used to control the pressure in the primary reactor coolant circuit.

37. The ASME design assessments performed by Westinghouse provide a principal contribution to the Structural Integrity Safety Case for the **AP1000** reactor. In particular, for the UK the RPV, PRZ and SG are classified by Westinghouse as highest safety significance (HSS), which is equivalent to a highest reliability claim in the ONR SAPs to discount gross failure (Ref.38). A demonstration that these components adhere to an established nuclear design code makes a significant contribution to underpinning a claim for highest reliability (SAP EMC.1 to EMC.3 and ECS.2). Indeed, the ONR expectations are based on 'high burden of proof' because nuclear safety is entirely dependent on the structural integrity case when highest reliability is claimed.

38. At Step 4 of GDA, ONR commissioned Engineering Analysis Services Limited (EASL) to review a sample of the design calculations to provide confidence that the RPV and PRZ were compliant with ASME III (generally 1998 Edition with 2000 addenda) (Ref. 1). The fact that the 1998 Edition with 2000 Addenda of the ASME III code was used by Westinghouse for GDA was questioned because it was not the current edition of the ASME code. Westinghouse confirmed this is simply the version of the code chosen by Westinghouse for its design reference point; later changes to the code to the current date will be accounted for in a reconciliation exercise under extant assessment finding AF-AP1000-SI-40 (Ref. 1).

*AF-AP1000-SI-40**:** The Licensee shall carry out a review the changes to the design which would be required if the current version of ASME III were used and either make these changes or justify why these changes are not practical.*

39. The EASL review concentrated on the 'Design by Rule' method for sizing the main vessel shells and the reinforcement around nozzles and penetrations (Ref.39). In addition, EASL reviewed the FEA approach taken by Westinghouse, which underpins the 'Design by Analysis' assessment for the RPV inlet and outlet nozzles.

40. In general, EASL found the Westinghouse reports difficult to follow and this resulted in a large number of comments. ONR requested Westinghouse to respond to EASL's comments, which related to the adequacy of the supporting design calculations, in particular, the assumptions and approximations made for some locations in the RPV and PRZ. Westinghouse's responses to EASL's comments, along with updated design calculations for the PRZ, were received late in GDA Step 4. The ONR was therefore unable to undertake a full review of Westinghouse's responses and updated calculations at that time. ONR gained sufficient confidence to issue an IDAC, but GDA Issue GI-AP1000-SI-05 A1 was raised to complete the review:

'Support the assessment of Westinghouse's response to ONR's findings on the AP1000® Stress Analysis. The review of the reactor pressure vessel report identified a number of areas where it was unclear why specific assumptions and approximations had been made. In its response to this review, Westinghouse justified these. The review of the pressuriser report identified errors in the calculations for the safety relief nozzle however a revision of this report was in preparation during ONR's review; this corrected all the main errors.
The response to the comments on the reactor pressure vessel report and the revision of the pressuriser report were both supplied too late for ONR to undertake a full assessment of these documents within GDA step 4. Activities by Westinghouse should comprise:

- *Provide adequate responses to questions arising from ONR assessment of documents submitted during GDA Step 4 or in response to this Action.*

With agreement from the Regulator this action may be completed by alternative means.'

4.2.1.1 Post GDA Step 4

41. Westinghouse remobilised in September 2014 to close-out the GDA issues. As part of this close-out, Westinghouse committed to provide responses to questions arising from the ONR assessment of documents supporting the design calculations issued at GDA Step 4. The Westinghouse responses are detailed in (Ref.40).

42. Following GDA Step 4, the inputs to the design calculations and FEA for the RPV and PRZ (and other major pressure vessels) were revised to take account of design changes and construction developments in China and the United States.

43. I commissioned Frazer-Nash to support me with the detailed review work, (Ref.41). The scope of the review covered:

- the unresolved responses to the EASL review.
- a sample of the updated design calculations resulting from post-GDA Step 4 changes.

4.2.1.2 Review of RPV & PRZ Design Documentation

44. A key objective post-GDA Step 4 was to ‘sentence’ the Westinghouse responses to the original EASL comments relating to the RPV and PRZ. In practice, sentencing of the Westinghouse responses proved extremely difficult because of the large number of comments raised, the subsequent updates to many of the reports and the time elapsed since the comments were raised. I established a hierarchy for the significance of the comments and then focused my resources on the most important comments from the EASL review work. The most significant comments were pursued with Westinghouse.

45. I also undertook sampling reviews of the updated design calculations for the RPV and PRZ (Refs. 12 to 23). Notably, these design documents were now verified and approved in accordance with the Westinghouse design procedures. I raised many comments on these documents, which were prioritised according to its significance for Westinghouse to address. I found that some reports (for example, the Design by Analysis of the lower pressuriser head) to be of good standard. However, after consolidating the review work from Ref. 41 with the remaining comments from GDA Step 4, the following concerns were identified:

- Errors and inconsistencies were found in some of the Westinghouse reports.
- It was not clear how the interactions between adjacent features (e.g. nozzles, penetrations, manways, etc.) have been accounted for by the ‘Design by Analysis’ approach, given that preliminary ‘Design by Rule’ calculations indicated that such interactions exist. This concern was compounded in some cases by errors identified in the ‘Design by Rule’ calculations. It is an important part of pressure vessel design to recognise how adjacent features can interact with one another and take this into account in the design calculations.
- With today’s computing power, it is entirely feasible to model the whole, or significant parts of, a complete pressure vessel in 3D so that the ‘Design by Analysis’ calculations account for all the interactions between the various features in the vessel. In some cases, Westinghouse has done this (e.g. the PRZ top head), but for others (e.g. the PRZ manway) it has relied on simplified 2D axisymmetric FEA models of non-axisymmetric vessel features and loadings without sufficient justification that this produces accurate and conservative stress results for the design code assessments.
- Situations were identified where the ‘Design by Rule’ approach was not strictly applicable and it was not clear if the alternative ‘Design by Analysis’ approach

was applied. ‘Design by Rule’ is not mandatory in the ASME III design code and ‘Design by Analysis’ can be used on its own instead in more complicated situations, where the simple ‘Design by Rule’ is not applicable.

- ‘Design by Analysis’ requires the extraction of stresses for assessment at various sections (cut lines) within the components. In general, the choice of these sections is down to the experience of the analyst, with a demonstration by independent check that the most limiting sections have been selected for analysis. However, it was not clear in all cases that the limiting sections for stress extraction had been identified and there were some cases where stresses had been extracted very close to model boundaries, but without a demonstration that the extracted stresses are unaffected by the model boundary. It is important that the most highly stressed areas in the design are correctly identified and that the stresses extracted from the FE models are reliable.

46. I also noted that, although errors in the ‘Design by Rule’ calculations may be overridden by a demonstration of ASME III compliance using ‘Design by Analysis’, the FEA modelling for ‘Design by Analysis’ was often linked to the results of the scoping ‘Design by Rule’ calculations, e.g. in modelling (or not) the interaction between closely spaced nozzle openings. Thus it may not be valid to assume that the ‘Design by Analysis’ approach was completely independent of the ‘Design by Rule’ approach. As a result errors in the ‘Design by Rule’ approach, if not corrected, may subsequently affect the demonstration of ASME III design compliance either in updated ‘Design by Rule’ type calculations or as part of the ‘Design by Analysis’ approach. There was therefore the potential for incoherency in the ASME III compliance demonstration.

47. I drew the following conclusions:

- There was uncertainty relating to demonstrating that the RPV and PRZ were compliant with the ASME III design criteria.
- The majority of the most important points raised in the Step 4 assessment report were unresolved.
- Overall, there was limited progress post Step 4 to provide evidence to close out GDA issue GI-AP1000-SI-05.

48. In summary, there was a lack of an appropriate level of demonstration of ASME III Code compliance for the RPV and PRZ.

49. The RPV and PRZ are classified by Westinghouse as HSS, equivalent to a highest reliability claim in the SAPs, and so uncertainty in compliance with the design criteria of a recognised nuclear code was unacceptable (EMC.1 to EMC.3, ECS.3). However, it was difficult to establish the nuclear safety implications, because of the uncertainty relating to the significance of the errors. I indicated to Westinghouse the significant implications for ONR’s confidence in the veracity of Westinghouse processes and procedures for assuring design code compliance and hence subsequent closure of GI-AP1000-SI-05 (Ref.42).

50. In view of the potential implications for nuclear safety, Westinghouse responded to this ONR observation by initiating its corrective action, prevention and learning process (CAPAL), (Ref.43).

51. Westinghouse also informed the US Nuclear Regulatory Commission (US NRC). I had also recognised the potential wider implications for the **AP1000** reactor plants under construction and commissioning in the US and China respectively. ONR has a bilateral agreement with the US NRC and so as part of that commitment, I also briefed

the US NRC along with the Chinese nuclear regulator; China National Nuclear Safety Administration (NNSA). I kept both the US NRC and NNSA informed through the auspices of the Multi-Disciplinary Regulatory Evaluation Panel (MDEP). (Ref.42)

52. I discussed my conclusions with Westinghouse at a semi-annual meeting in the US (Ref.44). The US NRC observed my discussion of GI-A1000-SI-05. Westinghouse accepted the validity of my comments and subsequently initiated several recovery actions:

- convened an expert panel which underwrote the ONR conclusions
- re-evaluation of ASME III design compliance with updated calculations extended to all ASME III Class 1 vessels
- a nuclear safety evaluation under 10 CFR Part 21 (design and delivery) and 50.55 (e) (plants in construction). (Ref.45)
- Two investigations:

  A Westinghouse Root Cause Analysis (RCA) to establish the root cause(s) for the breakdown in addressing ONR's GDA Step 4 comments i.e. the Management for Safety and Quality assurance (MSQA) aspects

  A Westinghouse Apparent Cause Analysis (ACA) investigation to determine the causal factors and scope of the issues relating to the ASME III design calculations (engineering evaluation)
- A recovery plan informed by the CAPAL and RCA, which will identify the root causes and actions to prevent reoccurrence.

53. I welcomed the positive Westinghouse response and considered its proposals constructive and proportionate. Assessment of the Westinghouse recovery actions, including the findings from its investigations, is covered in Section 4.2.2. The Westinghouse nuclear safety evaluation included initial engineering appraisals that were relevant to the engineering evaluation and are discussed next.

4.2.1.3 Westinghouse Nuclear Safety Evaluation

54. Westinghouse undertook a nuclear safety evaluation to assure compliance with US law, specifically 10 CFR Part 21 and 10 CFR 50.55(e) to determine if the issue was a 'substantial safety hazard' and reportable to the US NRC (Ref.46). Section 10 CFR 50.55(e) applies to a licensee or permit holder for plants under construction. The nuclear safety evaluation is performed when a 'failure to comply' or a 'deviation' is discovered that has the potential to adversely affect a delivered basic component to the extent that it could create a substantial safety hazard if left uncorrected. A substantial safety hazard is defined as a loss of safety function to the extent that there is a major reduction in the degree of protection provided to public health and safety for any facility or activity. The US process includes 'discovery' and 'evaluation' stages which inform a decision on reporting:

- 'discovery' identifies a failure to comply with the law/regulation or a deviation from a technical requirements document that could result in a 'substantial safety hazard' if left uncorrected.
- 'evaluation' evaluates the nuclear safety consequence, if any, with the failure to comply or delivered deviation and concludes whether there is a 'defect' (defects are reportable). Note that the reporting criteria relate to nuclear safety and not conventional or environmental safety.

