New Reactors Programme

GDA close-out for the AP1000® reactor

GDA Issue GI-AP1000-SI-06 – Structural Integrity Categorisation and Classification

Assessment Report: ONR-NR-AR-16-013
Revision 0
March 2017

Published 03/17

EXECUTIVE SUMMARY

Westinghouse Electric Company, LLC (Westinghouse) is the reactor 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) which had 51 GDA issues attached to it. 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 structural integrity of the Westinghouse **AP1000** reactor design. Specifically this report addresses GDA Issue GI-AP1000-SI-06 – Structural Integrity Categorisation and Classification.

This GDA issue arose in Step 4 because:

- ONR required further evidence that categorisation and classification has been applied in an appropriate manner to components with an important structural integrity claim.
- ONR required further evidence that components in **AP1000** plant Equipment Class C have been assigned a class that is consistent with their duty and implied reliability.
- ONR required further evidence that the steam generator vertical support is designed to withstand missile impact resulting from failure of the reactor coolant pump bowl.

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

- evidence that categorisation and classification has been applied in an appropriate manner to components with an important structural integrity claim.
- evidence that components in **AP1000** plant Equipment Class C have been assigned a class that is consistent with their duty and implied reliability.
- evidence that the steam generator vertical support will not be unduly challenged by the failure of the reactor coolant pump bowl. This evidence was judged necessary to better justify Westinghouse's classification of the reactor coolant pump bowl.

My assessment conclusion is:

- GDA Issue GI-AP1000-SI-06 can be closed.

My judgement is based upon the following factors:

- Westinghouse has provided sufficient evidence that a categorisation and classification process has been developed which should ensure that those SSCs with an important structural integrity claim are assigned a code or standard in line with ONR's expectations.
- Westinghouse has provided sufficient evidence that components in **AP1000** plant Equipment Class C have been assigned a class that is consistent with their duty and implied reliability.
- Westinghouse has provided sufficient evidence that catastrophic failure of a reactor coolant pump bowl would not unduly challenge the effectiveness of the vertical support, and the steam generator can survive without the support.

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

- The licensee shall complete the refined review of UK Class 2 pressure equipment (including pumps and valves) and storage tanks and demonstrate that the codes and standards (including EIMT) applied are commensurate with the safety classification and UK regulatory expectations.

- The licensee shall apply in-service inspection for the accumulator subsystem and other Class C systems, structures and components that is commensurate with UK expectations and relevant good practice.

LIST OF ABBREVIATIONS

AC	Alternating Current
ADS	Automatic Depressurisation System
ALARP	As Low As Reasonably Practicable
ANSI	American National Standards Institute
ANS	American Nuclear Society
ASME	American Society of Mechanical Engineers
BLEVE	Boiling Liquid Expanding Vapour Explosion
BMS	Business Management System
BSO	Basic Safety Objective
CMT	Core Make-up Tank
DAC	Design Acceptance Confirmation
DEG	Double Ended Guillotine
DRAP	Design Reliability Assurance Programme
DVI	Direct Vessel Injection
ECCS	Emergency Core Cooling System
EIMT	Examination, Inspection, Maintenance And Testing
FE	Finite Element
GDA	Generic Design Assessment
IDAC	Interim Design Acceptance Confirmation
IRWST	In-Containment Refuelling Water Storage Tank
ISI	In-Service Inspection
LBLOCA	Large-Break Loss Of Coolant Accident
LOCA	Loss of Coolant Accident
MOV	Motor Operated Valve
ONR	Office for Nuclear Regulation
OPEX	Operational Experience
PCCWST	Passive Containment Cooling Water Storage Tank
PCS	Passive Containment Cooling System
PCSR	Pre-Construction Safety Report
PSA	Probabilistic Safety Analysis
PWR	Pressurised Water Reactor
PXS	Passive Core Cooling System
QA	Quality Assurance
RAW	Risk Achievement Worth
RCL	Reactor Coolant Loop
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RGP	Relevant Good Practise
RNS	Normal Residual Heat Removal System
RPV	Reactor Pressure Vessel
RTNSS	Regulatory Treatment of Non-Safety Systems
SAPs	Safety Assessment Principles
SG	Steam Generator
SSCs	Systems, Structures and Components
TAG	Technical Assessment Guide
US NRC	United States (of America) Nuclear Regulatory Commission

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1 INTRODUCTION

1.1 Background

1. Westinghouse is the reactor 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) which had 51 GDA issues attached to it. 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.

2. This report is the Office for Nuclear Regulation's (ONR's) assessment of the structural integrity of the Westinghouse **AP1000** reactor design. Specifically this report addresses GDA Issue GI-AP1000-SI-06 – Structural Integrity Categorisation and Classification. There are three actions under this issue, given in Section 1.2.

3. The related GDA Step 4 report (Ref. 1) provides the assessment underpinning the GDA issue. Further information on the GDA process in general is also available on our website (Ref. 2).

1.2 Scope

4. The scope of this assessment is detailed in the ONR assessment plan (Ref. 3) and is limited to submissions for Step 4 GDA issue GI-AP1000-SI-06.

5. The scope of assessment focused on the following GDA Step 4 assessment actions, which need further substantiation by Westinghouse:

    - GI-AP1000-SI-06.A1: Provide evidence to show that the principal design and construction codes adopted for Class 2 Pressure Equipment and Storage Tanks are consistent with ONR's expectations as detailed within the SAPs.

    - GI-AP1000-SI-06.A2: Provide evidence to show that components in **AP1000** plant Equipment Class C have been assigned a class that is consistent with their duty and implied reliability.

    - GI-AP1000-SI-06.A3: Provide arguments to show that catastrophic failure of a reactor coolant pump bowl would not challenge the effectiveness of the vertical support for the steam generator.

6. This Step 4 GDA Issue GI-AP1000-SI-06 is captured in the Resolution Plan (Ref. 4) as follows:

    - A1 - Westinghouse has used non-nuclear codes for design of Class 2 categorised Systems, Structures and Components (SSCs) in lieu of nuclear codes as prescribed by the ONR Safety Assessment Principles (SAPs) for the same class. Westinghouse needs to justify their classification based on the safety significance, duties and demands of those SSCs against the relevant requirements of the SAPs.

    - A2 – Westinghouse needs to provide evidence to justify why it is appropriate to design and construct the Accumulator Tanks in the Passive Core Cooling System (PXS) to American Society of Mechanical Engineers (ASME) III Class 3, when previous reactor designs complied with ASME III Class 2 requirements.

- A3 – Westinghouse needs to justify that under postulated failure of the ASME III Class 1 Reactor Coolant Pump, the integrity of the vertical support for the Steam Generator would not be challenged.

7. My assessment concentrated on the claims / arguments / evidence provided as part of the pre-construction safety report (PCSR) to address the above mentioned GDA actions.

8. In my opinion the scope of the assessment is appropriate for the closure of the Step 4 GDA Issue GI-AP1000-SI-06.

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

1.3 Method

10. This assessment complies with ONR guidance on the mechanics of assessment (Ref. 5) and with the requirements of the ONR Business Management System (BMS) document "Purpose and Scope of Permissioning" (Ref. 6), which defines the process of assessment within the ONR.

1.3.1 Sampling strategy

11. It is rarely possible or necessary to assess an entire safety submission, therefore ONR adopts an assessment strategy of sampling. Ref. 6 explains the process for sampling safety case documents.

12. The sampling strategy for this assessment focused on aspects of the categorisation and classification identified in the GDA Step 4 report (Ref. 1) as requiring further evidence to establish compliance with UK expectations of relevant good practise (RGP).

2 ASSESSMENT STRATEGY

2.1 Pre-Construction Safety Report

13. ONR's GDA Guidance to Requesting Parties (Ref. 7) states that the information required for GDA may be in the form of a PCSR, and Technical Assessment Guide (TAG) 051 (Ref. 8) sets out regulatory expectations for a PCSR.

14. At the end of Step 4, ONR and the Environment Agency raised GDA Issue GI-AP1000-CC-02 (Ref. 9) 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.

15. 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 overall structural integrity issues covered in the PCSR. This assessment focused on the supporting documents and evidence specific to GDA issue GI-AP1000-SI-06.

2.2 Standards and Criteria

16. The standards and criteria adopted within this assessment are principally the SAPs (Ref. 10) internal TAGs, relevant standards and RGP informed by existing practices adopted on UK nuclear licensed sites

2.2.1 Safety Assessment Principles

2.2.2 The key SAPs that have informed my assessment are listed in Table 1.

2.2.3 Technical Assessment Guides

17. The TAGs that have informed my assessment are listed in Table 2.

2.2.4 National and International Standards and Guidance

18. The international standards and guidance that have informed my assessment are listed in Table 3.

2.3 Integration with Other Assessment Topics

19. 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. I have consulted with ONR specialists in fault studies, internal hazards, mechanical engineering, radiation protection and probabilistic safety analysis to inform my assessment.

2.4 Out of Scope Items

20. This report does not consider structural integrity aspects of the PCSR, which is addressed by a separate ONR cross disciplinary assessment.

3 REQUESTING PARTY'S SAFETY CASE

21. The latest full safety case for the **AP1000** reactor design and SSCs are detailed in the PCSR Ref. 11. The central theme for this reactor design is the use of Class 1 passive safety systems, relying solely on natural phenomena, such as natural cooling, gravity or the energy stored in pressurised pipes.

22. To address the three extant Step 4 GDA structural integrity actions related to GI-AP1000-SI-06, the safety case in the PCSR (Ref. 11) along with various supporting references have been assessed.

3.1 SI-06.A1

23. To address Action GI-AP1000-SI-06.A1 Westinghouse submitted Ref. 25 as evidence that categorisation and classification has been applied in an appropriate manner to components with an important structural integrity claim.

24. Westinghouse's stated purpose of Ref. 25 is to provide evidence to show that the non-nuclear principal design and construction codes adopted for United Kingdom **AP1000** plant Safety Class 2 pressure retaining equipment and storage tanks are adequate based on the component's safety significance, demands, and consequence of failure.

3.2 SI-06.A2

25. To address Action GI-AP1000-SI-06.A2, Westinghouse submitted Ref. 12 as evidence that components in **AP1000** Equipment Class C are assigned a class that is consistent with their duty and implied reliability.

26. Westinghouse's stated purpose of Ref. 12 is to provide evidence to show that components in **AP1000** plant Class C have been assigned a class that is consistent with their intended duty and implied reliability. In particular it provides arguments and evidence to show why it is appropriate to design and construct the accumulator tanks in the passive core cooling system (PXS) to ASME Section III, Class 3 when previous designs of reactors would have designed and constructed the accumulators to ASME Section III, Class 2 in line with the guidance provided in American National Standards Institute (ANSI) and American Nuclear Society (ANS) standard ANSI/ANS-51.1-1983. The arguments and evidence presented in Ref. 12 are intended to address the intended duty and implied reliability of the vessel, and provide evidence to justify why the **AP1000** design has deviated from ANSI/ANS-51.1-1983 in classifying the accumulators as Class 3.

3.3 SI-06.A3

27. Westinghouse has addressed the Step 4 GDA issues for GI-AP1000-SI-06 Action A3 in the following submission:

    - **AP1000** Plant Assessment of Impact from Reactor Coolant Pump Failure on Steam Generator Column Report; UKP-GW-GL-107 Revision 1 (Ref. 13)

28. The purpose of the safety case submissions referred below is to address the categorisation and classification principles used for some of the SSCs used in the Reactor Coolant Loop (RCL) and substantiate the commensurate methodology used for the assessments of those SSCs. This was raised in the Step 4 GDA and recorded under GI-AP1000-SI-06 (Ref.1).

29. The Steam Generator (SG) is a vertical pressure vessel supported laterally at the upper regions by several struts and by a single vertical support column at the bottom with the reactor coolant pumps (RCPs) in close proximity (Figure 1). The column is a built-up structure made of high-strength and low-alloy structural steel.

30. The **AP1000** design Reactor Pressure Vessel (RPV) is fed by two independent thermal cycle loops, each connected to a vertical SG and each SG is connected to a pair of RCPs. For each loop, as the “hot” reactor coolant is returned to the dedicated SG via the hot leg pipe, the “cold” coolant is pumped out of the SG by the pair of RCPs and returned to the RPV via pair of cold leg pipe, connected to each RCP bowl casing (Figure 2).

31. Westinghouse has considered the **AP1000** plant RCPs as ASME Class 1 components, which is not sufficient on its own is right to discount the possibility of gross failure. This leads to the requirement for considering direct and indirect consequences of the gross failure of the RCP bowl casing.

32. For brevity and succinctness, I am presenting below the main highlights of the safety case claims and arguments, the details are available in Section 5.2 of Reference 13.

3.3.1 Postulated Reactor Coolant Pump Casing Failure

33. The **AP1000** plant RCPs are high-inertia, high-reliability, low-maintenance, seal-less pumps that circulate the reactor coolant through the RPV, Reactor Coolant System (RCS) loop piping, and SGs. Seal-less pumps have an extensive operational history in both conventional and nuclear plants with a record of very good reliability and significantly reduced maintenance as compared to shaft seal pumps. The pumps are integrated into the steam generator channel head in the inverted position. The RCP is directly connected to each of two outlet nozzles on the steam generator channel head.

34. The **AP1000** design utilises four reactor coolant pumps. Two pumps are coupled with each steam generator and have shafts that rotate in the same direction. Each **AP1000** plant RCP is a vertical, single stage mixed flow pump designed to pump at high pressures and temperatures.

35. The RCP casing is a single piece forging connected to the SG nozzle via similar metal weld. The RCP bowl casing being a forging does not contain any weld. Hence, postulated disintegration of the RCP bowl casing would generate “missile” fragments. In Reference 13, Westinghouse has used two methodologies, Boiling liquid expanding vapour explosion (BLEVE) (Ref. 14) and R3 (Ref. 15), to determine the most bounding “fragment” size for the “missile”. Whilst BLEVE recommended that disintegration of a pressurised spherical vessel could result in 10 to 20 fragments, R3 recommended [redacted] as the mean fragment size for a similar vessel. Since the shape of the RCP bowl casing is similar to that of an inverted hemisphere in shape, in Reference 13, Westinghouse concluded following R3 recommendations that [redacted] fragment size [redacted] is most conservative, based on kinetic energy.

36. In Table 5-1 of Reference 13, following R3 (Ref. 15), Westinghouse has also used the bounding velocity [redacted]) for the fragment (hence kinetic energy, [redacted]) at normal operating conditions and assumed no loss of strain energy due to ductile (normal operating temperature) fragmentation of the casing.

3.3.2 Dynamic Analysis and Postulated SG Support Column Failure

37. Westinghouse performed a transient structural analysis of the reactor coolant loop (RCL) with the RCP casing failure, launching the “bounding” fragment, which strikes the vertical support column near the connection with the SG. The transient analysis captures the time it takes the fragment to impact the SG column, the time for the SG column to absorb all the R3 level energy and the dynamic response loads as the RCL responds to the transient loadings. The analysis assumes failure of the SG vertical support column and predicts the system response when the SG is left to be supported by the hot leg connection with the RPV, the remaining cold leg connection with the

other RCP, lateral supports, and SG secondary piping (main steam, feed water etc.) (Figure 2 and Figure 3).

38. The dynamic analysis results are used to generate support loads, piping stresses and strains. The load combinations via three load cases are used on the stress evaluations include operation loadings combined with the additional RCP blast loads plus the effects of cold leg break thrust loadings (Figure 3).

39. Westinghouse used ASME Section III Level D (Ref. 16) design by analysis plastic failure criteria for calculating allowable stresses and claims the integrity of SG and the associated hot and cold leg piping (including nozzles) in the absence of the vertical support column. The assumptions and conservatism Westinghouse used in the analysis to justify the structural integrity is stated in Section 5.2.2.3 of Reference 13.

40. Westinghouse undertook additional parametric studies over and above the three load cases of thrust loadings as discussed above to check for the variation in the response pattern of the SSC and to identify any potential “cliff-edge”. The parameters included assumed integrity of the SG vertical support column under RCP “bounding” fragment impact, assumption of conservative thrust coefficient to maximise thrust loads and varying the duration of impact. Based on the results: displacements, stresses, strains, forces and response times presented in Reference 13, Westinghouse claims that the analyses results do not show any abnormal trend or “cliff edge” and that the primary load cases (three) remain bounding.

3.3.3 Modification of the SG Support Column

41. Westinghouse undertook a separate FE impact analysis (Section 5.2.7 of Ref. 13), which shows that the SG column maintains its structural integrity following the [redacted] fragment impacting with the velocity of [redacted] The energy was dissipated through the local plastic deformation of the fragment, column, pins, connector, and the clevis. The majority of the column remains elastic and no gross structural failure was observed (Figures 5-12 & 5-13 of Ref. 13). The SG column assembly bounced back after the fragment impact.

42. However, to allow the SG vertical support column to better withstand “missile” impact from a postulated RCP casing failure and reduce the risks from such impacts to as low as reasonably practicable (ALARP), Westinghouse proposed additional options: addition of missile barrier, using high strength material for the SG support column, enlarging the support column and local strengthening of the column built-up section near the impact region (last option based on suggestion from ONR, References 18, 19 and 20).

43. In Reference 13, Westinghouse discussed the first three options but discounted them due to either lack of available space in the location or other operational and cost related issues.

44. Westinghouse proposed a design change to the current “built up” column design by welding additional plates parallel to the web of the column (Figure 5-17 of Ref. 13). This results in an increase to the built-up section area and bending moment of inertia. The effect of the modification is expected to decrease the extent of plasticity that occurs in the built-up section extreme fibres, increase the width of the elastic core, and therefore increase the confidence that the built-up section will not fail in bending as a result of lateral impact loading. The proposed design adds vertically oriented web plates to effectively [redacted] the built-up column while maintaining the space envelope requirements. Westinghouse claims that this design is expected to increase the weld distortion and affect the manufacturability of the column. Therefore, Westinghouse is considering a balanced approach in only reinforcing the [redacted] of the built-up section, which is most likely to be affected by the postulated impact.

45. Westinghouse claims that the proposed modification to the SG column design will produce significant increase in bending capacity; therefore, improved survivability is evidenced. Westinghouse argues that the column is not expected to fail for the evaluated conditions because the elastic section modulus is increased by 97%, whilst the plastic section modulus is increased by 66%. Therefore, the elastic core width is expected to significantly increase for the proposed modification to the SG column. This is based on large increases to the section moduli and that any risks to the integrity of the vertical support column from the postulated RCP casing failure is reduced to ALARP.

46. Westinghouse also claims that the modified design is also not expected to have a large impact on the interfacing SSCs and associated analyses. The change in vertical column stiffness is small (< 5%) in order to minimise downstream effects on structural SSC calculations and interfacing designs of the steam generator and reactor coolant loop piping. Additionally, as necessary the detailed geometric parameters of the SG column design may be refined (such as the built-up column length) to more closely match the SG column vertical stiffness, as necessary, to avoid detailed reanalysis of the RCL and SG for vertical column stiffness variations upon implementation.

47. Westinghouse claims that this option would provide the benefit of providing a more robust design for the support to withstand the impact from the RCP casing, while not impacting the configuration of the interfacing components.

4 ONR ASSESSMENT OF GDA ISSUE GI-AP1000-SI-06

48. This assessment has been carried out in accordance with HOW2 guide NS-PER-GD-014, "Purpose and Scope of Permissioning" (Ref.6).

4.1 Scope of Assessment Undertaken

49. The scope of the assessment is to address the three structural integrity issues raised under GI-AP1000-SI-06 (Ref. 1).

- Provide evidence to show that the principal design and construction codes adopted for UK Class 2 Pressure Equipment and Storage Tanks are consistent with ONR's expectations.
- Provide evidence to show that components in **AP1000** plant Equipment Class C have been assigned a class that is consistent with their intended duty and implied reliability.
- Westinghouse has assigned the RCP as a Standard Class 1 component, which would be designed and constructed to ASME III (Ref. 167). In the UK this is insufficient evidence to discount gross failure. Hence, Westinghouse needs to address the consequences of failure of the pump bowl. Further, the RCP bowl is in close proximity to the vertical support column for the SG, a HSS component. Westinghouse needs to provide evidence that the effectiveness of that vertical support column will not be challenged by the postulated failure of the pump bowl, in order to support the structural integrity classification of the RCP.