55. The **AP1000** reactor components considered included the RPV, PRZ, SG, Accumulators, Core Make-Up Tank, and Passive Residual Heat Removal Tank.

56. In my opinion, loss of the integrity of the reactor coolant pressure boundary would affect delivery of several safety functions and constitutes a 'substantial safety hazard', if not justified to ASME III. Westinghouse reported that the errors and inconsistencies identified by ONR would not result in a 'substantial safety hazard' and so were not reportable to the US NRC. Westinghouse confirmed its conclusion relating to the nuclear safety hazard was also applicable to the China **AP1000** plant. (Ref.47).

57. I questioned Westinghouse, whether by inference, the RPV and PRZ were now demonstrably compliant with ASME III design criteria. Westinghouse clarified that 10 CFR 50.55 (e) covered the delivery of components and the reporting criteria related to consideration of a deviation from a procurement document that had the potential to become a 'substantial safety hazard'. Thus, follow-up activities to the 10 CFR Part 21 evaluations to underpin ASME III design compliance were not precluded. Indeed, Westinghouse committed to reviewing and updating its ASME III design calculations for all ASME III Class 1 vessels. However, for the nuclear safety evaluation under 10 CFR Part 21, best estimate approaches and engineering judgement could be invoked to inform the reporting decision.

58. I subsequently undertook an inspection of the Westinghouse processes and procedures for demonstrating ASME III compliance, with the focus on its initial engineering evaluation to US regulations. This inspection is reported in Section 4.2.2 of this report. In terms of the engineering evaluation, I concluded:

- Westinghouse had completed its nuclear safety evaluation in accordance with US Regulations, 10 CFR Part 21.

- Westinghouse had adopted a logical and pragmatic approach to guide its nuclear safety evaluation.

59. The 10 CFR Part 21 evaluation of no 'substantial nuclear safety hazard' underpinned the Westinghouse decision not to formally report the uncertainty in ASME III design compliance under US Regulations. I noted the conclusions of the Westinghouse nuclear safety evaluation. However, this did not significantly affect the progression of my GDA assessment, because I still needed to gain evidence and confidence in the engineering substantiation. I outlined my expectations to Westinghouse to restore my confidence in the engineering substantiation for the ASME III Class 1 pressure vessels (Ref.48):

- I expected adequate responses from Westinghouse to my comments and regulatory queries relating to the RPV and PRZ.

- I would extend my review and sample the design calculations for other **AP1000** reactor ASME III Class 1 pressure vessels and raise regulatory queries where appropriate

- I would review the Westinghouse verification and governance arrangements, along with any improvement initiatives for demonstrating ASME III design compliance.

4.2.1.4 Engineering Evaluation and Regulatory Queries

60. For the engineering evaluation of the RPV and PRZ, my comments were communicated to Westinghouse via RQ-AP1000-1620 (SI-05 Action 1 – Demonstration of ASME III Design Compliance) and RQ-AP1000-1621 (SI-05 Action 1

– 'Interpretation' of ASME Section III Clause NB-3213.10) and Westinghouse provided several responses covering the RPV and PRZ (Refs. 49 and 50)

61. As mentioned above, to restore my confidence in the Westinghouse engineering evaluation for ASME III design compliance, I broadened my sampling and commissioned additional reviews of the ASME III design code assessments for two other important pressure vessels in the **AP1000** reactor; namely, the Steam Generators (SGs) and the Passive Residual Heat Removal Heat Exchanger (PRHR HX). I chose the SG because Westinghouse classifies this component as HSS and ONR had not previously sampled any design documentation for this major vessel. My selection of the PRHR HX vessel was based on its safety significance and common design features with the other HSS vessels sampled. I discussed the consequences of a postulated failure of the PRHR HX with ONR's fault studies and PSA specialists. The PRHR HX system offers significant protection for intact circuit faults and the vessel design includes a large oblique nozzle penetration in close proximity to a manway, which provides a complex challenge for the ASME III design evaluation. My focus for these additional reviews was the key point that emerged from the reviews of the RPV and PRZ ASME design calculations.

4.2.1.5 Review of Westinghouse Response to RQ-AP1000-1621

62. This RQ relates to the interpretation of ASME Section III Clause NB-3213.10 for assessing the interaction of discontinuities in 'Design by Analysis' assessments. It covers 'local primary membrane stress' (P_L) and gives rules that are open to interpretation. After discussions with Westinghouse, I agreed with its interpretation of this clause to assess the potential for interaction effects in the 'Design by Analysis' assessments. Westinghouse subsequently revisited all of its 'Design by Analysis' assessments and provided additional reports for each vessel covering the interaction assessment of all the discontinuities in the vessels to this clause. The implementation of this clause has been successfully revisited for the originally reviewed vessels, the RPV and the PRZ, and therefore on this basis this RQ was successfully closed (Ref. 50)

4.2.1.6 Review of Westinghouse Responses to RQ-AP1000-1620 - General

63. RQ-AP1000-1620 includes many queries, but several of the most important comments relate to the relationship between the 'Design by Rule' and 'Design by Analysis' sections of Subsection NB of ASME Section III and the required stress evaluations in the vessel shells resulting from local discontinuities, such as nozzles.

64. Westinghouse advised me that, in most cases, the existing 'Design by Rule' analyses were superseded by the 'Design by Analysis' analyses and it is therefore on the latter that I concentrated my resources.

4.2.1.7 Review of Westinghouse Responses to RQ-AP1000-1620 - RPV

65. Most of the comments that I raised on the RPV ASME III design in RQ-AP1000-1620 were readily resolved by suitable responses from Westinghouse.

66. However, some of the comments I raised questioned the analysis methodology adopted by Westinghouse. Resolution of these comments was more challenging and required several iterations between Westinghouse and myself. I provide a summary of the key points below:

- The RPV in the **AP1000** reactor is supported entirely by its inlet nozzles, which are cantilevered off the sides of the vessel wall just below the vessel flange. Therefore, any mechanical loads applied anywhere on the vessel, be they pipework, core, head and vessel; dead, live and thermal loads, have to be reacted through the inlet nozzles and into the RPV support structure. As such, it is paramount that Westinghouse's analysis represents the application of all of these loads and its load paths from its point of application, through the vessel, through the inlet nozzles and into the supporting structure.

- I questioned the way in which Westinghouse had applied the loads in its analysis of the inlet nozzles and in particular that its analysis concentrated on the pressure and thermal loadings and the basis for excluding mechanical loads. Westinghouse acknowledged the approximations made in its analysis and provided evidence that demonstrates that the mechanical loads have little influence on the outcome of the ASME III design code assessments. Therefore, the imprecise application of the mechanical loads acting above the inlet nozzles would not affect the overall conclusions. I was satisfied with the Westinghouse response.
- In several assessment locations around the RPV, I questioned Westinghouse's choice of 'cut lines' through the vessel and nozzles at which stresses are extracted for the ASME III assessments. Some cut lines were taken at what I considered unsuitable locations and it was not clear in Westinghouse's reports that the most limiting cut lines had been selected for assessment. However, Westinghouse provided suitable evidence that the cut lines that they had selected were generally limiting and, in cases where there was doubt, that sufficient margins existed to absorb any uncertainties. I was satisfied with the Westinghouse response.
- I was concerned with the low margins to the ASME III limits calculated by Westinghouse for the Control Rod Drive Mechanism (CRDM) penetrations and vent pipe sleeves in the RPV head. These required simplified elastic-plastic shakedown analysis per ASME III NB-3228.4 and NB-3228.5 to demonstrate avoidance of plastic ratcheting. These locations also appear difficult to inspect in-service. When questioned, Westinghouse clarified that a postulated gross failure at this location was protected and so would not result in unacceptable consequences. In addition, Westinghouse advised that a high stress intensity range and corresponding low margins in these locations is typical of such RPV head designs and well recognised by industry and the US NRC. The materials chosen for use in this location have a reduced likelihood of Primary Water Stress Corrosion Cracking (PWSCC), a known issue in these areas. Such areas are subject to enhanced in-service inspections on existing world-wide Westinghouse plants (and have been so since 2001) and similar inspections will be included in the UK **AP1000** reactor pre- and in-service inspection plans to guard against problems in these areas. I was satisfied with the Westinghouse response for the purposes of GDA. However, I expect the future licensee to review developments in design and material selection to ensure the risks to structural integrity for the RPV CRDM penetrations and vent pipe sleeves are reduced so far as is reasonably practicable. This is the subject of my Assessment Finding CP-AF-AP1000-SI-12, see Annex 1.

67. Westinghouse therefore provided adequate responses to close my regulatory queries for the RPV (Ref. 49)

4.2.1.8 Review of Westinghouse Responses to RQ-AP1000-1620 - PRZ

68. Most of the comments that I raised on the PRZ ASME III design in RQ-AP1000-1620 were readily resolved by suitable responses from Westinghouse.

69. Westinghouse went to considerable effort to address comments relating to the interactions between discontinuities, such as the nozzles in the PRZ upper head, via the provision of substantial pieces of new FE modelling work.

70. However, some of the comments I raised questioned the fundamental analysis methodology adopted by Westinghouse. Resolution of these comments was more challenging and required several iterations between Westinghouse and myself. I provide a summary of the key points below:

- As previously mentioned, Westinghouse employed an axisymmetric FE model of the PRZ manway, using 2D symmetric loading conditions to represent the non-axisymmetric structure and loadings. However, this method of analysis is only valid for linear-elastic behaviour and the model contains non-linear contact elements. Furthermore, the studs, nuts, washer and holes are discrete entities around the circumference of the manway; these cannot be explicitly represented in an axisymmetric analysis and so smearing techniques have been employed that are not necessarily amenable to bounding analysis. Based on its longstanding experience with the design of replacement steam generators, Westinghouse *is confident that its approach is conservative and it has raised a CAPAL item to carry out the work necessary to demonstrate this. If its approach is shown not to be conservative, Westinghouse is confident that* there is sufficient margin within the design *to absorb any differences and* demonstrate continued compliance with the ASME III Code. I am therefore satisfied that Westinghouse has provided sufficient evidence for the purposes of the GDA. However, I expect further evidence to validate the 2D analysis route to support licensing. This is the subject of my Assessment Finding CP-AF-AP1000-SI-13, see Annex 1.

- The close proximity of the manway to the junction between the PRZ cylindrical shell and spherical head did not appear to have been addressed in any of the 'Design by Analysis' reports. Westinghouse promptly produced a 3D FEA model of the pressuriser manway and upper head, showing that code compliance is achieved. However, the results from this model highlight the non-axisymmetric nature of the stresses around the manway and the difficulty of modelling such regions with a 2D axisymmetric analysis. This reinforces the need to provide further validation of the 2D axisymmetric approach. This is a generic issue that I capture for licensing. This is the subject of my Assessment Finding CP-AF-AP1000-SI-13, see Annex 1.