4.2 Assessment

50. This part of the report is divided into four sections and which describe in turn the following aspects:

- Assessment of Westinghouse response to Action GI-AP1000-SI-06.A1.
- Assessment of Westinghouse response to Action GI-AP1000-SI-06.A2.
- Assessment of Westinghouse response to Action GI-AP1000-SI-06.A3.
- Key assessment considerations and regulatory judgements.

4.2.1 Action GI-AP1000-SI-06.A1

Categorisation and Classification of SSCs

51. Allocating appropriate classification to pressure equipment and tanks is an important part of the structural integrity safety case as it defines the design, construction, inspection and through life maintenance of the component. It is therefore appropriate to consider the impact of the classification of SSCs and the design and manufacturing standards applied to SSCs, including appropriate quality assurance commensurate with the safety classification, to ensure the risk of failure is reduced so far as is reasonably practicable.

52. At GDA Step 4, Westinghouse adopted a UK specific classification scheme (Ref. 21) which was a development of their global approach to categorisation and classification. These modifications were commensurate with the expectations of the SAPs. In their UK Categorisation and Classification methodology, SSCs are classified according to their contribution to a safety function in line with SAP ECS.2, where a 3-tier approach

is suggested with SSCs allocated to Class 1, 2 or 3. The safety classification informed by the functional requirements then informs the selection of codes and standards (SAP ECS.3).

53. ONR was generally satisfied that the categorisation and classification scheme developed for the UK and contained in Chapter 5 of the PCSR is compliant with the SAPs, but late in GDA Step 4, ONR became aware that the application of the methodology differed from what ONR anticipated. Notably, within the structural integrity discipline, nuclear specific codes were applied to UK Class 1 pressure equipment and storage tanks, but non-nuclear codes and standards were proposed for UK Class 2 pressure equipment and storage tanks.

54. In the UK, Class 2 is assigned where a SSC provides either a significant (but not principal) means of delivering a Category A safety function, or the principal means of delivering a Category B safety function (SAP ECS.3). On the basis of the safety functional delivery, ONR's initial expectations call for nuclear codes and standards to be applied for design, construction and through-life examination, inspection maintenance and testing (EIMT), of UK Class 2 SSCs. Therefore any proposal to apply non-nuclear codes and standards needs to be justified. The matter was taken forward as GDA Issue GI-AP1000-SI-06 A1:

'Provide evidence to show that the principal design and construction codes adopted for Class 2 Pressure Equipment and Storage Tanks are consistent with ONR's expectations as detailed within the SAPs, particularly ECS.3 and supporting paragraphs 157-161. In particular, where non-nuclear Pressure Equipment and Storage Tank design and construction codes are used in the design of Class 2 components Westinghouse will need to fully justify each case to show the arguments and evidence which support the use on non-nuclear codes. The arguments and evidence should take account of:

- *the safety significance of the component;*
- *the demands that are placed on the system in terms of loadings, fatigue, temperature etc., and;*
- *the consequences of failure of pressure boundary in terms of both the loss of system function and on the Internal Hazards safety case.*

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

55. From the structural integrity perspective pressure equipment and storage tanks is taken to comprise a system of pressure retaining SSCs, which may include pipework, pump casings, valve bodies, storage tanks and vessels.

56. In the case of pressure equipment and storage tanks there are nuclear design and construction codes available e.g. ASME III. Thus applying non-nuclear codes for the design and construction of UK Class 2 pressure equipment and storage tanks does not meet ONR's normal expectations (SAP ECS.3). ONR's expectations were affirmed at GDA Step 4 (Ref. 22):

57. 'The topic of categorisation and classification is discussed in the Step 4 Cross cutting report (Ref. 23). However, with the exception of the structural integrity discipline, the proposed use of non-nuclear codes and standards is not discussed in Ref. 23. Post GDA Step 4, the proposed use of non-nuclear codes and standards is addressed on a discipline specific basis, and that the conclusions drawn, are captured as part of the close-out of various other GDA Issues. Assessment by ONR of the PCSR is described in Section 2.1.

58. The scope of this assessment was initially limited to the use of non-nuclear codes and standards for UK Class 2 pressure equipment and storage tanks. However, Westinghouse adopted a system-based approach for UK Class 2 SSCs, hence I considered it appropriate to increase the scope of my assessment to include other disciplines.

US Approach

59. The **AP1000** reactor design was developed in the United States (US), where a two tier approach to safety classification; "safety" and "non-safety" is widely used. In the US, RG1.26 (Ref. 33) is used to establish the design standards used for various quality groups of components for light water reactors. These quality groups are divided into Quality Group A, B, C, and D. Quality Groups A, B, and C are the groups associated with equipment important to safety (safety related):

- Quality Group A - the reactor coolant system (RCS)
- Quality Group B - the emergency core cooling system (ECCS) functions and radioactive material boundaries e.g. containment for a LOCA
- Quality Group C - the support systems such as service water, component cooling, etc.
- Quality Group D - non-safety, allows the use of industry codes, i.e. ASME B31.1 the power piping code and various industry codes for tanks and pumps. This is consistent with ANSI/ANS 58.14. This standard just divides Quality Group D into Class 4 and 5. Class 4 in 58.14 has augmented quality requirements.

60. For passive plants such as the **AP1000** reactor design, the US Nuclear Regulatory Commission (USNRC) uses Regulatory Treatment of Non-Safety Systems (RTNSS) guidance to address those non-safety SSCs in passive plants that would normally be safety related in an active plant (Ref. 24).

UK Regulatory Expectations

61. The focus in US guidance for RTNSS (Ref. 24) is on availability controls and not codes/standards. In addition, the guidance is heavily reliant on probabilistic risk assessment. The output is a graded, risk informed approach to Technical Specifications, whereas for the UK the expectation is a graded approach to category, classification and codes/standards (ECS.1 to ECS.3).

62. Other key differences under the UK regulatory regime that inform the classification and subsequent expectations for the selection of codes and standards include:

- There is a greater emphasis on the deterministic safety case - classification is informed by the consequences (direct and indirect) of postulated gross failures
- Within the structural integrity discipline more emphasis is placed on ISI to provide forewarning of failure than reliance on leakage monitoring
- There is a legal duty on the licensee to reduce risks so far as is reasonably practicable.

63. SAP ECS.3 indicates that for Class 2 SSCs, nuclear codes and standards are expected. A safety case to use non-nuclear codes and standards should therefore provide evidence to show equivalence to a nuclear code Alternatively, and subject to

adequate evidence of their lower importance, a re-classification of certain UK Class 2 SSCs may be justified.

64. The key principles, TAGs and guidance that informed my assessment are listed in Tables 1-3.

AP1000 Plant Review of UK Class 2 Structures, Systems and Components

65. Westinghouse submitted its AP1000 Plant Review of UK Class 2 Structures, Systems and Components, UKP-GW-GL-105 (Ref. 25), with a declared purpose of 'providing evidence to show that the non-nuclear principal design and construction codes adopted for the UK **AP1000** plant Safety Class 2 pressure retaining equipment and storage tanks are adequate based on the component's safety significance, demands, and consequences of failure.'

66. To inform assessment of Ref. 25, I raised RQ-AP1000-1501. Key themes included:

- the scope and basis for the proposed use of non-nuclear codes
- the reliability of safety function delivery
- document revision and equivalence studies
- safety classification and reliability claims (nuclear and non-nuclear codes and standards)
- ALARP optioneering and risk reduction

67. Westinghouse provided a detailed response RQ-AP1000-1501 (Ref. 26). I summarise the main points emerging from consideration of the Westinghouse responses below.

- With regard to the scope of the proposed application of non-nuclear codes and standards I established that this was primarily based on Westinghouse's design philosophy rather than exemption or the unavailability of nuclear codes and standards.
- Westinghouse also confirmed that certain UK Class 2 SSC have 'augmented' quality assurance (QA) provisions. These UK Class 2 SSCs were identified under the **AP1000** Design Reliability Assurance Programme, as identified in Ref. 29. The 'augmented QA' included QA Group D within Regulatory Guide 1.26 with supplements (Ref. 216).
- Westinghouse proposed a system-based approach to assessing the importance of the UK Class 2 SSCs. With this approach the significance of SSC's were informed by the probabilistic safety analysis (PSA) which shows that failure of the active safety function of SSCs e.g. pumps and valves, on demand during a fault, more significantly contributes to plant risk in the PSA than failure of the pressure boundary.
- Assessment of the indirect consequences of failure to UK expectations which was based on the consideration of gross failures was in progress as part of the response to internal hazards issue GI-AP1000-IH-03. However, the Westinghouse case was based on generic claims rather than the consideration of consequences of failure of the individual UK Class 2 SSCs.
- The Westinghouse optioneering and ALARP case was predicated on the premise that the use of non-nuclear codes and standards was the ALARP

option rather than presenting a balanced view of the benefits and detriments with cross-discipline input.

68. A corollary of the ‘system-based approach’ is that ‘active SSCs’ that deliver safety functions for example pumps and valves warranted consideration. Within ONR these ‘active SSCs’ are the remit of the mechanical engineering specialism. The consequences (direct and indirect) of postulated gross failures of these UK Class 2 pressure equipment and storage tanks along with the increased frequency of their failure were also relevant. Given the cross-discipline implications, I consulted ONR fault studies, internal hazards, PSA and mechanical engineering specialist assessors in developing my subsequent regulatory queries.