- I noted that the nozzles in the PRZ upper head are very close together such that its P_L (local primary membrane) stress regions might overlap; this would not be code compliant because the nozzles would interact unacceptably with one another. I questioned this with Westinghouse and its response was that they agreed that the P_L stress regions do overlap and so are not code compliant. However, Westinghouse then repeated the analysis using its latest "Actual" nozzle loads as opposed to the preliminary bounding "Design" nozzle loads as used in the original assessment. It is not unusual for preliminary vessel design to be based on bounding "Design" nozzle loads, because early in design such loads have not yet been calculated. These are then replaced, as in this case, at some point in the design with the "Actual" nozzle loads, once the necessary pipework layout design and pipestress analysis has been finalised. Whilst the "Actual" nozzle loads represent best estimates of the real nozzle loads, they will be subject to the inherent conservatisms in the code and pipestress analyses. This new and more realistic assessment demonstrates that the P_L stress regions do not now overlap and so is code compliant. I was satisfied with the Westinghouse response.

- In the thermal analysis of the PRZ, Westinghouse assumed an adiabatic (no heat flow) boundary condition on the outside of the vessel. I questioned whether this was a conservative assumption. Westinghouse carried out a sensitivity study where they modelled the small heat flow from the outside of the vessel and showed that the results were not sensitive to the thermal boundary condition. However, I established that Westinghouse had incorrectly applied the heat flow such that heat is flowing into the vessel instead of out of the vessel. Westinghouse claimed that, despite the error, the results were not significantly affected and I concur. Westinghouse subsequently submitted a lessons learned item to its CAPAL database to track this error and raise awareness of the potential shortfalls. I welcome the Westinghouse commitments, but also note this reinforces the need for adequate implementation of verification arrangements for ASME III design documentation.

- Westinghouse carried out a limit analysis of the PRZ Lower Head, Support Pads and Shell to show ASME III compliance with a small margin of 7% on design internal pressure. A limit analysis calculates the margins on primary loads (plastic collapse) by considering the elastic-perfectly-plastic stress-strain behaviour of the material and is less conservative (but still conservative overall) than the standard ASME III assessment route assuming elastic behaviour. This analysis (similar to the RPV above) ignored the mechanical loads at the support pads from the weight of the vessel and attached pipework loadings. My concern was that the small margin could be eroded by the effects of such loads. Westinghouse's response was that the effects of the support pad loadings are small compared to those of the dominant pressure loading (which has been considered) and that they were confident that the analysis results would remain acceptable, even if revised during Site Licensing, because the ASME III Code also allows for the use of plastic analysis (NB-3228.3) to show acceptability. Since Westinghouse has shown the vessel to be code compliant (with a small margin) I am content with this response for the purposes of the GDA. However, the loadings may change during site licensing and so I will seek confirmation that the inputs remain bounding and conservative and that adequate margins are demonstrated to substantiate the structural integrity case for the PRZ through-life. This is the subject of my Assessment Finding CP-AF-AP1000-SI-14, see Annex 1.

- Further to the above point, I also raised a concern regarding the interaction between the surge nozzle and the inner ring of heaterwell penetrations and whether Westinghouse had identified correctly the relevant ASME III stress intensity limits in this region. The inner ring of heaterwell penetrations, despite being located in the P_L region of the surge nozzle, should be evaluated for Pm and Pm+Pb, consistent with the perforated region defined by Table NB-3217-1 of ASME III rather than to higher P_L limit as Westinghouse has done. For a perforated head or shell applicable for the pressuriser lower head heaterwells, there is no P_L stress classification given in Table NB-3217-1 of ASME III. Westinghouse raised an item in its CAPAL database to update its analysis at its next revision and they are confident that ASME III code compliance will be demonstrated. This judgement is supported by the positive results from the limit analysis described above. It is crucial that this revision to the analysis covers the most highly-stressed region, which I believe is between the surge nozzle and the inner ring of heaterwell penetrations, as suggested by the limit analysis. This reinforces the need for robust training in 'Design by Rule' and 'Design by Analysis' for

ASME III design assessments, and especially that training covering the relationship between the methods is developed and implemented to underpin design to the ASME III code.

71. Westinghouse provided adequate evidence to close out the majority of my regulatory queries for the PRZ for the purpose of GDA (Ref. 49). I identified a few matters to follow-up in licensing.

4.2.1.9 Review of the Steam Generator (SG)

72. There are two Steam Generators (SGs) per **AP1000** plant. Its function is to transfer heat from the primary reactor coolant water into the boiling two-phase steam mixture on its secondary side. The SGs separate off the dry steam on the secondary side and supply this to the turbines to generate electricity from the plant. The SGs contain a large, continuously replenished, feedwater inventory that is available as a heat sink in the event of primary-side high temperature transients or emergency conditions.

73. I selected several design documents for my sampling, (Refs. 24 to 32). My review focussed on the design aspects relevant to testing the evidence of the Westinghouse demonstration of ASME III design compliance that had proved problematic for the RPV and PRZ. My assessment was further informed by the TSC report (Ref. 54).

74. In general, I found the Westinghouse SG design documentation was compiled competently. Most of the comments I raised were minor and satisfactorily addressed by Westinghouse (Ref. 51). My main comments related to modelling assumptions and the use of axisymmetric analysis of nozzles, manways and other features on cylinders and heads with non-axisymmetric geometry. My comments were therefore similar to those important questions regarding methodology I raised for the PRZ. Westinghouse held the view that its analysis was conservative and there were adequate margins. This is a generic matter point that could not be fully resolved within the GDA timescale and so is taken forward to licensing. This is the subject of my Assessment Finding CP-AF-AP1000-SI-13, see Annex 1.

75. A specific comment for the SG related to the effect of the stiffness of the feedwater (FW) ring on the stresses and displacements in the main feedwater (MFW) nozzle thermal sleeve. The MFW nozzle is modelled using a 2D axisymmetric model and the FW ring is modelled with a separate 3D model. Westinghouse took the displacements from the 2D axisymmetric model of the MFW and applied them to the 3D model of the FW ring. The crux of my comment was how the 3D loads (three orthogonal forces and moments) were applied to the 2D axisymmetric model of the MFW nozzle (which has only two forces and one moment). Westinghouse provided a detailed response, explaining how the “missing” degrees of freedom (loads) in the 2D axisymmetric model of the MFW nozzle were represented in the FEA software (ANSYS) that they used. I was satisfied with the Westinghouse response and content that the analysis is conservative.

76. Westinghouse provided adequate evidence to close out the majority of my regulatory queries for the SG for the purpose of GDA (Ref. 51). I identified the matter to follow-up in licensing (Assessment Finding CP-AF-AP1000-SI-13).

4.2.1.10 Review of the Passive Residual Heat Removal Heat Exchanger (PRHR HX)

77. The PRHR HX is connected via a normally open inlet line to one of the hot legs of the reactor primary coolant system and provides passive emergency core decay heat removal. The PRHR HX outlet is connected to the Steam Generator (SG) cold leg plenum and is normally closed. The PRHR HX is submerged within the In-Containment Refuelling Water Storage Tank (IRWST), which provides the heat-sink for

the PRHR HX. When needed, primary reactor coolant circulates by natural convection around the PRHR HX headers and tubes, discharging heat to the IRWST water.

78. I selected various documents for review (Refs. 33 to 37). My review focussed on the design aspects relevant to testing the evidence of the Westinghouse demonstration of ASME III design compliance that had proved problematic for the RPV and PRZ. My assessment was further informed by the TSC report (Ref. 55).

79. In general, the Westinghouse design documentation for the PRHR HX appeared to be compiled competently. I raised four comments relating to minor discrepancies in the calculations and reports that do not affect the outcome of the assessments, (Ref. 52). Although these were minor errors, its presence in verified and approved design documentation was unexpected.

80. One comment related to a slightly incorrect application of the 'Design by Rule' Clause in ASME III NB-3334.1, which does not affect the outcome from the assessment, but again was unexpected in verified and approved design documentation.

81. A final comment for the PRHR HX relates to the Westinghouse corrosion allowances. Where necessary, pressure vessels are designed with an additional thickness allowance to guard against corrosion, such that the remaining material thickness after a lifetime of corrosion is still adequate to meet the design code limits.

82. I asked Westinghouse to provide relevant evidence from operating plant to justify its corrosion allowances for the PRHR HX materials. Westinghouse replied that, for the austenitic stainless steel parts of the PRHR HX, including clad areas and supports, along with the low-alloy nickel tubes, the corrosion allowance is standard for these corrosion-resistant materials and that this is supported by corrosion studies and operating experience. The PRHR HX components are exposed internally to RCS water and externally to IRWST water. I noted that there may be un-clad low carbon alloy steel exposed to the IRWST water which may occasionally boil. The ASME design assessment indicated reasonable margins on basic strength to the code allowable limits, so I am satisfied with the design for the purposes of the GDA. However, for licencing I expect further evidence to justify the corrosion allowance for the PRHR HX materials, and especially the low carbon alloy steel components, which are more vulnerable to corrosion than the other corrosion-resistant parts.

83. Overall, I was satisfied that Westinghouse had provided adequate responses to my regulatory queries for PRHR HX for the purposes of GDA (Ref. 52). I identified one matter to follow-up in licensing. This is the subject of my Assessment Finding CP-AF-AP1000-SI-15, see Annex 1.

4.2.1.11 ASME III Design Documentation Verification and Approval Arrangements

84. In support of gathering evidence for GDA Issue GI-AP1000-SI-05 Action 2, the scope of my inspection of Westinghouse included consideration of the verification and approval arrangements that had underpinned its nuclear safety evaluation. I concluded that the level of documentation verification and approval was commensurate with the intent of the US regulations (Ref. 45). Notably, pending the completion of detailed evaluation, the reporting arrangements under 10 CFR Part 21 allow best estimate approaches and engineering judgement to be invoked to inform the reporting decision.

85. I sought evidence of verification and approval of the preliminary calculations that had underpinned the Westinghouse nuclear safety evaluation. Westinghouse provided verbal assurance, but detailed documented evidence of verification to the expectations of UK relevant good practice was not yet available. This was unexpected from a UK safety case context, where the UK expectation was for adequate, proportionate and sufficient verification/approval of design documentation. I viewed this as equivalent to

'claims and arguments' in UK safety case space. I indicated that for a UK safety case the expectation is that claims, arguments and evidence is presented to justify that the component design was commensurate with reducing risks so far as is reasonably practicable.

86. As Westinghouse had committed to updating its ASME III design documentation for all the major vessels in the **AP1000** reactor, for which ONR undertook sampling reviews to inform the GDA closure, I judged that the shortfall against the UK expectations from Westinghouse nuclear safety evaluation to the US regulations would not affect the way forward for closure of GI-AP1000-SI01 A1. However, I indicated to Westinghouse that, for highest reliability components, this level of verification and approval would be problematic for licensing in the UK. I highlighted to Westinghouse that ONR expects a proportionate approach to verification and approval of design documentation taking cognizance of the significance to safety (Section 4.2.2)

4.2.1.12 Improvement Initiatives

87. In response to the errors of omission relating to the engineering evaluations for GI-AP1000-SI-05, i.e. those relating to inconsistencies between evaluations using 'Design by Rule' and 'Design by Analysis' methods, Westinghouse identified the importance of satisfying the intent of all the design code requirements. This was particularly important in considering the interaction between adjacent discontinuities e.g. nozzle openings. Westinghouse proposed two improvements (Ref. 48):

- training covering ASME III design evaluation.
- an ASME III 'check list' as a job aid for updating its design procedures.

88. Westinghouse explained that the training would provide information on the principles/concepts behind the various ASME Code requirements so that they can be properly evaluated in component analyses (Ref. 53). Analysts would therefore be better prepared to select appropriate stresses and locations to evaluate against the various code limits. Emphasis would also be placed on the various local primary stress intensity (P_L) aspects, such as what constitutes a P_L stress and how it is limited in both magnitude and extent.

89. The training would also cover key aspects of both the 'Design by Rule (NB-3100) and 'Design by Analysis (NB-3200) sections of the ASME III code. More importantly, it covered the link between the two sections including which specific Design by Analysis sections must be shown to be satisfied when certain 'Design by Rule' sections cannot be satisfied (e.g. meeting all NB-3213.10 requirements when area of reinforcement rules (NB-3330) are not satisfied)

90. The ASME III 'check list' was a more formal verification tool that provides a visual/ rigorous representation of the ASME Code requirements to ensure all items have been addressed. The aim was to ensure that both the author and verifier were considering all ASME Code requirements when performing an analysis. Some ASME III design code requirements may not be applicable for a particular analysis, but the checklist allows all requirements to at least be considered for applicability and the recording of assumptions.

91. I established with Westinghouse that training and working to the revised ASME III design compliance procedures would now be mandatory for authors and verifiers (Ref. 48). I welcome and support the Westinghouse initiatives. My regulatory judgement is also informed by the following observations:

- The errors and inconsistencies found in the ASME III design documentation suggest weakness in Westinghouse verification and approval process within the structural integrity discipline.

- The evidence suggests there is scope to improve the overall oversight and technical governance to ensure integration of component design and that the individual design analyses fit together.

4.2.2 Assessment of GDA Issue SI-05 Action 2

92. A key underpinning assumption of the ONR sampling approach to the review of safety cases is that the Requesting Party has in place arrangements that will ensure that documentation supporting the safety case is produced to a consistently high quality. This assumption is tested through ONR assessment of key processes that control the production of safety case documentation, these processes include;

- Competency of engineers, safety case authors and managers, including familiarity with UK specific expectations;
- Management of interdependencies between different technical disciplines, including the interface between design and safety case production;
- Design development, including design review;
- Safety case production;
- Management of the configuration baseline, including management of changes;
- Verification and approval of documentation for inclusion into the GDA submission.

93. In addition to undertaking assessments of these key processes, ONR also monitor the on-going quality of documentation submitted to ONR.

94. The observations described in Section 4.2.1 of this report concerning the identification of errors in Westinghouse structural integrity reports, and the “high burden of proof” required for these components meant doubt was cast on the adequacy of the Westinghouse safety case documentation verification and approvals process to provide the expected level of consistent high quality. ONR was able to issue an IDAC on the basis that the errors were known and subject to remedial action, however GDA Issue GI-AP-1000-SI-05 A2 was raised to provide confidence that there would not be similar errors elsewhere in the GDA safety case submission:

“Provide evidence that there will not be similar errors elsewhere in the design support documentation.

ONR has identified errors on a sample review of the design calculations. The calculations were verified and issued, and referred to within the GDA submissions, but not approved as the formal issue (Rev 0) of the report. In this circumstance the formal issue of the report corrected the errors in the calculational route of ‘design by rule’, and in this case, even if error had not been detected, the design was still secure because the design route ‘design by analysis” had also been followed.
Nevertheless, since a sample review identified significant errors in a verified document, evidence is required to demonstrate that the process in raising design reports to Rev 0 is sufficiently robust to ensure that errors missed by the author and verifier of the earlier revisions will be reliably detected.

Activities by Westinghouse should comprise:

- *Provide evidence that the process for raising verified documents to Revision 0 is sufficiently robust.*
- *Provide adequate responses to any questions arising from assessment by ONR of the response*

With agreement from the regulator this action may be completed by alternative means.”

95. I undertook a preliminary inspection of the Westinghouse verification and approval arrangements within the structural integrity discipline. I raised questions relating to the root cause(s) of the difficulties, the lessons learnt, changes made and hence the basis for confidence that a repetition was unlikely (Ref. 56). Westinghouse held the view that at GDA Step 4 there was greater dependency on partner organisations. Westinghouse had subsequently introduced new procedures to establish formal processes for reviewing, accepting and archiving **AP1000** reactor design partner documents. Westinghouse considered it unlikely that errors in the design documentation would now be uncorrected because more work was done 'in-house' and subject to internal design reviews and updates. I indicated to Westinghouse that I needed to gain confidence that mistakes were unlikely to reoccur based on the understanding of the circumstances. To inform my judgement I would review a sample of the updated design documentation for the RPV and PRZ for closure of GDA Issue GI-AP-1000-SI-05 A2 (Ref. 57).

96. As described above (section 4.2.1.2), the review of the documentation submitted by Westinghouse in response to GDA Issue GI-AP-1000-SI-05 A1 identified that:

- The overall response did not resolve the majority of the most important points raised in the Step 4 assessment report (these are known as errors of omission);
- Some of the documentation contained inconsistencies and technical errors in the calculations (these are known as errors of commission).

97. In view of the potential implications for nuclear safety, Westinghouse responded to these ONR observations by initiating its corrective action, prevention and learning (CAPAL) process, which in turn initiated the following two causal analyses;

- Root Cause Analysis – "Failure to address ONR Regulatory issues on ASME Section III Calculations for Structural Integrity" (100377138 Revision 2) (referred to hereafter as the RCA report). This report details the investigation into why Westinghouse failed to capture and address the GDA Step 4 Assessment Report findings provided by the ONR in 2011.

- Apparent Cause Analysis – "Deficiency in ASME Calculations for AP1000" (100382797 Revision 6-01-16) (referred to hereafter as the ACA report). This report details the investigation into why the revised structural integrity documentation submitted to the ONR in 2015 contained inconsistencies and technical errors in the calculations.

98. To restore confidence in Westinghouse's quality assurance and management arrangements relating to design code compliance, I requested a statement on the nuclear safety significance, the status of the CAPAL, and the status of the Westinghouse investigations (causal analyses) and a recovery plan (Ref. 42). In August 2016 ONR received the Westinghouse Recovery Plan Report [1] which brought together the two causal analyses and summarised the Westinghouse plans for recovery.

99. This part of the report is split into three sections which describe my assessment of the Westinghouse response to GI-AP-1000-SI-05 A2:

- Review of the Westinghouse Recovery Plan Report.

- Inspection to demonstrate compliance with ASME III design criteria.

- Review of response to inspection findings (RQ-AP-1000-1769).

4.2.2.1 Review of the Westinghouse Recovery Plan Report

100. The purpose of Ref. 58 was to provide the root cause analysis, apparent cause analysis, CAPALs, and associated information related to the Westinghouse investigation into the shortfalls, identified by ONR, in documentation submitted in response to GI-AP-1000-SI-05 A1.

101. The objective of my review of this report was to determine whether it provided suitable evidence to support closure of GI-AP1000-SI-05 A2. To do this I needed to determine whether it represented an independent and searching review across an adequate sample of the organisation, with collation of adequate evidence to support the conclusions. In addition, I needed to understand whether suitable improvement actions had been identified and implemented.

102. The report comprised of an introductory section followed by the complete RCA and ACA reports.

103. The RCA report identified the investigation team and contained; the problem statement, investigation scope, description of key events, description of the root and contributing causes attributed to the event and an extent of condition statement. The RCA report also included the nuclear safety evaluation which is discussed in section 4.2.1.3 of this report. The identified causes included non-conservative decision making with respect to tracking queries from parties external to Westinghouse and inadequate checks and balances in place to ensure queries were appropriately managed, including query tracking.

104. The report concluded that no new corrective actions to prevent recurrence were needed to address these root causes as several prior root cause analyses, undertaken between when the problem occurred in 2011 and when the documentation was re-submitted in 2016, had identified similar issues regarding the safety culture and mindset with respect to code compliance and had already initiated significant actions (including corrective actions to prevent recurrence) to address the concerns and to change the mindset and culture of Westinghouse and all staff working on the **AP1000** plant project. Several actions were taken to address the contributing causes to the weaknesses in the GDA processes.

105. The extent of condition describes whether similar problems may be present elsewhere within the organisation; it stated that there was the potential that for the UK GDA effort there were other items from the Step 4 process that were not adequately captured and tracked. The report concluded that an additional review of the GDA Step 4 Assessment Report was needed to determine whether all the feedback from the ONR had been adequately captured.

106. The ACA report identified the investigator and contained; the problem statement, description of key events, description of the apparent causes attributed to the event, an extent of condition statement and a list of the actions needed to prevent reoccurrence.

107. The report identified ten apparent causes for the errors in calculation, each of which was linked to an initiating action. The extent of condition statement made it clear that the investigation had been limited to documentation and calculations undertaken by the Structural Integrity team. The report concluded that eight corrective actions were needed to prevent reoccurrence of similar errors in future structural integrity calculations.

108. In my opinion, the Westinghouse Recovery Plan Report, RCA report and ACA report each demonstrated that a systematic approach had been taken to the analysis of the events which lead to the shortfalls which had been identified by ONR. The description of due process followed in undertaking the investigations met my expectations for independence of the investigation team, investigation techniques employed and collection and recording of evidence. The identification of underlying causes was

compelling and indicated an open and honest approach to reporting. Generally speaking the corrective and preventive actions identified would, in my opinion, lead to improved performance in the future.

109. There were, however, gaps in the extent of scope and evidence presented in the Westinghouse Recovery Plan Report.

110. In the case of the RCA report, it was concluded that the root cause for the errors of omission was weakness in the safety culture and the assertion was made that improvement action to address this deficiency had already been made through the intervention of prior root cause analyses. The RCA report contained reference to these prior investigations and a description of the work that had been carried out. The effectiveness of the prior investigations was not reviewed during this assessment, but the Westinghouse RCA process requires an effectiveness review of the corrective actions.

111. With respect to the ACA, the extent of scope of the analyses carried out in response to the errors of commission was in accordance with Westinghouse processes, but was limited in two ways. Firstly, the analyses considered only the design process, with little attention paid to input information and assumptions and none to verification and approval. This means that only a part of the process map which delivers documentation to ONR for review had been considered. Secondly the analyses considered only the Structural Integrity technical discipline, there was no consideration given to the possibility that the same problems might exist in other technical disciplines, despite the fact that they potentially follow the same processes and exist within the same business and cultural environment. Thus by limiting the scope of the extent of condition, an opportunity had been lost to demonstrate that all aspects of the GDA production process, in particular the verification and approval arrangements, were effective and consistently implemented across all the Westinghouse departments.

112. In order to ensure a full understanding of the work undertaken by Westinghouse and to communicate the two key points described above, in September 2016 I asked a number of questions of Westinghouse through the issue of RQ-AP1000-1678 (Ref. 59). Westinghouse were able to provide a timely and informative response to these questions (Ref. 60); however (1) it was not possible to effectively communicate the evidence required to substantiate the RCA report assertion by remote sharing of documents and teleconference and (2) the Westinghouse CAPAL implementation was not aligned with the ONR need to collect evidence of the effectiveness of the verification and approvals process across multiple technical disciplines.