Review of UKP-GW-GL-105, Revision 1

69. In response to my queries, Westinghouse updated submission UKP-GW-GL-105 (Ref. 27). I reviewed the revised submission. I issued RQ-AP1000-1779 to clarify the expected evidence to justify the UK classification and to infer the achievement of an appropriate level of structural reliability consistent with UK expectations. The questions, in brief related to:

- Through life assurance of SSCs
- Provision of a list of risk important Class 2 SSCs
- Confirmation that the purpose of its categorisation and classification guidance correspond to the PCSR.
- Confirmation of ASME inspection standards
- Confirmation of the application of nuclear codes and standards together with exemptions
- Claims against the fault schedule
- What improvements Westinghouse could put in place to improve its safety justification
- Clarification of the seismic withstand capability; and
- Achievement of PSA risk targets

70. In my opinion, downgrading to non-nuclear codes and standards for UK Class 2 pressure equipment and storage tanks incurs an attendant increased risk to the delivery of the safety function of pressure boundary integrity. The key question relates to the role and importance of the individual UK Class 2 pressure equipment and storage tanks in the **AP1000** plant safety case. In my view the following considerations were also relevant:

- visibility of the safety case in terms of claims, arguments and evidence to justify the use of non-nuclear codes and standards for UK Class 2 pressure equipment and storage tanks. .
- the diverse range of functions and consequences (direct and indirect) of postulated gross failure of the UK Class 2 pressure equipment and storage tanks was not amenable to the use of a generic approach.

- an understanding of significance of the loss and consequences of gross failure of the UK Class 2 pressure equipment and storage tanks from the **AP1000** plant safety case.

- the importance of the UK Class 2 components on an individual and collective basis for example to demonstrate sufficient resilience to seismic events in design basis analysis and the PSA.

71. I revisited the Westinghouse resolution plan and noted that the provision of evidence was expected to cover a review of the UK Class 2 systems on an individual basis (Ref. 4). In RQ-AP1000-1779 I therefore asked Westinghouse to collate and reference the evidence covering several factors relevant to a review on an individual UK Class 2 system basis for pressure equipment and storage tanks:

- The safety significance with consideration of the direct consequences of the loss of the UK Class 2 system – a demonstration of adequate redundancy and diversity to meet UK expectations. This should cover the loss of the function and any dependencies for service systems that support UK Class 1 or UK Class 2.

- The demands placed on the systems (e.g. whether the pressures and temperatures warrant exclusion from nuclear codes and standards e.g. ASME III exceptions).

- The indirect consequences of failure – a demonstration that following the loss of the identified system the indirect consequences in terms of the internal hazards safety case (e.g. flooding, pressure part failure, missiles, explosions, environment etc.) have been considered i.e. effects on UK Class 1 and/or UK Class 2 SSC with evidence to show adequate delivery of safety functions and no significant implications for the UK safety case.

- Confirmation the proposed provision of non-nuclear codes and standards including augmented QA in design and EIMT provisions (prior to and during service) equate to UK Class 2 or whether UK Class 3 classification is more appropriate.

- A reasoned argument to justify any differences in the selection of codes and standards for UK Class 2 pressure equipment and tanks in the **AP1000** reactor design compared to similar systems in current pressurised water reactors (PWRs).

72. I held a meeting to discuss the Westinghouse response to RQ-AP1000-1779 and my expectations. I advised Westinghouse of the points above. I also highlighted the need to consider safety functional requirements under both normal (duty system) and fault or accident conditions (safety system) (Ref. 29). Indeed, SSCs often have dual functions and the safety classification and subsequent selection of codes and standards are informed by both sets of functional requirements.

Cross-Discipline Considerations

73. Westinghouse provided a detailed response to my queries of RQ-AP1000-1779 (Ref. 29). Key aspects of the Westinghouse response together with ONR's consideration are summarised below.

74. 'Risk important' UK Class 2 SSCs are subject to augmented QA commensurate with the UK Class 2 safety function(s) of the SSC. However, the companion through-life EIMT provisions to maintain this level of functional reliability are implemented by the licensee and feature in the operating rules under GDA Issue GI-AP1000-CC-01.

75. I make the following observations:

- Westinghouse proposed that for UK Class 2 systems only the 'risk important' SSCs are subject to nuclear codes and standards. Hence, UK Class 2 systems may include SSCs with a mixture of provisions. In these situations the expectation is that there would be a demonstration that delivery of the safety functions is not affected by any lower standard of provision.

- The through-life assurance and maintenance of UK Class 2 SSCs is a licensee choice.

76. Westinghouse identify the Normal Residual Heat Removal System (RNS) as an example where due to the significance of the normal pressure boundary function i.e. the 'duty' function, UK Class 1 is assigned (with the UK Class 2 applied to the 'active SSCs'). Thus nuclear codes and standards will be applied for design, construction and through-life EIMT off the pressure boundary of the RNS. Indeed, these exceed UK Class 2 for parts of the RNS.

77. Westinghouse provided its criteria and an example of the information drawn from mechanical engineering, probabilistic safety assessment, fault studies and internal hazards parts of the PCSR. Westinghouse used this to provide evidence on a UK Class 2 system level basis to underpin selection of codes and standards.

78. In principle, I consider that the process should capture the expectations in the SAPs, leading to proportionate and targeted selection of codes and standards.

79. In summary, in responding to RQ-AP1000-1779, Westinghouse presented a process with certain criteria to inform its view of the role of the individual UK Class 2 systems of pressure equipment and tanks (Ref. 29). As RQ-AP1000-1779 was relevant to a number of technical disciplines I discussed the Westinghouse response with ONR fault studies, mechanical engineering, internal hazards and PSA specialists. The items discussed below present a multidiscipline response.

Categorisation and Classification Methodology

80. ONR's Step 4 consideration of Westinghouse's categorisation and classification arrangements found them to be adequate. Nevertheless, there are several cat/class related assessment findings related to licensee's implementation. For example, AF-AP1000-ME-01,05 & 20, AF-AP1000-IH-05, AF-AP1000-EE-021 & 22, AF-AP1000-CI-004, AF-AP1000-SI-01, 02, 03 & 30, AF-AP1000-FS-01, 02 & 03 etc. (Ref. 30)

81. ONR considers that Westinghouse's approach to the use of codes and standards for Class 1 SSCs to be reasonable, i.e. its use of nuclear codes and standards. Westinghouse has proposed a graded approach (use of both nuclear and non-nuclear standards) to classifying Class 2 / 3 SSCs that deliver safety functions. ONR has questioned some points of detail with the application of the graded approach to UK

Class 2 SSCs, see paragraph 84ff. This has resulted in Assessment Finding AF-CP-AP1000-SI-17, see paragraph 168.

82. Nevertheless, putting Westinghouse's application of category / classification to one side, ONR expects the licensee to implement its own categorisation and classification arrangements. These arrangements need to:

- build on the Westinghouse graded approach to classification developed for GDA.
- Take cognisance of the UK expectations below, and in particular, any differences identified in the importance of the UK Class 2 SSCs which may mean that a lower classification is warranted.

Codes and Standards Relative to Class 2/3 SSCs

83. Westinghouse has confirmed that all mechanical mitigation Class 2 SSCs, presented in the fault schedule, are designed and built to nuclear codes and standards. For the purposes of GDA this is considered appropriate. However, ONR is not confident that all risk important SSCs are appropriately identified (see paragraph 84ff). Consequently, ONR would expect the licensee to identify all risk important Class 2 / 3 SSCs and manage them appropriately i.e. design, manufacture, construct, install, commission, quality assure, maintain, test and inspect them to appropriate nuclear codes and standards. This should include appropriate through life management to facilitate ONR's expectations regarding forewarning of failure.

Risk Important SSCs

84. Westinghouse present a list of risk important systems that warrant inclusion in its Design Reliability Assurance Programme (DRAP) (identified in Ref. 29). However, ONR has identified more Class 2 / 3 systems from internal events at-power PSA (Ref. 31) that are considered risk significant. For example, the main AC power system (RAW 7610) and the chemical and volume control system (RAW 84).

85. Westinghouse's risk significance measure is based on Risk Achievement Worth (RAW) which is agreed as appropriate. ONR has identified that the values quoted, and their use in the response to RQ-AP1000-1779 Query No 7, for example, differ from those presented in the PSA. RAW measures can be presented for both initiating events and the provision of protection to the reactor as a safety measure. Notably, these two measures provide different information about the risk significance of a system. ONR notes that the PSA can provide a selection of importance measures, for example, system importance, fault mitigation importance and initiating event importance. These measures provide different information about the risk significance of a system. The licensee should consider use of multiple importance measures to fully understand the risk importance of the individual SSCs for their different functions. This will help clarify whether a particular system attracts different categorisation and classification for its normal duty safety function, its pressure retention function (where applicable) or its response as a safety measure to a fault. The risk importance should inform the type of codes and standards used for each of the Class 2 / 3 SSCs.

86. ONR assessment has identified that core damage frequency and large release frequency is both individually and cumulatively sensitive to the reliability of the Class 2/3 SSCs. A risk sensitivity study in the PSA illustrates that ONR's target 8 Basic Safety Objective (BSO) would be exceeded by a factor of 102 (if all Class 2 / 3 SSCs are assumed to be unavailable). It is noted that this assessment includes the normal residual heat removal system, which is designed and constructed to nuclear codes and standards, and that the simultaneous unavailability of all Class 2/3 SSCs is beyond design basis. Consequently, the collective effect on risk should demonstrate that the

graded choice of codes and standards across multiple systems provides overall plant risks which are reduced so far as is reasonably practicable.

87. ONR has raised internal hazard finding CP-AF-AP1000-IH-06, which captures the need to complete the indirect consequences analysis, i.e. dynamic effects of pipe whip, jet impact etc. The output from this assessment finding may determine the addition of an unacceptable indirect consequence (and therefore affect the choice of codes and standards) for the Class 2 / 3 SSCs.

Reliability Claims

88. Westinghouse currently proposes to use a graded approach to classification, i.e. not all of its Class 2 SSCs will designed, built, quality assured and asset managed (EIMT) to the same standards. Hence, the reliability offered by the Class 2 SSCs may vary. The licensee will need to satisfy itself that the codes and standards used for the SSC correspond to their risk importance (reliability) claimed within the PSA.

89. ONR is satisfied that Westinghouse has developed an adequate process (Ref. 29) to collate the evidence necessary to inform a reasoned judgement on the selection of the codes and standards for UK Class 2 pressure equipment and tanks. ONR notes that, at the time of writing this report, this process has yet to be fully implemented.