113. It was for these two reasons that the decision was made to undertake a site inspection at Westinghouse. This inspection was undertaken by an ONR structural integrity specialist inspector and an ONR MSQA specialist inspector. In addition, the inspection was undertaken with the co-operation of the US NRC, who observed some parts of the ONR inspection whilst undertaking a separate planned inspection of Westinghouse around the same time (Ref. 48).

4.2.2.2 Inspection to Demonstrate Compliance with ASME III Design Criteria

114. The stated objective (Ref. 61) for this inspection was:

"For ONR to gain confidence that Westinghouse response to GI-AP1000-SI-05 and its assessment into Structural Components Code Compliance are comprehensive and have adequately identified and corrected any corporate or systemic issues that might affect future safety case submissions".

"The aim is to seek evidence that WEC have considered the potential impact of shortfalls found in relation to the Structural Components on other elements of the plant/ safety case. The inspection shall focus on some parts of the assessment process undertaken and some perceived gaps in coverage of the assessment."

115. The scope of the inspection covered both GI-AP1000-SI-05 A1 and A2. The findings related to the A1 "Engineering Evaluation" parts of the inspection are described in section 4.2.1.11 of this report. The findings related to the A2 "Management for Safety and Quality Assurance" parts of the inspection are described below.

116. The inspection comprised of three assessment areas ("RCA and ACA Processes", "Nuclear Safety Culture Excellence Plan (NSCEP)" and "Verification and Approval"), which were selected to facilitate the collection of relevant evidence to provide the required confidence in the adequacy of the Westinghouse response to GDA Issue GI-AP1000-SI-05 A2. The evidence sampled during the inspection has been provided by Westinghouse (Ref. 62). The following section comprises for each assessment area; the reason for undertaking this assessment, the key findings identified and my judgement on adequacy.

RCA and ACA Processes

117. These processes were reviewed to gain confidence that the Westinghouse approach to the investigation of events was clearly defined, met the UK regulatory expectations for a Requesting Party and were consistently implemented. During the inspection I was able to interview the process owner and two lead investigators and to attend the Issue Review Committee where potential events are considered and the appropriate inspection technique assigned. I was able to review the process documentation and to sample the historical record of RCA and ACA investigation records. I also specifically reviewed the records supporting the RCA and ACA undertaken in response to the GI-AP1000-SI-05 shortfalls; I was able to interview the RCA lead investigator, gain an insight into the scope of the investigations and review the purpose and progress of the identified improvement actions.

118. It is my view that the RCA process is clearly defined and robustly implemented, both at an individual and organisational level. Segregation of lead analyst from routine duties is a good feature which promotes effective and timely identification of root causes. I felt that internal communication within Westinghouse using the learning clocks was an example of good practice.

119. I am content that the specific GI-AP1000-SI-05 RCA scope, findings and actions were appropriate to identify and correct shortfalls in the conduct of Westinghouse operations at the time that the GDA Issue GI-AP1000-SI-05 was received and decisions on response were first made.

120. The reasons why the GI-AP1000-SI-05 ACA was scoped as it was, to focus only on the Structural Integrity discipline were explained and I am content that this is in accordance with the Westinghouse process. I am also content that the corrective action being taken will reduce the likelihood of similar errors of commission being experienced by this team in the future. I cover the potential wider implications for other GDA technical disciplines in the sections below covering the verification and approval arrangements.

Nuclear Safety Culture Excellence Plan (NSCEP)

121. This part of the inspection was aimed at collecting evidence to substantiate the assertion made in the RCA report that cultural issues present in Westinghouse in 2011 have been addressed and that the organisation safety culture had been significantly improved over the period 2011 to 2016. During the inspection I was able to interview

the Nuclear Safety Culture Director and the NSCEP manager. The NSCEP was presented to me; it includes five primary improvement projects, each of which has a senior vice president executive sponsor, thus demonstrating Westinghouse's commitment to make these improvements:

- Organizational Structure
- Leadership Behaviours & Development
- Rewards & Recognition
- Best Practices & Communications
- Safety Advocate

122. The programme addresses organisational concerns using both a top down approach (e.g. addressing the behaviours of leaders) and bottom up approach (e.g. providing routine awareness initiatives such as "safety advocate").

123. An early improvement was to put in place an appropriate safety culture organisational structure, which included the Nuclear Safety Review Board, which monitors progress of the NSCEP and report on nuclear safety culture health to the Executive Leadership Team; and the Nuclear Safety Culture Monitoring Panel which has now started to generate nuclear safety health scorecards.

124. I was able to ask a number of clarifying questions concerning; the reasons why this programme and its contents had been deemed necessary, what key improvements had been made in the period 2011 to 2016 and how the effectiveness of improvements had been assessed.

125. I was able to trace the actions claimed within the RCA report to programmed initiatives, and I was able to review an example of the Quarterly Health Check report that is presented to the Nuclear Safety Committee Review Board to demonstrate progress against the initiatives.

126. I am content that the improvement actions to address the errors of omission (failure to address the GDA Step 4 Assessment comments) identified by the RCA report are within the scope of the NSCEP. I understand Westinghouse's conclusion in the RCA report that no additional action is necessary due to the identification and inclusion of these actions in the NSCEP. I have seen evidence that the improvements have been embedded and have been judged to be effective by Westinghouse senior leadership. It is therefore my judgement that adequate evidence exists to substantiate the assertion in the RCA report that safety cultural improvement has been made, notwithstanding the fact that cultural improvement is a long term and continually evolving goal, and that making, sustaining and demonstrating improvement is extremely difficult to quantify.

Verification and Approval

127. The fact that the Westinghouse CAPAL implementation was not aligned with the ONR need to collect evidence of the effectiveness of the verification and approvals process across multiple technical disciplines necessitated that these arrangements be subject to detailed ONR examination during this inspection. My approach was to assess three key elements of the arrangements:

- Processes and systems
- Organisation
- Output quality

128. During the inspection I interviewed the following key staff:

- Manager – New Plants & Major Projects, Business & Project Development Quality;

- Principal Quality Engineer – UK Moorside Project;
- Service Manager – Information & Document Management Services;
- A number of engineers, including accredited Professional Engineers, who had produced or overseen production of verification reports which I had sampled.

129. I was able to review the documented process and procedures and received a demonstration of the Electronic Document Management System. Verification and approval procedures are available at three levels. Level 1 is an overarching process flow showing linkage to other processes, Level 2 is the procedure indicating the mandatory activities and Level 3 is work instructions and job aides that provide clarity and guidance and are in some cases specific to particular tasks or technical areas.

130. In reviewing the verification arrangements I noted that they did not require a proportional level of verification depending on the safety significance of the plant. As this deviates from relevant good practice in the UK, I discussed this matter with one of the professional engineers that had overseen verification activities. Professional Engineers are independently certified by individual state boards in the United States against the ASME standards and have a legal duty within the USA to ensure compliance with these ASME standards. They are generally seen as being the most technically senior engineers. His answer regarding proportionality indicated that he was aware of the principle and in practice would give more attention to the verification of more safety significant plant; however this was not a requirement under US law and was not a requirement of the Westinghouse process.

131. During the inspection I enquired into the use of alternative calculation techniques for verification of high safety significance components. An example of use of alternate calculations was provided, although this was described as rare. There is no requirement to use this technique within the Westinghouse procedures and it is left to the judgement of the engineer undertaking the verification if this is considered to be necessary.

132. The Westinghouse engineering/ design team organisation follows the usual industry approach of senior engineers leading teams of more junior members. This structure has the inevitable (and usually positive) consequence of encouraging teams to share learning, experience and working practices. The Westinghouse procedures require that verifier(s) to be independent of the work being reviewed to reduce the risk of a common cause failure. Conversely, as the verifier(s) may be drawn from the same limited pool of resource as the authors, this can promote a common cause of failure where alternate techniques and approaches are not always used.

133. An important part of the inspection was to sample the records generated by the documented arrangements. These records included general evidence of completion of the process, such as competence and appointment records.

134. I also sampled seven GDA submission documents that had undergone verification. These were selected from across five different technical disciplines to provide a comparative assessment of consistency.

135. The sample included three structural integrity documents. These included evidence that the mandatory process had been conducted, with the appropriate verification report form and signatures attached. However there was no evidence indicating the level of challenge presented by the verifier. By contrast the sampled fault schedule and control & instrumentation documents gave a full picture of the verification strategy and demonstrated challenge level through completion of “query/ response” forms. I considered these last two examples to represent good practice.

136. Overall it is my view (Ref. 48) that the verification and approval arrangements are clear, comprehensive and meet ONR expectations. My examination of evidence

indicates that these arrangements “enable excellence” in that some of the examples provided showed the expected level of verification.

137. The limited sample of verification activities demonstrated compliance with the Westinghouse arrangements. However, they showed considerable variation in the apparent depth, challenge and clarity of reporting presented by the verifier. I noted that of the documents sampled, those generated by the Structural Integrity team showed the least evidence of effective verification.

138. I concluded, therefore, that the inspection had not generated sufficient evidence to demonstrate that GDA Issue GI-AP1000-SI-05 A2 could be closed. In order to close the action I would require additional evidence that implementation of the verification process was consistent within all Westinghouse engineering/ design teams. Given that I now had confidence in the Westinghouse investigation and learning processes, I was content to allow this evidence to be generated internally by the Requesting Party, thus I raised a Regulatory Query (Ref. 63) to include the following action:

> *“Westinghouse to review the implementation of Level 1, 2 and 3 verification arrangements to determine to what extend implementation is uniform across the different technical disciplines, consider what risks are presented by any inconsistency and whether improvement is needed.”*

4.2.2.3 Review of Response to Inspection Findings (RQ-AP1000-1769)

139. In response to the RQ following the Westinghouse site inspection requesting that it undertook a review of the implementation of its verification arrangements across all the technical disciplines, Westinghouse provided Ref. 64. This report described how it had undertaken its review, the range of information and data considered, its findings and its recommended improvement actions.

140. The approach consisted of performing an assessment of design documentation to gather new data as well as reviewing existing data to determine whether inconsistencies in the verification arrangements or its implementation had been identified previously.

141. The assessment of design documentation was carried out by staff with experience from: **AP1000** reactor international licensing, **AP1000** reactor systems design, radiation protection and quality assurance. The team also included a NuGeneration Limited (NuGen) engineer acting as a representative of a prospective holder of a nuclear site licence to install and operate **AP1000** reactor units at the Moorside site in Cumbria, UK. The team considered 92 documents which were selected from across all the technical disciplines that had contributed to the UK AP1000 project. The review of each document considered both compliance with Westinghouse verification processes and whether the verification report demonstrated good practice with regards the presentation of evidence of challenge.

142. The review of existing data considered external assessments by the Nuclear Utilities Procurement Issues Committee and Lloyd’s Register Quality Assurance ISO 9001 and internal assessments including Internal Audits, Self-Assessments and CAPAL Sampling.

143. The report identified that there was good evidence that implementation of the Westinghouse verification arrangements was consistent across all the technical disciplines, though there were some noted inconsistencies. The report confirmed that there were a larger percentage of Structural Integrity documents identified as having minimum compliance compared to other engineering disciplines but with the larger sample size, this was not by a significant amount.

144. The overall conclusion was that three minor improvement opportunities had been identified. However they did not lead to significant risks for the overall design programme or the health and safety of the public.

145. I have reviewed this report and have held discussions with the author and some of the members of the assessment team, during which I posed queries and challenges to its approach and conclusions which they answered well.

146. It is my opinion that:

- The challenge presented by the assessment met my expectations. In particular the inclusion of the NuGen engineer demonstrated this challenge and the openness of Westinghouse to present accurate conclusions.
- The sample size was good – larger than expected and provided an effective representation of overall performance.
- The analysis was effective in identifying improvement.

147. Overall, I am content that Westinghouse has now been able to provide adequate evidence that implementation of the verification and approval processes is consistent across all the technical disciplines. I agree with the conclusion that the inconsistencies do not represent a risk to programme delivery or to the adequacy of the GDA submission.

4.2.3 Key Assessment Considerations and Regulatory Judgements

4.2.3.1 Engineering Evaluation

148. A demonstration that components comply with an established nuclear design code makes a significant contribution to underpinning a claim for highest reliability (SAP EMC.1 to EMC.3 and ECS.2). ONR's expectations are based on 'high burden of proof' because nuclear safety is entirely dependent on the structural integrity case when highest reliability is claimed.

149. At GDA Step 4, ONR supported by EASL, identified a number of comments relating to the Westinghouse demonstration of ASME III design compliance for the RPV and PRZ.

150. Post GDA Step 4, I sampled several Westinghouse documents related to the ASME III design and analysis of major vessels in the **AP1000** plant. My review concluded that the majority of the most significant comments raised at GDA Step 4 for the RPV and PRZ were extant. This resulted in uncertainty in the demonstration of ASME III compliance. There was also insufficient evidence to support closure of GI-AP1000-SI-05.

151. Westinghouse accepted my conclusion without challenge and responded constructively with several recovery actions. Taking cognisance of the available margins, using scoping estimates and by invoking engineering judgements, Westinghouse concluded that the errors and inconsistencies identified by ONR would not result in a 'substantial safety hazard' and so were not formally reportable to the US NRC under 10 CFR Part 21. However to support the closure of GI-AP1000-SI-05, Westinghouse committed to the re-evaluation of all ASME III Class 1 vessels.

152. I viewed the Westinghouse response to my conclusions post GDA Step 4 as proportionate and appropriate.

153. For the engineering evaluation, I sought adequate responses to my regulatory queries raised against the RPV and PRZ ASME III design documentation. I also broadened my review and raised regulatory queries through sampling of the ASME III design

documentation for the SG and the PRHR HX vessels. My reviews focused on the resolution of the key points to emerge from GDA Step 4.

154. Westinghouse readily addressed the majority of my comments raised against the RPV ASME III design documentation through adequate responses to my questions. However, resolution of certain comments required several regulatory interventions with Westinghouse. I subsequently gained adequate evidence to close my regulatory queries for the RPV for the purposes of GDA. I identified that although the RPV CRDM penetrations and vent pipe sleeves were code compliant, Westinghouse invoked more advanced methods to justify its structural integrity. The CRDM penetrations are also at risk of PWSCC, this aspect is outside the scope of the GDA Issue GI-AP1000-SI-05, I therefore raise the following assessment finding:

CP-AF-AP1000-SI-12 – The Licensee shall review developments in design and material selection to ensure the risks to structural integrity for the RPV CRDM penetrations and vent pipe sleeves are reduced so far as is reasonably practicable.

155. Similarly, Westinghouse readily addressed the majority of my comments raised against the PRZ ASME III design documentation through adequate responses to my questions. However, resolution of certain comments required several regulatory interventions with Westinghouse.

156. Westinghouse went to considerable effort to address my comments relating to the interaction between the PRZ design features. Westinghouse claimed there was sufficient margin to demonstrate compliance with the ASME III code. However, Westinghouse was unable to provide further validation to demonstrate its 2D analysis route provided bounding stresses within the GDA timescale. This matter is the subject of Assessment Finding CP-AF-AP1000-SI-13, see Annex 1.

157. Westinghouse provided adequate evidence to close out my regulatory queries for the PRZ for the purposes of the GDA. I identified the following matters to follow-up in licensing:

158. Westinghouse was unable to provide sufficient validation to demonstrate its 2D analysis route provides conservative stresses when representing non-axisymmetric structures and loadings. Westinghouse is confident there is sufficient unused margin within the PRZ design to absorb any differences and demonstrate continued compliance with the ASME III Code. I view this as a generic matter that could not be fully resolved within the GDA timescale and so raise the following finding to progress:

CP-AF-AP1000-SI-13 – The licensee shall ensure that adequate evidence is provided to demonstrate that the ASME III analysis methods adopted for HSS and Class 1 components provides conservative stresses.

159. Westinghouse carried out a limit analysis of the PRZ Lower Head, Support Pads and Shell to show ASME III compliance with a small margin of 7% on design internal pressure. Since Westinghouse has shown the vessel to be code compliant, albeit with a small margin, I am content with this response for the purposes of a demonstration of ASME III design compliance for GDA. However, the loadings may change during site licensing and so I will seek confirmation that the inputs remain bounding and conservative and that adequate margins are demonstrated to substantiate the structural integrity case for the PRZ and other HSS or ASME III Class 1 vessels through-life.

CP-AF-AP1000-SI-14 – The licensee shall demonstrate that if the loadings on any HSS or ASME III Class 1 vessel are revised to support licensing its ASME III design analysis remains valid and conservative to substantiate the structural integrity case through-life.

160. The updated ASME III design documentation for the SG and PRHR HX clearly benefited from the Westinghouse improvement initiatives. In general, I found the ASME III design documentation for the SG and PRHX HX of good standard. Westinghouse provided adequate responses to close out my regulatory queries for the SG and PRHR HX for the purposes of the GDA. I identified one matter for the SG to follow-up for licensing this is captured in CP-AF-AP1000-SI-13. I also identified one matter to follow-up in licensing for the PRHR HX:

161. The ASME III design assessment of the PRHR HX indicated reasonable margins on basic strength for the purposes of the GDA. However, to support the through-life case, I expect further evidence to justify the corrosion allowance for the PRHR HX materials:

CP-AF-AP1000-SI-15 – The licensee shall provide evidence to justify the corrosion allowances for the PRHR HX materials

162. In summary, Westinghouse provided adequate responses to my regulatory queries relating to the RPV and PRZ to close-out RQ-AP1000-1620 and RQ-AP1000-1621. In so doing, Westinghouse addressed the outstanding comments from GDA step 4. There are a few new comments that require resolution in the detailed design stage. I have raised assessment findings for the licensee to address during licensing.

163. The updated design documentation for the SG and PRHR HX was compiled competently. I raised a limited number of comments following my extended review of the ASME III Class 1 design documentation.

164. Overall, Westinghouse provided adequate responses to my regulator queries to restore my confidence in the engineering substantiation for the ASME III Class 1 pressure vessels.

4.2.3.2 Verification and Approval of ASME III Design Documentation

165. ONR expects a proportionate approach to the development of the safety case and its supporting design documentation. To underpin highest reliability claims, robust verification and approval arrangements are expected (SAP SC.1 and EMC.3).

166. The Westinghouse design documentation verification and approval processes were the subject of a joint ONR structural integrity and MSQA inspection, described earlier. In terms of the structural integrity discipline, I draw the following conclusions from the Westinghouse engineering evaluation work to address GI-AP1000-SI-05 Action 1:

167. Multiple errors and inconsistencies in the ASME III design documentation were identified during GDA Step 4, which were initially not corrected by Westinghouse. This led to uncertainty in the demonstration of ASME III design compliance for major pressure vessels in the **AP1000** plant.

168. I resolved my regulatory queries for the RPV and PRZ with Westinghouse for the purposes of the GDA. However, this was not without multiple regulatory interactions, which included the closure of comments akin to points raised at GDA Step 4.

169. I acknowledge that Westinghouse has introduced additional training and a useful checklist as improvement initiatives. Following these initiatives, there was evidence of improvement in the ASME III design documentation for the SG and PRHR HX reviews. Nevertheless, errors, albeit not significant, were identified in the recently supplied verified and approved design documentation for the PRHR HX. I am aware of the importance of the achievement of compliance with an established nuclear code for UK Class 1 vessels, and the even higher burden of proof expected to underpin a claim for highest reliability. I therefore consider that further measures are needed to ensure that robust verification and approval arrangements are developed and implemented to

support the detailed design licensing. However, for the purposes of the GDA it was also necessary to establish the veracity of the Westinghouse verification and approval arrangements both within the structural integrity discipline and other technical disciplines that support the PCSR. I cover these aspects in Section 4.2.2 of this report

4.2.3.3 Management for Safety and Quality Assurance

170. As the assessment of the Westinghouse response to GDA Issue GI-AP1000-SI-05 A2 has unfolded, three key areas of MSQA were reviewed:

- Demonstration that Westinghouse investigation, learning and improvement processes (ACA and RCA processes) are effective in identifying the root cause of the shortfalls (errors, inconsistencies and omissions) found in documentation submitted for GDA.
- Demonstration that the improvement actions taken to prevent recurrence of the shortfalls found in GDA documentation have been effective. In particular, the demonstration that effective improvements have been made to nuclear safety culture in the period 2011 to 2016.
- Demonstration that the Westinghouse verification and approval processes meet UK expectations, are effective and are implemented consistently across all the GDA technical disciplines.

171. My assessment of the Westinghouse investigation, learning and improvement processes provided me with evidence that these arrangements met UK expectations for independence, competency and freedom of the investigation team to undertake a meaningful investigation. The processes provided for suitable collection and recording of evidence and analysis of this evidence to identify root causes. I was able to see effective implementation of the processes and the identification of meaningful root and contributing causes and improvement actions to prevent recurrence. Examination of a sample of investigation reports, including those raised in response to the shortfalls in Structural Integrity documentation, gave me confidence that the processes were being implemented consistently by Westinghouse within the **AP1000** reactor project.

172. My assessment of the Westinghouse arrangements for management of improvement actions and communication of learning provided me with evidence that, in general, these arrangements are effective. I considered that the Westinghouse use of databases to manage actions and learning and in particular the use of learning clocks was good practice. I focused a significant amount of my inspection time to review the Westinghouse assertion that improvements had been made to nuclear safety culture since the errors of omission had occurred in 2011. I am content that Westinghouse has implemented, and made significant progress against, a nuclear safety culture improvement programme. The programme addresses organisational concerns using a comprehensive top down and bottom up approach. It is my judgement that this approach and programme meet UK expectations for an effective and capable organisation. Projects within the programme are sponsored at the vice president level and progress is monitored and reported to senior leadership, thus demonstrating a suitable level of commitment to making the required improvements. Though it is very difficult to demonstrate quantitatively that nuclear safety culture has actually improved, the evidence provided gives me confidence that improvement has been achieved and will continue to be made.