90. In addition, during my assessment I have raised a number of detailed points that require consideration by a licensee to meet UK regulatory expectations. These cover:

- claims made in the safety case for normal (duty system) and fault (safety system) conditions and the safety functional requirements;
- consideration of operating conditions for the system during duty and fault conditions i.e. system demands;
- consideration of the direct and indirect consequences of failure;
- consideration of the through-life EIMT required to provide assurance of SSC integrity; and
- consideration of the impact of using non-nuclear codes and standards for design, manufacture, examination, inspection, maintenance and testing through-life in assuring safety functional requirements of SSCs are delivered.

91. These are considerations that a licensee should take forward in assuring itself that the choice of codes and standards for UK Class 2 SSCs and through-life asset management are commensurate with nuclear safety significance. I have raised Assessment Finding AF-CP-AP1000-SI-17 (see Annex 1) covering SSC codes and standards for UK Class SSCs to progress for licensing. In addressing that assessment finding, the licensee should take account of:

- Claims in the fault schedule
- PSA consideration of the risk importance of the SSCs. The impact on the core damage frequency and large release frequency in both normal and shutdown states should be considered.
- SSC reliability claims within the PSA.
- Overall balance of safety offered by different SSCs to demonstrate risks are reduced ALARP by measures including design, QA and EIMT.

4.2.2 Action GI-AP1000-SI-06.A2

92. Nuclear pressure vessel design and construction codes such as ASME III set out a range of requirements for the design and construction of pressure vessels and associated pressure equipment. The requirements are graded according to which of three classes is specified for the component. ASME III Class 1 components are designed, constructed and inspected to higher standards than ASME III Class 2 and likewise to ASME III Class 3. Whilst the ASME III code provides rules for the design and construction of nuclear components, ASME III does not provide the criteria for allocating the class that should be specified for a particular component. In the US, SSC classification is informed by guidance from other sources e.g. ANSI/ANS 51.1 (Ref. 32) and RG 1.26 (Ref. 33).

93. At GDA Step 4, ONR was generally satisfied that the categorisation and classification scheme developed for the UK and contained in Chapter 5 of the PCSR is compliant with the SAPs. However, late in GDA Step 4, ONR identified that the classification scheme as applied to pressure equipment and tanks needed further justification. Notably, the PXS is a Class C system in the US **AP1000** plant classification scheme and is assigned a UK Class 1 safety classification by Westinghouse. However, in current operating PWRs, SSCs that perform the emergency core cooling system (ECCS) function is typically classified the equivalent of US **AP1000** plant Class B. UK Class 1 in the Westinghouse classification scheme includes the US Class A, B and C with a graded approach of design, quality and ISI provisions (Ref. 33). Thus the use of ASME III Class 3 for the design and construction of the accumulators in the PXS effectively downgrades the ASME III code provisions. The matter was taken forward as GDA Issue GI-AP1000-SI-06 A2:

*'Provide evidence to show that components in **AP1000** Equipment Class C have been assigned a class that is consistent with their intended duty and implied reliability. In particular Westinghouse need to provide arguments and evidence to show why it is appropriate to design and construct the Accumulator Tanks in the Passive Core Cooling System to ASME III Class 3 when previous designs of reactor would have designed and constructed the Accumulators to ASME III Class 2 in line with the guidance provided in ANS-51.1-1983. The arguments and evidence should address:*

- *the intended duty and implied reliability of the vessel, and;*
- *provide evidence to justify why the **AP1000** design has apparently downgraded the classification of the core cooling system from the criteria set in ANS-51.1-1983.*

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

94. A corollary of the proposed adoption of a lower ASME III design class is that there is also a reduction in the through-life inspection provisions under ASME XI. Therefore reference to downgrading of the ASME III class also includes the commensurate ASME XI ISI provisions. I use the term 'ASME nuclear provisions' for brevity below. Several SAPs (see Table 1) and TAGs (see Table 2) were relevant to my assessment. The salient principles and guidance that informed my assessment are given in Tables 1 to 3.

4.2.2.1 Review of UKP-GW-GL-106

95. Westinghouse provided submission UKP-GW-GL-106 Rev 0 (Ref.12). This submission is intended to provide arguments and evidence to show that components in **AP1000** plant Class C have been assigned a class that is consistent with their intended duty and implied reliability. As an example, the submission is intended to justify that the

design and construction of the accumulator tanks in the PXS to ASME Section III Class 3 (rather than ASME III Class 2) is commensurate with reducing risks ALARP.

96. Intuitively, downgrading of the ASME nuclear provisions infringes my confidence in the through-life structural integrity of the accumulator sub system. Not only are these changes difficult to quantify but over reliance on uncertain estimates of the failure frequencies should not provide the main basis for a demonstration that the risks are reduced ALARP. Instead, my judgement is primarily based on establishing the importance of these Class C systems i.e. understanding the consequences of failure (direct and indirect) and gaining assurance that delivery of the safety functions is robust and commensurate with the level of reliability implied from the Westinghouse UK safety classification.

97. I carried out an initial review of Westinghouse submission UKP-GW-GL-106. The scope of my initial review was limited to accumulator vessels. I raised RQ-AP1000-1413 (Ref. 35). Key themes included:

• UK safety categorisation, classification and SAPs

• implied and implication of reliability claims

• defence in depth provisions

• coherency in safety case (including cross-discipline considerations)

• in-service inspection & monitoring

• optioneering & reducing risks ALARP

98. Westinghouse provided a detailed response, given in Ref. 35. I subsequently raised several points for further discussion (Ref. 36), in brief these covered:

- whether the intent was to cover individual SSCs or the class C system.
- whether the ALARP case proposed for the PXS could be applied to the other Class C systems
- the implications for the implied reliability and the level of defence in depth provision associated with the proposed adoption of lower ASME nuclear provisions
- ISI provisions and the knowledge of the condition of the accumulators through-life
- The safety functional requirements for the accumulators in the **AP1000** plant compared to current PWR plant; and
- The need for a balanced ALARP case i.e. a fair consideration of the benefits, detriments and implications associated with options to either retain or lower the ASME nuclear provisions.

99. In response to my queries, Westinghouse updated submission UKP-GW-GL-106 to Revision 1 (Ref. 37). The main changes to UKP-GW-GL-106 included:

- further information covering the **AP1000** plant methodology development
- the inclusion of a ‘system-based approach’ in considering the significance of the loss of the accumulator pressure boundary i.e. with consideration of ‘active SSCs’ e.g. valves that actively rather than passively deliver functions.

- a review of operating experience for the accumulators in operating plants
- further consideration of the consequences of PXS failure and system reliability; and
- an expanded ALARP case for the selection of the ASME nuclear provisions for the accumulators with the inclusion of RGP, limited optioneering, dose and cost estimates.

100. ONR's structural integrity discipline is primarily concerned with assessing the integrity of the pressure boundary of components. A corollary of the Westinghouse proposal to consider on a 'system basis' was the inclusion of 'active SSCs' i.e. SSCs that deliver functions e.g. valves and pumps. Within ONR these 'active SSCs' are the remit of ONR's Mechanical Engineering discipline.

101. Given the potential wider implications for the safety case my subsequent regulatory queries were informed by internal discussions with ONR's fault studies, internal hazards, PSA and mechanical engineering specialist assessors. I raised a further regulatory query RQ-AP1000-1785 (Ref. 38) to inform my assessment. Key points included:

    - the scope of the Class C systems in the **AP1000** plant
    - Design and ISI provisions
    - Duties, implied reliability and consequences of failure; and
    - Optioneering to and evidence to underpin the Westinghouse ALARP position

102. My opinion and regulatory considerations take cognisance of the Westinghouse responses to main points raised in Refs. 35 and 35. Key aspects for the accumulators and the other Class C systems are summarised below.

AP1000 Plant Accumulators

Duties, Implied Reliability and Role in the Safety Case

103. The accumulators are part of the PXS, a passive safety system in the **AP1000** design, which Westinghouse claim as highly reliable in providing the principal means to deliver a Category A safety function. The accumulators are UK Class 1 and have the following functional requirements:

    - deliver a large volume of borated water to the RPV at a high flow rate in the event of a large break loss of coolant accident
    - provide adequate core cooling during smaller LOCA events
    - during small LOCAs e.g. a direct vessel injection (DVI) line break, the accumulators assist in keeping the core covered with water; and
    - a duty function to maintain the accumulator pressure boundary during, standby, normal operation and under design basis faulted conditions for the 60 year design life.

104. The PSA also shows that, if the Core Make-up Tanks (CMT) fail to operate, there are many initiating events which depend on operation of the accumulators.

105. In summary, these functional requirements include safety injection for protection against faults and a duty function to maintain the accumulator pressure boundary. The accumulators play a prominent role in the **AP1000** plant safety case and so the consequences of their failure and any implied reduction in their reliability or the PXS

system are important. I confirmed that Westinghouse has considered the direct and indirect consequences of a postulated gross failure of an accumulator and assigned a structural integrity class of Standard Class 1. I consider the assignment of Standard Class I as appropriate and note that confirmation of the structural integrity classification is the subject to AF-AP1000-SI-02.

106. The Westinghouse **AP1000** plant classification was informed by ANSI/ANS 51.1 (Ref. 32). ANSI/ANS 51.1 and the subsequent standard ANSI/ANS 58.14 (Ref. 34) are now withdrawn, nonetheless pending the provision of updated classification guidance; I view as a source of RGP. In ANSI/ANS 51.1 the accumulators form part of the ECCS and would warrant ASME III Class 2. In contrast the lower nuclear provisions of ASME III Class 3, are reserved for support systems that help to deliver plant safety functions e.g. service water systems

107. ANSI/ANS 51.1 is based on plants with 'active' rather than passive protection systems. Westinghouse adapted the ANSI/ANS 51.1 standard for use in the **AP1000** design.

108. I asked Westinghouse to explain the changes in the functional requirements for the **AP1000** plant accumulators compared to plant with active protection systems. Westinghouse confirmed there were no changes in the functional requirements for fault protection (Ref. 35). I conclude that the accumulators are not part of the novel passive safety features of the **AP1000** plant.