173. My assessment of the Westinghouse verification and approval processes, through inspection and the subsequent request to Westinghouse to carry out an additional detailed review, provided evidence that its arrangements were clear, comprehensive and consistently implemented. The level of verification demonstrated generally meets UK expectations and the relevant good practice applied by UK operators. The main

shortfall against relevant good practice is that the level of verification is not graded depending on the safety significance of the claim being made (IAEA GSR Part 2, see Table 3). This is discussed next.

174. The need for the level of verification to be proportional to the safety claims is important to the development of the safety case for GDA. Irrespective of the stage of the safety case development, the expectation for robust verification is particularly important when the Requesting Party, or licensee, invokes highest reliability claims (SAP EMC.3). It is therefore my judgement that additional measures are necessary to demonstrate that verification undertaken during licensing meet UK expectation and relevant good practice in the application of a graded level of scrutiny and approach depending on the safety significance of the component.

175. The above concern must be considered in relation to the points made in section 4.2.1.11 concerning the history of errors found in documentation supporting high safety significance components (RPV and PRZ) and the need to take an integrated approach to ensure individual component design analyses fit together.

176. I therefore consider that the following Assessment Finding is required:

CP-AF-AP1000-SI-16 – The licensee shall demonstrate that its Licence Condition 17 “management systems” arrangements provide a robust technical governance framework and a graded verification and approvals approach with the highest standard of its graded approach being applied to HSS and High Integrity components.

4.3 Comparison with Standards, Guidance and Relevant Good Practice

177. Section 2.2 of this report identifies standards, guidance and RGP that has informed my assessment. In particular, my assessment has been guided by ONR’s SAPs and TAGs, see

178. Table **1**, Table 2 and 3.

4.4 Assessment Findings

179. During my assessment five items were identified for a future licensee to take forward in its site-specific safety submissions. These are collated in Annex 1. These matters do not undermine the generic safety submission and are primarily concerned with the provision of site specific safety case evidence, which will usually become available as the project progresses through the detailed design, construction and commissioning stages. These items are captured as Assessment Findings.

180. Residual matters are recorded as Assessment Findings if one or more of the following apply:

- site specific information is required to resolve this matter.
- the way to resolve this matter depends on licensee design choices.
- the matter raised is related to operator specific features / aspects / choices.
- the resolution of this matter requires licensee choices on organisational matters.
- to resolve this matter the plant needs to be at some stage of construction / commissioning.

5 CONCLUSIONS

181. This report presents the findings of the assessment of GDA Issue GI-AP1000-SI-05 Revision 7 - Compliance of AP1000 Main Structural Components with ASME III Design Rules, relating to the **AP1000** reactor GDA closure phase.

182. To conclude:

- Westinghouse has adequately demonstrated compliance with the rules of Section III of the ASME Code for the components sampled in my assessment.
- Westinghouse has demonstrated that shortfalls in organisational performance in 2011 are understood and that action has been taken to prevent recurrence.
- Westinghouse has demonstrated that verification and approvals processes are robust and consistently applied in accordance with its internal arrangements.
- Westinghouse verification is not proportionately enhanced for highest reliability components.

183. My judgement is based on the following factors:

- The satisfactory outcome of my detailed assessment of a sample of submissions by Westinghouse as evidence of compliance with the rules of Section III of the ASME Code for the RPV, PRZ, SG and PRHR HX.
- ONR assessment of Westinghouse investigation, learning and improvement processes (ACA and RCA processes).
- ONR assessment of effectiveness of corrective actions, including improvements made to nuclear safety culture.
- ONR assessment of Westinghouse verification and approval processes, including sampling of verification report outcomes.
- Westinghouse enhanced inspection of its verification and approval processes covering all technical disciplines and subsequent improvement action.

184. The Assessment Findings collated in Annex 1 remain for a future licensee to consider and take forward in its site-specific safety submissions. These matters do not undermine the generic safety submission and require licensee input/decision.

185. I consider that, from a structural integrity perspective, the **AP1000** design is suitable for construction in the UK.

186. In summary I am satisfied that GDA Issue GI-AP1000-SI-05 can be closed.

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22. APP-MV20-Z0R-020, AP1000 Pressurizer Spray Nozzle Analysis, Revision 2. (TRIM 2016/71102)

23. APP-MV20-Z0R-101, AP1000 Pressurizer ASME Code Stress Report, Revision 0. (TRIM 2011/79916)

24. APP-MB01-VDR-021, AP1000 Steam General Local Primary Membrane Stress Evaluation (TRIM 2016/484439)

25. APP-MB01-Z0C-105, AP1000 Steam Generator Structural Analysis of Channel Head, Tubesheet, and Lower Shell Complex, Revision 2. (TRIM 2016/484454)

26. APP-MB01-Z0C-102, AP1000 Steam Generator Hot Leg Primary Inlet Nozzle Analysis, Revision 3 (TRIM 2016/484448)

27. APP-MB01-Z0C-104, AP1000 Steam Generator PRHR Nozzle Analysis, Revision 2. (TRIM 2016/484450)

28. APP-MB01-Z0C-105, AP1000 Steam Generator CVS Nozzle Analysis, Revision 2. (TRIM 2016/484454)

29. APP-MB01-Z0C-106, AP1000 Steam Generator Primary Manway Analysis, Revision 1. (TRIM 2016/484461)

30. APP-MB01-Z0C-113, AP1000 Steam Generator Minor Shell Taps Analysis, Revision 5. (TRIM 2016/495372)

31. APP-MB01-Z0C-115, AP1000 Steam Generator Start-up Feedwater Nozzle Analysis, Revision 4. (TRIM 2016/484503)

32. APP-MB01-Z0C-116, AP1000 Steam Generator Feedwater Nozzle and Thermal Sleeve Analysis, Revision 3. (TRIM 2016/484510)

33. APP-ME02-Z0R-100, AP1000 Passive Residual Heat Removal Heat Exchanger Generic Design Report, Revision 2. (TRIM 2016/451234)

34. APP-ME02-Z0C-52, AP1000 PRHR HX Tentative Sizing Calculation, Revision 1. (TRIM 2016/451296)

35. APP-ME02-Z0C-43, AP1000 PRHR HX Header Nozzle Analysis, Revision 3. (TRIM 2016/451261)

36. APP-ME02-Z0C-41, AP1000 PRHR HX Header and Extended Flange Analysis, Revision 4. (TRIM 2016/451250)

37. APP-ME02-Z0C-44, AP1000 PRHR HX Manway Analysis. (TRIM 2016/451286)

38. UKP-GW-GLR-004, Rev. 2, ‘AP1000 UK Structural Integrity Classification’, June 2016, Revision 4. (TRIM 2016/243277)

39. Review of the Stress Analysis for the AP1000, HSE-NII/1322/11, Issue 1, March 2011, (TRIM 2011/136721)

40. AP1000 Stress Analysis, GDA Technical Query TQ-AP1000-1290, 9 March 2011

(TRIM 2011/276546)

41. FNC 50167/43942R, Westinghouse UK-AP1000 Nuclear Reactor New Build, Review of ASME III Compliance for the Main Structural Components on the UK AP1000, Issue 1, March 2017.

42. ONR letter ONR-WEC-0030N, J Bowie, ONR to M Corletti, WEC, 12 May 2016, (TRIM 2016/196027)

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44. ONR-NR-CR-16-231, Revision 0, AP1000 Structural Integrity GDA Issues – Semi-Annual Workshop with WEC – 03 (TRIM 2016/234492)

45. 10 CFR Part 21

46. WEC-REG-998N, Letter from Westinghouse, Westinghouse Structural Integrity Update, 2 June 2016. (TRIM 2016/223282)

47. ONR-NR-CR-16-455, Revision 0, AP1000 Structural Integrity (SI) GDA Issues – Monthly Video Conference with WEC – 12 (TRIM 2016/331064)

48. ONR-NR-CR-16-589 - AP1000 SI05, ONR Inspection of Westinghouse - 27th to 30th September 2016, 14 October 2016. (TRIM 2016/401074)

49. RQ-AP1000-1620, SI05 Action 1 – Demonstration of ASME III Design Compliance, (TRIM 2016/321831)

50. RQ-AP1000-1621, SI05 Action 1 – 'Interpretation' of ASME Section III Clause NB-3213.10.(TRIM 2016/321834)

51. RQ-AP1000-1753, RQ-AP1000-1620 RQ-AP1000-1678 AND TQ-AP1000-SI-1290, (TRIM 2016/495335)

52. RQ-AP1000-1754, AP1000 Structural Integrity – Request for Updated ASME III Design Documentation (SI05) – Passive Residual Heat Removal (PRHR) Heat Exchanger (TRIM 2016/451225)

53. DCP_DCP_008623, Correspondence to ONR on ASME Checklist and Training Lessons Learnt, January 2017, (TRIM 2017/106096)

54. FNC 50167/45466R, Westinghouse UK-AP1000 Nuclear Reactor New Build, Review of ASME III Compliance for the Main Structural Components on the UK AP1000 - Steam Generator, Issue 1, March 2017, (TRIM 2017/122196)

55. FNC 50167/45415R, Westinghouse UK-AP1000 Nuclear Reactor New Build, Review of ASME III Compliance for the Main Structural Components on the UK AP1000 – Passive Residual Heat Removal Heat Exchanger (PRHR HX), Issue 1, March 2017, (TRIM 2017/122199)

56. DCP_DCP_008624, Correspondence to Westinghouse on Review of WEC’s Report Verification and Approval Process for Structural Integrity Design Support Documentation, December 2015(TRIM 2015/489188)

57. ONR-GDA-CR-15-342, Revision 1 AP1000 Structural Integrity GDA Issues – Semi-Annual Workshop with WEC – 02, 11 March 2016 (TRIM 2016/108800)

58. WEC-REG-1160N, Enclosure 1, GI-AP1000-SI-05 RCA ACA - AP1000 Plant Main Structural Components ASME III Code Compliance Recovery Plan Report, 3

August 2016, (TRIM 2016/0310521).

59. RQ-AP1000-1678 - AP1000 Structural Components ASME III Code Compliance Recovery Plan Report, 26 August 2016. (TRIM 2016/339907).

60. WEC-REG-1272N, RQ-AP1000-1678 - AP1000 Structural Components ASME III Code Compliance Recovery Plan Report - Full Response, 19 September 2016. (TRIM 2016/366826)

61. Plan for ONR Inspection of WEC Readiness to close GDA Issue GI-AP1-000-SI-05: Demonstration of Compliance with ASME III Design Criteria, Rev. 0. (TRIM 2016/358179).

62. WEC-REG-1311N, Letter from Westinghouse, Transmittal of Documentation Reviewed During September 2016 ONR Inspection, 4 October 2016. (TRIM 2016/386947).

63. RQ-AP1000-1769, ONR Inspection of Westinghouse - Management for Safety and QA (SI05 Action 2), 25 November 2016. (TRIM 2016/461680).

64. WEC-REG-1555N, Letter from Westinghouse - Transmittal of Full Response to RQ-AP1000-1769, 27 January 2017. (TRIM 2017/39754)

65. UKP-GW-GL-060, AP1000 Design Reference Point for UK GDA, September 2010, Revision 4. (TRIM 2011/353689)

Table 1

Relevant Safety Assessment Principles Considered in the Assessment

SAP No	SAP Title	Description
SC.4	The regulatory assessment of safety cases - Safety case characteristics	A safety case should be accurate, objective and demonstrably complete for its intended purpose.
EMT.2	Engineering principles: maintenance, inspection and testing - Frequency	Structures, systems and components should receive regular and systematic examination, inspection, maintenance and testing as defined in the safety case.
EMT.5	Engineering principles: maintenance, inspection and testing - Procedures	Commissioning and in-service inspection and test procedures should be adopted that ensure initial and continuing quality and reliability.
ECS.2	Safety classification of structures, systems and components	Structures, systems and components that have to deliver safety functions should be identified and classified on the basis of those functions and its significance to safety.
ECS.3.	Engineering principles: safety classification and standards - Codes and standards	Structures, systems and components that are important to safety should be designed, manufactured, constructed, installed, commissioned, quality assured, maintained, tested and inspected to the appropriate codes and standards.
ECS.5	Engineering principles: safety classification and standards - Use of experience, tests or analysis	In the absence of applicable or relevant codes and standards, the results of experience, tests, analysis, or a combination thereof, should be applied to demonstrate that the structure, system or component will perform its safety function(s) to a level commensurate with its classification.
EMC.4	Engineering principles: integrity of metal components and structures: general - procedural control	Design, manufacture and installation activities should be subject to procedural control.
EMC.5	Engineering principles: integrity of metal components and structures: general - Defects	It should be demonstrated that components and structures important to safety are both free from significant defects and are tolerant of defects.
EMC.6	Engineering principles: integrity of metal components and structures: general - Defects	During manufacture and throughout the full lifetime of the facility, there should be means to establish the existence of defects of concern.
EMC.7	Engineering principles: integrity of metal components and structures: design - loadings	The schedule of design loadings (including combinations of loadings) for components and structures, together with conservative estimates of its frequency of occurrence should be used as the basis for design against normal operation, fault and accident conditions. This should include plant transients and tests together with internal and external hazards.
EMC.8	Engineering principles: integrity of metal components and structures: design - providing for examination	Geometry and access arrangements should have regard to the need for examination.
EMC.9	Engineering principles: integrity of metal components and structures: design - product form	The choice of product form of metal components or its constituent parts should have regard to enabling examination and to minimising the number and length of welds in the component.

SAP No	SAP Title	Description
EMC.10	Engineering principles: integrity of metal components and structures: design - weld positions	The positioning of welds should have regard to high-stress locations and adverse environments.
EMC.11	Engineering principles: integrity of metal components and structures: design - failure modes	Failure modes should be gradual and predictable.
EMC.12	Engineering principles: integrity of metal components and structures: design - brittle behaviour	Designs in which components of a metal pressure boundary could exhibit brittle behaviour should be avoided.
EMC.13	Engineering principles: integrity of metal components and structures: manufacture and installation - materials	Materials employed in manufacture and installation should be shown to be suitable for the purpose of enabling an adequate design to be manufactured, operated, examined and maintained throughout the life of the facility.
EMC.14	Engineering principles: integrity of metal components and structures: manufacture and installation - techniques and procedures	Manufacture and installation should use proven techniques and approved procedures to minimise the occurrence of defects that might affect the integrity of components or structures.
EMC.15	Engineering principles: integrity of metal components and structures: manufacture and installation - control of materials	Materials identification, storage and issue should be closely controlled.
EMC.16	Engineering principles: integrity of metal components and structures: manufacture and installation - contamination	The potential for contamination of materials during manufacture and installation should be controlled to ensure the integrity of components and structures is not compromised.
EMC.18	Engineering principles: integrity of metal components and structures: manufacture and installation - third-party inspection	Manufacture and installation should be subject to appropriate third-party independent inspection to confirm that processes and procedures are being followed.
EMC.19	Engineering principles: integrity of metal components and structures: manufacture and installation - non-conformities	Where non-conformities with procedures are judged to have a detrimental effect on integrity or significant defects are found and remedial work is necessary, the remedial work should be carried out to an approved procedure and should apply the same standards as originally intended.
EMC.20	Engineering principles: integrity of metal components and structures: manufacture and installation - records	Detailed records of manufacturing, installation and testing activities should be made and be retained in such a way as to allow review at any time during subsequent operation.
EMC.21	Engineering principles: integrity of metal components and structures: operation - safe operating envelope	Throughout its operating life, components and structures should be operated and controlled within defined limits and conditions (operating rules) derived from the safety case.
EMC.22	Engineering principles: integrity of metal components and structures: operation - material compatibility	Materials compatibility for components should be considered for any operational or maintenance activity.
EMC.23	Engineering principles: integrity of metal components and structures: operation - ductile behaviour	For metal pressure vessels and circuits, particularly ferritic steel items, the operating regime should ensure that they display ductile behaviour when significantly stressed.

SAP No	SAP Title	Description
EMC.24	Engineering principles: integrity of metal components and structures: monitoring - operation	Facility operations should be monitored and recorded to demonstrate compliance with, and to allow review against, the safe operating envelope defined in the safety case (operating rules).
EMC.25	Engineering principles: integrity of metal components and structures: monitoring - leakage	Means should be available to detect, locate, monitor and manage leakages that could indicate the potential for an unsafe condition to develop or give rise to significant radiological consequences.
EMC.26	Engineering principles: integrity of metal components and structures: monitoring -forewarning of failure	Detailed assessment should be carried out where monitoring is claimed to provide forewarning of significant failure.
EMC.27	Engineering principles: integrity of metal components and structures: pre- and in-service examination and testing - examination	Provision should be made for examination that is capable of demonstrating with suitable reliability that the component or structure has been manufactured to an appropriate standard and will be fit for purpose at all times during future operations.
EMC.28	Engineering principles: integrity of metal components and structures: pre- and in-service examination and testing - margins .	An adequate margin should exist between the nature of defects of concern and the capability of the examination to detect and characterise a defect.
EMC.29	Engineering principles: integrity of metal components and structures: pre- and in-service examination and testing - redundancy and diversity	Methods of examination of components and structures should be sufficiently redundant and diverse.
EMC.30	Engineering principles: integrity of metal components and structures: pre- and in-service examination and testing - qualification	Personnel, equipment and procedures should be qualified to an extent consistent with the overall safety case and the contribution of examination to structural integrity aspects of the safety case.
EMC.31	Engineering principles: integrity of metal components and structures: in-service repairs and modifications - repairs and modifications	In-service repairs and modifications should be carefully controlled through a formal procedure for change.
EMC.32	Engineering principles: integrity of metal components and structures: analysis - stress analysis	Stress analysis (including when displacements are the limiting parameter) should be carried out as necessary to support substantiation of the design and should demonstrate the component has an adequate life, taking into account time-dependent degradation processes.
EMC.33	Engineering principles: integrity of metal components and structures: analysis - use of data	The data used in analyses and acceptance criteria should be clearly conservative, taking account of uncertainties in the data and its contribution to the safety case.
EMC.34	Engineering principles: integrity of metal components and structures: analysis - defect sizes	Where high reliability is needed for components and structures and where otherwise appropriate, the sizes of crack-like defects of structural concern should be calculated using verified and validated fracture mechanics methods with verified application.
EAD.1	Engineering principles: ageing and degradation - safe working life	The safe working life of structures, systems and components that are important to safety should be evaluated and defined at the design stage.
EAD.2	Engineering principles: ageing and degradation - lifetime margins	Adequate margins should exist throughout the life of a facility to allow for the effects of materials ageing and degradation processes on structures, systems and components.

SAP No	SAP Title	Description
FP.1	Responsibility for safety	The prime responsibility for safety must rest with the person or organisation responsible for the facilities and activities that give rise to radiation risks.
FP.2	Leadership and management for safety	Effective leadership and management for safety must be established and sustained in organisations concerned with, and facilities and activities that give rise to, radiation risks.
FP.4	Safety assessment	Dutyholders must demonstrate effective understanding and control of the hazards posed by a site or facility through a comprehensive and systematic process of safety assessment.
MS.1	Leadership	Directors, managers and leaders at all levels should focus the organisation on achieving and sustaining high standards of safety and on delivering the characteristics of a high reliability organisation.
MS.2	Capable organisation	The organisation should have the capability to secure and maintain the safety of its undertakings.
MS.3	Decision making	Decisions made at all levels in the organisation affecting safety should be informed, rational, objective, transparent and prudent.
MS.4	Learning	Lessons should be learned from internal and external sources to continually improve leadership, organisational capability, the management system, safety decision making and safety performance.

Table 2

Technical Assessment Guides Considered in the Assessment

Technical Assessment Guide No	Description
NS-INSP-GD-17	Management Systems
NS-TAST-GD-004	Fundamental Principles
NS-TAST-GD-005	Guidance on the Demonstration of ALARP (As Low As Reasonably Practicable)
NS-TAST-GD-006	Deterministic Safety Analysis and The Use of Engineering Principles in Safety Assessment
NS-TAST-GD-009	Examination, Inspection, Maintenance and Testing of Items Important to Safety
NS-TAST-GD-016	Integrity of Metal Components and Structures
NS-TAST-GD-049	Licensee Core and Intelligent Customer Capabilities
NS-TAST-GD-051	The purpose, scope, and content of safety cases
NS-TAST-GD-094	Categorisation of safety functions and classification of structures, systems and components

Table 3

Standards & Guidance Considered in the Assessment

ASME	Boiler and Pressure Vessel Code	1998 Edition, through 2000 Addenda
WENRA	Reference Levels : Issue B: Operating Organisation Issue C: Management System	September 2014
IAEA	Safety of Nuclear Power Plants: Design	No. SSR-2/1 (Rev. 1)
IAEA	Safety of Nuclear Power Plants: Commissioning and Operation	No. SSR-2/2 (Rev. 1)
IAEA	Leadership and Management for Safety	GSR Part 2
ISO	Quality management systems - Requirements	ISO 9001:2015

Annex 1

Assessment Findings – GI-AP1000-SI-05: Compliance of AP1000 Main Structural Components with ASME III Design Rules.

Number	Assessment Finding	Report Section
CP-AF-AP1000-SI-12	The licensee shall review developments in design and material selection to ensure the risks to structural integrity for the RPV CRDM penetrations and vent pipe sleeves are reduced so far as is reasonably practicable.	4.2.3.1
CP-AF-AP1000-SI-13	The licensee shall ensure that adequate evidence is provided to demonstrate that the ASME III analysis methods adopted for HSS and Class 1 components provide conservative stresses.	4.2.3.1
CP-AF-AP1000-SI-14	The licensee shall demonstrate that, if the loadings on any HSS or ASME III Class 1 vessel are revised to support licensing, its ASME III design analysis remains valid and conservative to substantiate the structural integrity safety case through life.	4.2.3.1
CP-AF-AP1000-SI-15	The licensee shall provide evidence to justify the corrosion allowances for the PRHR HX materials.	4.2.3.1
CP-AF-AP1000-SI-16	The licensee shall demonstrate that its License Condition 17 “management systems” arrangements provide a robust technical governance framework and a graded verification and approvals approach with the highest standard of that graded approach being applied to HSS and High Integrity components.	4.2.3.3