109. However, for the duty function i.e. maintaining the accumulator pressure boundary, Westinghouse noted that a key difference was that the PXS and (including accumulators) were within containment, so that any radioactive release would be contained. This differed from older PWR plant designs where the ECCS systems were located outside of containment and so include a containment function. I acknowledge the change in duty functional requirements as a safety improvement compared to older PWR designs, but note that more recent PWR designs also include the accumulators within containment e.g. the EPR. In these designs the nuclear provisions for the accumulators are equivalent to ASME III Class 2. However, the ECCSs of EPR plant are partially outside containment such as legacy PWRs, unlike the **AP1000** reactor design.

110. Westinghouse pointed out that the PSA identified that operation of the accumulator check valves as the main source of unreliability of the PXS system (Ref. 35). In the **AP1000** plant the check valves are designed to ASME III Class 1 and they are expected to be more reliable than those in operating plants because:

- As identified in Ref. 29, the **AP1000** plant D-RAP requires an operating experience report by the manufacturer of the valves to list operating problems and how they were addressed; this report was not a part of the procurement program for operating plants.
- the **AP1000** plant in-service testing program for these valves is enhanced over that used for operating plants.

111. I discussed the Westinghouse claims with an ONR PSA specialist. ONR is satisfied based on the PSA insights that loss of the accumulator function to support the PXS delivery function during faults is likely to be dominated by the operability of the check valves. I note that this is also likely to be the case for the accumulators which support the ECCS function in plant with active protection systems.

112. At GDA Step 4, ONR's mechanical assessors judged that the Westinghouse proposed categorisation and classification for 'active' SSCs were in-line with expectations identified within the SAPs (ECS.1 and ECS.2). However, proposals to downgrade the

ASME nuclear provisions for Class C systems were not discussed. From a mechanical perspective, the assurance of material, manufacturing method, testing, examination and qualification of the SSC are key to providing confidence that the SSC will perform its safety function. I was informed that for the PXS the only active SSCs are the check valves which are assigned UK Class 1 and designed to ASME III Class 1. ONR is satisfied with the design provision for the accumulator check valves and that they will receive appropriate EIMT through-life commensurate with ASME III Class 1.

113. I am satisfied that for the **AP1000** plant, loss of the PXS delivery function during faults is likely to be dominated by the operability of the check valves. However, I consider this also to be the case for plant where protection against faults is delivered by active protection systems.

ASME Nuclear Provisions

114. Westinghouse carried out a comparison between their design and QA provisions for ASME III Class 2 and Class 3 components (Ref. 37); these are broadly equivalent for each class. I noted that the ASME III design requirements for Class 3 require only 'design by rule' and questioned the acceptability of this limitation. Westinghouse indicated that for the accumulators this was supplemented with some design by analysis to ASME III NB. Consequently I am satisfied that Westinghouse have identified suitable rules of ASME III for the design and QA of the accumulators.

115. However, the main implication associated with the assignment of the lower ASME III Class is that the commensurate ISI provisions under ASME XI are less stringent with ISI limited to periodic visual inspection (ASME III Class 3, IWD ASME XI) rather than volumetric inspection (ASME III Class 2, IWC ASME XI).

116. Table 1 of the ONR SAP provide a hierarchy of the level of protection and means of achieving defence in depth. The proposed change from ASME III Class 2 to 3 is one from prevention via the provision of sample volumetric inspection to provide forewarning of failure (Level 1) to the use of visual inspections to control of abnormal operation and detection of failures (Level 2) or to control faults within the design basis to protect against escalation (Level 3). Given the importance of the safety functions delivered by the PXS i.e. safety injection, I asked Westinghouse how the reduction in the level of defence in depth provision was justified, (Ref. 35).

117. Westinghouse acknowledged that the Level 1 defence/barrier is lower than that implemented in current operating plants, but claimed that the defence/barrier for Level 1 is adequate, to meet SAP EKP.3. The Westinghouse position centred on its view that SAP EKP.3 primarily covered the expectation for multiple levels of defence against significant faults or failures of components.

118. I acknowledge that SAP EKP.3 primarily covers the expectation for multiple layers of defence in depth. However, in my opinion the guidance underpinning SAP EKP.3 is also appropriate and the ONR expectation is that defence in depth should prevent faults (SAP paragraph 150). This is reflected in the aim for Level 1 protection, which is detailed in the IAEA Safety Requirements SSR2/1 (Ref. 39) on which Table 1 in the SAP is based. This is reinforced in SAP paragraph 156:

‘The availability and reliability of the safety measures should be commensurate with the significance of the radiological hazards being controlled and their safety functions within the defence in depth hierarchy (Principle EKP.3). In particular, mitigating safety measures (Level 4) should not be regarded as a substitute for fault prevention (Levels 1 and 2) or protection (Level 3) barriers, but as further defence in depth.

119. More generally, priority should be given to providing reliable and effective barriers

(inherent features, equipment and procedures earlier in the hierarchy) so that later barriers, though in place, need not be called upon.'

120. Westinghouse claimed that design of the accumulators to ASME Section III Class 3 instead of ASME Section III Class 2 did not affect their reliability and availability to deliver the safety functions (Ref. 35). This was based on operation to technical specifications (monitoring pressure and water level) in combination with leakage detection. Westinghouse also claimed that their proposals were consistent with the requirements for Standard Class 1 in their structural integrity classification document (Ref. 40):

'…….an adequate level of reliability through 'Achievement of Integrity' as a starting position. Thus, the safety case for components in this category is made on the basis of the quality of the design and build. A code assessment is sufficient and regular ISI is not required, but as a prudent measure speculative inspections should be considered. This provides some assurance against the unexpected.'

121. I consider full compliance with provisions of a recognised nuclear design code which includes the associated ISI provisions as fundamental to inferring that the level of structural integrity demonstration accords with the implied reliability for the safety case (ECS.3, EMC.3, and EMC.27). I therefore consider that for SSCs classified as Standard Class 1, ISI is likely to be required.

122. I recognise operation to technical specifications and leakage detection as supporting arguments to the structural integrity case, but in this case, they play a prominent role which does not accord with meeting UK expectations. (ECS.3). In particular, the guidance in TAST/16 states:

'In-service examinations should be carried out where they are reasonably practicable to enable the present condition of the structure to be confirmed, and to verify that the component or structure is behaving as the safety case assumes. In-service examination provides a means of assuring that components and structures remain at all times fit for purpose (EMC.27, EMC.28).'

123. Westinghouse cited operational experience (OPEX) data as a means to judge the benefits of volumetric examination. In brief, the evidence was based on historical data covering the integrity of accumulators and their associated piping in Westinghouse PWRs and other nuclear reactors from 1968 onward. The absence of gross failure or cracking was held to underpin their view that there was no benefit from volumetric inspection of the accumulators and PXS.

124. In my opinion the data was commensurate with that expected for pressure vessels built to good nuclear and non-nuclear standards. I also note the population of accumulators would include a proportion designed, built and inspected to the higher ASME nuclear provisions i.e. ASME III Class 2 and examination to ASME XI IWC. There was no evidence of failures or cracking in accumulators, but three events were reported where unexpected leakage had occurred in the connecting pipework systems. I conclude that the OPEX data was of limited value in underwriting a claim that the accumulator vessel failure rate would not be affected by downgrading of the ASME nuclear provisions.

125. Westinghouse also claimed that even with leakage at the start of an event requiring accumulator injection, the functions of providing core cooling could still be met. I do not dispute the view that minor leakage (or partial failure) is unlikely to affect the delivery of the safety functions. My concern is larger leakage and achieving forewarning or failure. In the UK the expectation is that the safety case is predicated on a demonstration of forewarning of failure. In this context, I am concerned as to whether there would be adequate knowledge of the condition of the accumulators and

PXS system though-life (Ref. 41). I subsequently confirmed with Westinghouse that in general there is access to inspect the accumulators and other Class C systems and that if further improvements are necessary to facilitate access these would involve minor design changes.

126. In summary, Westinghouse were placing the emphasis on the leakage monitoring and the safety management arrangements (tech specs) rather than using ISI as a means to forewarn against failure. As indicated above this approach does not accord with meeting UK expectations where the emphasis is placed on achieving forewarning to prevent failure using ISI.

ALARP Optioneering and Risk Reduction

127. The Westinghouse ALARP case for the accumulator design and through-life inspection centred on their view that there are significant detriments that are disproportionate to the benefits from imposing the higher inspection requirements of ASME XI Subsection IWC.

128. As discussed above the Westinghouse inspection proposals for ISI do not meet UK expectations, and I do not accept the Westinghouse claim that in-service volumetric inspection has no benefit in underpinning the reliability of the system (Ref. 41).

129. In an early submission, Westinghouse claimed there would be a large increase in the dose and inspection times over the 60 year design life (Ref. 12). Westinghouse subsequently confirmed that the added dose associated with a volumetric inspection of the accumulators was circa 0.055 mSv (Ref. 37). Having consulted with radiation protection specialists within ONR, I consider the Westinghouse dose estimate as indicative that the doses associated with through-life periodic volumetric inspection of the accumulators are not unduly excessive.

130. Westinghouse ALARP case for the accumulators was therefore heavily biased towards reducing the ASME ISI on the basis of costs. Notwithstanding my view that the Westinghouse proposals fall short of meeting UK expectations, I am not convinced that the Westinghouse ALARP case presents a balanced approach with a considered view of the benefits and detriments of retaining or reducing the ASME nuclear provisions. I also consider other options e.g. the use of ultrasonic inspection which may provide a more effective and efficient means to both meet UK expectations and demonstrate that risks are reduced ALARP.

AP1000 Plant Class C Systems

131. Westinghouse clarified that the scope of the Class C equipment covers the following (Ref. 38):

- The accumulator subsystem within the passive core cooling system (PXS)
- The in-containment refuelling water storage tank (IRWST) injection subsystem and containment recirculation subsystem within the PXS
- The passive containment cooling water storage tank (PCCWST) injection subsystem within passive containment cooling system (PCS)
- The downstream automatic depressurisation system (ADS) piping to the IRWST within the reactor coolant system (RCS) and PXS.

132. I conclude that the Class C systems relate to the delivery of water injection and depressurisation safety functions for the **AP1000** plant. The Class C systems therefore perform an emergency core cooling function following postulated design-basis events. As with the accumulators such systems would be expected to be

classified as ASME Class 2 in accordance with relevant guidance (Refs. 32 and 34).

133. The direct and indirect consequences of postulated gross failures of all the Class C systems are covered in the PCSR. Westinghouse confirmed that for all the Class C systems the only “pressure retaining” and non-atmospheric subsystem is the accumulator within the PXS. It follows that in the absence of significant stored energy other internal hazards, in particular, flooding, warrant consideration. The internal hazards flooding assessment is based on identification of all flooding sources and bounding consequences analysis. The analysis does not differentiate between Class 1, 2 or 3 source systems. The consequences analysis is based on gross failure. Therefore, Class C systems should have been captured by the analysis presented.

134. Westinghouse initially proposed a ‘system based’ approach to their justification of the downgrading of the ASME nuclear provisions (Ref. 12). This ‘system approach’ is akin to that used by Westinghouse to claim that the selection of non-nuclear codes is appropriate for UK Class 2 systems. With this approach ‘risk important active SSCs’ e.g. pumps and valves are claimed to dominate the risk of the loss of the function. It follows that for these systems the assessment of the design and through-life EIMT provisions for pumps and valves, which is the remit of ONR’s mechanical engineering discipline, formed a prerequisite to consideration of the pressure boundary components. I therefore sought clarity on which of the Class C systems included ‘risk important active SSCs’.

135. Westinghouse confirmed the majority of the Class C systems do not include ‘active SSCs’, which accords with the Westinghouse design philosophy of achieving passive safety. The **AP1000** plant Class C systems with ‘active SSCs’ were therefore limited to:

- PXS containment recirculation squib valves
- PCS motor operated valve (MOV) for PCCWST drainage

136. I discussed these active SSCs with ONR’s mechanical engineering specialists and I am content that Westinghouse have adequately classified these components.

ALARP Optioneering and Risk Reduction

137. Westinghouse proposed downgrading the ASME nuclear provisions (ASME III Class 3 and ASME XI Subsection IWD). An ALARP case for the individual Class C systems was not provided. Instead the classification was deemed appropriate by Westinghouse for similar reasons to those presented for the accumulators (Ref. 37):

- minor leakage would not compromise the delivery of the safety functions
- location of the Class C systems within containment would contain minor leakage
- the design, fabrication and QA standards imposed by Westinghouse are similar for ASME III Class 2 and 3.
- inspection provision is reduced to visual inspection during service because of the higher dose associated with the extra time to undertake volumetric inspection.

138. In my opinion further evidence is needed to underpin the ASME ISI that Class C systems are commensurate with reducing risks ALARP. The evidence should consider the benefits provided from greater ISI provisions for the mitigation of the direct and indirect consequences from failure.

4.2.3 Action GI-AP1000-SI-06.A3

139. I looked at each of the claims made along with the supportive arguments and checked if they were suitably underpinned by the relevant evidence. My assessment of the safety case is discussed below.

4.2.3.1 Postulated RCP Casing Failure

140. Westinghouse stated that the RCP is an ASME Section III Class 1 (Ref. 16) component and hence, failure of the pump bowl casing could not be discounted. They have assumed that the casing would disintegrate into finite number of fragments with kinetic energies.

141. Westinghouse has considered two methodologies to calculate the “bounding” fragment mass and the associated velocity under normal operating reactor condition. They have considered the BLEVE (Ref. 14) and the R3 (Ref. 15) methodologies for comparative determination of the “bounding” missile fragment and velocity.

142. The BLEVE method is based on disintegration of a spherical vessel due to overpressure, resulting in producing [redacted] fragments. The RCP bowl casing is like an inverted hemisphere containing pressured hot reactor coolant. Hence, it is reasonable to consider that the BLEVE methodology is applicable in this case. The R3 procedure states that ductile failures of a pressurised spherical vessel typically generates smaller number of large missiles and recommends a median fragment size of [redacted] The methodology is also applicable to RCP bowl casing given its hemispherical geometry. Considering both the BLEVE and R3 comparative determinations of the missile fragment size, I consider that assuming [redacted] size is conservative by comparison with the recommended missile size recommended in R3, and so is acceptable and in line with recognised good practice.

143. The velocities and kinetic energies of the [redacted] size “bounding” fragment have been calculated using the BLEVE (Ref. 14) and R3 (Ref. 15) methodologies and are stated in Table 5-1 of Reference 13. I have reviewed the data presented and by comparing, I conclude that the velocity ([redacted]) and kinetic energy ([redacted]) calculated following R3 procedure for spherical pressurised vessels is bounding and conservative for the bounding RCP bowl casing fragment size.

144. R3 (Ref. 15) is a well-recognised procedure in the UK nuclear industry for impact analysis using empirical solutions based on appropriate and relevant experiments / tests and research programmes. In my opinion R3 is regarded as a good source of RGP, as informed by the SAPs (Ref. 10).

4.2.3.2 Dynamic Analysis And Postulated SG Support Column Failure

145. The SG vertical support column is a built-up [redacted] of structural steel, connected to the SG at one end and a facility building structural column at the other. Westinghouse postulated that the bounding RCP casing fragment would impact on the SG vertical support column near the structural connection with the SG. The impact analysis predicted localised plastic strains developing in the column near the SG connection end, whilst remaining mostly elastic along the entire length. However, Westinghouse further postulated the loss of the vertical support column and claimed the integrity of the SG and the supporting hot and cold RCL piping.

146. In my opinion, the assumption of loss of the SG vertical support column is pessimistic, because, this configuration is most likely to produce maximum vertical displacement of the SG under gravity, potentially causing significant stresses / strains at the hot and cold piping connections. Figure 1 shows that the SG is supported laterally at several

locations, hence, any significant lateral displacement or rotation of the SG is highly unlikely following postulated loss of the vertical support column.

147. Westinghouse used ANSYS (Ref. 42), finite element dynamic analysis software, to calculate the dynamic thrust forces from escaping reactor coolant under "best estimate" (see further discussion in paragraph 152) normal operating pressure, following postulated failure of the RCP bowl casing and severance of the pipe connections (Figure 3). ANSYS is a well-recognised general purpose finite element tool for analysing dynamic systems and is used widely across various industries. Hence, in my opinion, ANSYS is fit for purpose for analysing the postulated fault.

148. The postulated fault occurs when the RCL piping is under normal operating reactor conditions (pressure and temperature). Figure 2 shows that due to postulated disintegration of the RCP casing bowl, the cold pipe connection to the RCP is severed as well as the connection with the SG. This results in thrust forces acting vertically and horizontally due to fast escaping reactor coolant under normal operating pressure and temperature. Further, the postulated loss of the SG vertical support column results in the SG being supported only by the hot pipe connection from the reactor pressure vessel (RPV) and the remnant cold pipe connecting the RPV and the neighbouring RCP bowl casing (Note – each SG is connected to a pair of RCPs). Westinghouse has used the following load cases to determine the structural integrity of the hot and cold RCL piping supporting the SG:

- Case 1 – RCP burst without vertical jet thrust, where the vertical SG jet load is ignored since it opposes gravity
- Case 2 – RCP burst without any jet thrust and column force, i.e., Case 1 rerun without any thrust or column loads
- Case 3 – RCP burst with vertical jet thrust, i.e., Case 1 rerun with the vertical SG jet thrust to evaluate effects on the SG supports

149. Considering the load cases stated above, firstly, the number of load cases is adequate since there are only two thrust loads acting on the system, and secondly, Case 1 is most conservative because postulated loss of vertical support column combined with ignoring the vertically upward thrust opposing gravity would result in maximum downward vertical displacement for the SG and the supporting RCL hot and cold piping. I judge that this would maximise the stresses and strains in the RCL hot and cold pipes and the associated nozzles.

150. Westinghouse has used the ASME Section III Level D ("faulted") Subsection F (Ref. 16) "plastic" analysis design criteria for evaluation of allowable stresses (Ref. 18). The RCP is an ASME Section III NB-3400 component (Chapter 20F of PCSR, Ref. 11) and Westinghouse has considered the initiating event, i.e., postulated disintegration of the RCP bowl casing as a low-probability event, although occurring under normal reactor operating conditions (pressure and temperature). Given that the RCP is a Class 1 component and its design acceptance shall ensure full ASME B&PV code compliance (material, design, manufacturing, inspection etc.), in my opinion it would be reasonable to consider that the casing failure is very unlikely under normal operating conditions (see further discussion in paragraph 152). It would therefore be reasonable to consider the postulated fault belonging to ASME Section III Level D class for enumeration of allowable stresses (Table 5-2 of Ref. 11) and loads (Table 5-3 of Ref. 11).

151. I have reviewed the assumptions and conservatism used in the ANSYS dynamic analysis and based on sampled observation stated below I am of the opinion that those assumptions and conservatism are valid and acceptable.

- Full reactor normal operating conditions, best estimate pressure and thermal stresses considering RCP as an ASME Section III NB-3400 component and the postulated fault being considered as a low probability event (see further discussion in paragraph 152);

- RCP missile fragment kinetic energy is fully transmitted (ignoring loss of strain energy due to ductile failure) to the column, with its weaker axis (lower flexural capacity) oriented along the direction of impact;

- The thrust loads are very short lived (in milliseconds), but for the postulated fault are assumed to remain constant in lieu of quick reduction due to depressurisation of reactor coolant. They are also assumed to be best estimate loadings, which is reasonable considering the pessimism on the “dwell time” factor;

- No pressure or thermal transient loads have been considered, since any transient load is generally considered as “secondary” in ASME B&PV and is most likely to be insignificant considering dominating effects of ASME Level D loads;

- For the dynamic analysis, the damping values used are conservative, using Rayleigh Damping, relative to seismic design damping levels. It is a reasonable assumption because the seismic forces would be much higher due to high inertia (“mass is the measure of inertia”) of the SG and RCP, when compared with the postulated fault.

152. I have reviewed the results of the ANSYS FE dynamic analysis in Reference 11 and observed the following:

- Displacements (Table 5-7) – the maximum displacement (e.g. vertical drop = [redacted] for the SG occurs under load Case 1, which is the most conservative load case due to postulated loss of the support column and ignoring the vertically upward thrust loading counteracting gravity. Thus it is reasonable to observe that maximum vertical displacement for the SG would occur at load Case 1;

- Primary component support loads (Table 5-8) – Load Case 1 is bounding, however, there are significant margins (> 30%) on both the vertical and lateral support loads, excepting one, however, which still remains acceptable (margin = 14%). Load Case 1 becomes bounding since it results in maximum downward vertical displacement for the SG and the attached pipework;

- RCL piping membrane and bending axial stresses (Table 5-9) –

    - Significant margins (>70%) for the hot and cold leg piping, with maximum stresses under load Case 1, which again is demonstrated to be bounding.

    - Westinghouse has also presented the ASME Section III Level B (“design”) stresses at those locations (Refs. 11 and 16), assuming existence of the RCP casing and the SG support column. I compared the Level D and Level B stresses in the table and observed that the factors range from [redacted] respectively. These factors are still higher when “normal” pressure (15.5 MPa) is compared with “design” (17.1 MPa, or 1.15 x “design”) and “overpressure” (23.4MPa, 1.5 x “design”). This shows that Level D loads are dominant over any other ASME B&PV primary load levels. Thus considering “best estimate” normal reactor operating conditions for the postulated fault is reasonable.

    - From the Level B results, it can be observed that the stresses / loads are symmetrically distributed between the hot and the cold pipes as would be expected of a geometrically balanced system. This also helps with understanding of how the system responds under the postulated fault starting from a steady balanced configuration and identifying any potential "cliff edge" effects;

- RCL piping total strains (Table 5-10) – significant margins exist for the dominant load case 1, when the predicted plastic strains are compared to 20% of the true ultimate strain for the material. The allowable strain estimate is reasonable considering the effect of multi-axial constraint effect on ductility;

- Primary nozzle stresses (Table 5-11) – significant margins (>89%) exist when the primary membrane and bending stresses (load case 1) are compared with the ASME Level D "plastic" allowable limit. Westinghouse has provided data for the Level B loadings and I observed that Level D stresses are generally dominant and show a similar trend as observed in Table 5-9 for the piping stresses.

- Steam generator secondary piping nozzles (Table 5-12) – Load case 1 is bounding and significant margins (>400%) exist when compared with the ASME Section III Level D allowable loads. This demonstrates the integrity of the main steam and feed water piping connections to the SG under the postulated fault (Figure 3), considering all the load cases.

- Westinghouse provided results from additional parametric analysis (e.g. unbroken column with RCP blast loadings, vertical thrust coefficients etc.) and studies in Table 5-13 to Table 5-17. All the results show significant margins when compared with the ASME Section III Level D plastic analysis criteria. These results provide additional confidence in the ANSYS analysis, helps with better understanding of the response of the system under the postulated fault and identify any potential "cliff edge" effect.

153. I reviewed the results of finite element (FE) impact analysis using a detailed 3 dimensional LS-DYNA (Ref. 43) model of the SG support column and the bounding RCP fragment (Section 5.2.7 of Ref. 13). Loads are modelled as imposed velocities on the FE model of the RCP casing fragment projectile (R3 based [redacted] mass) and conservatively allowed to strike normal to column's weaker axis (lower flexural / bending capacity). The results show that although the integrity of the vertical support column is maintained, however, energy is dissipated through local plastic deformation of column and other connection related structural components. I observed from Figure 5-13 of Reference 13 that majority of the column remains elastic and there is no gross structural failure.

4.2.3.3 Design Modification of the SG Support Column

154. Westinghouse proposed some further engineering options to make physical changes to the plant design to allow the SG vertical support column to better withstand impact from a fragment generated due to postulated RCP casing failure. Those options are:

- Additional missile barrier between the RCP and SG vertical support column – I observed following Westinghouse discussion that due to potential lack of available space in that area of the plant any such protection barrier would not only be difficult to install but would provide significant problems to EIMT activities. I agree that this could be a genuine problem during plant operations and accept Westinghouse decision to discount this option.

- SG support column material change to withstand RCP casing failure assessment – Table 5-20 in Reference 13 indicates the material suggested for the support column is ASME SA533 (current) in lieu of ASTM A588 (old) from the strength considerations only. Any further choice of material based on strength needs to also consider requirements for ductility to ensure proper functionality. Westinghouse stated that the current material for the vertical columns provides the right balance between the tensile strength and available ductility for the design basis events and any further change to the current material would result in revisiting the supporting structural analysis with associated time and cost implications. I judge and agree that revised justifications would be necessary because of the associated structural integrity related implications stated earlier.

- Enlarging SG support column – this would definitely increase the column capacity to withstand the impact from the RCP bounding fragment. However, Westinghouse stated that there is no space available for enlarging the support enough to ensure full elastic response for the column. In addition, the supporting analysis needs to be revisited with time and cost implications. I judge from the sketch in Figure 5-15 (Ref.11) that the space could be limited.

- Local strengthening of column built-up section near impact region – this is based on a suggestion from ONR (Refs.18, 19 and 20). Westinghouse has indicated a design modification to the existing support column to demonstrate improvement in the flexural capacity of the vertical support column. I have reviewed the argument and my assessment is stated below.

155. Westinghouse claims that the current build-up of the column contains a configuration that can be changed to provide a more robust design that provides the following:

- The number of welds used to fabricate the support is reduced in the proposed design. This is a key tenet of ONR SAP EMC.9 (Ref. 10) which states "The choice of product form of metal components or their constituent parts should have regard to enabling examination and to minimising the number and length of welds in the component."

- The welds of the proposed design are in locations that are more appropriate given the anticipated loadings. This is a key tenet of ONR SAP EMC.10 (Ref. 10) which states "The positioning of welds should have regard to high-stress locations and adverse environments."

156. Westinghouse provided an indicative sketch of the section of the vertical support column after the suggested design modification in Figure 4. I observed that additional web plates have been welded for [redacted] of the column, using the maximum available width of the flange plates. The revised cross section of the modified column is a [redacted] Westinghouse has calculated that the revised elastic modulus has increased by 97%, i.e., nearly by a factor of 2, whilst the plastic section modulus has increased by 66%. Considering that the additional web plates align with the extremities of the flange plates to create the [redacted] in my opinion, enhances the flexural capacity and structural stability of the column significantly.

157. Since the modified section does not take up any additional space over and above what was available before, hence I believe that the new proposed design is not expected to have a large impact of the interfacing SSCs and associated analyses.

158. I have observed that the additional plates do not contribute significantly to the cross section area, hence any effect on the axial stiffness, which is based on cross section area would be minimal. The additional plates add to the capacity of the column to accommodate for more bending (extreme fibre stress) along the weaker axis.

159. I therefore conclude that the design modification would benefit from providing a more robust design for the column support to withstand the postulated impact from the bounding RCP casing fragment, while not impacting the configuration of the interfacing components. I believe improved flexural capacity of the column would potentially reduce plastic strains developing in the column following impact and reduce the risks

of any potential failure of the SG vertical support column to ALARP, as informed by ONR SAPs.

160. My above judgement has been informed by the following safety principles EMC.3; EMC.7; EMC.11; EMC.13 and EMC.22 (Ref. 10), vide Table 1.

4.2.4 Key Assessment Considerations and Regulatory Judgements

GDA Issue GI-AP1000-SI-06.A1

161. GDA Issue GI-AP1000-SI-06.A1 relates to the provision of evidence to show that the principal codes and standards adopted for UK Class 2 pressure equipment and storage tanks are consistent with their implied reliability. In particular, where non-nuclear codes and standards are proposed for UK Class 2 pressure equipment and storage tanks, there is an expectation that Westinghouse will need to justify each case.

162. ONR expects the classification of SSC to be informed by the delivery of safety functions and to take cognisance of the significance to safety and other factors. The classification should consider both the delivery of functions to protect against faults and the duty function in this case maintaining the pressure boundary.

163. In the case of pressure equipment and storage tanks there are nuclear design and construction codes available e.g. ASME III. Thus applying non-nuclear codes for the design and construction of UK Class 2 pressure equipment and storage tanks does not meet ONR’s normal expectations.

164. For passive plants such as the **AP1000** design, the US NRC uses RTNSS guidance to address those non-safety systems in passive plants that would normally be safety related in a non-passive plant. However, in the US, the focus for regulatory guidance is on availability controls and not codes/standards. Furthermore, the guidance is heavily reliant on probabilistic risk assessment. The output is a graded, risk informed approach to Technical Specifications, whereas for the UK the expectation is a graded approach to category, classification and codes/standards.

165. The Westinghouse use of a ‘system-based approach’ for UK Class 2 meant that the selection of codes and standards is a cross-discipline matter beyond consideration of the pressure boundary integrity under the structural integrity discipline.

166. Through regulatory queries and the comprehensive and detailed responses provided by Westinghouse, ONR established a way forward with Westinghouse that would provide the means to gain the evidence that their selection of codes and standards was appropriate and commensurate with reducing risk so far as is reasonably practicable.

167. ONR is satisfied that Westinghouse has developed an adequate process to collate the evidence necessary to inform a reasoned judgement on the selection of the codes and standards for UK Class 2 pressure equipment and tanks for the GDA. In addition, ONR has identified some additional points that need to be considered to fully meet UK expectations.

168. It is my expectation that all UK Class 2 SSCs are designed, manufactured, constructed, installed, commissioned, quality assured, maintained, tested and inspected to appropriate nuclear (or equivalent) codes and standards. I acknowledge that Westinghouse has developed a more refined process to establish the safety-significance of individual UK Class 2 SSCs, but that this has yet to be fully implemented. It is my opinion that this process should and can continue in the licensing phase. I consider it necessary that classification is of sound basis to identify where application of nuclear (or equivalent) codes and standards are warranted. I also consider it possible that several SSCs, presently classified by Westinghouse as UK Class 2, may not necessarily merit application of such standards by virtue of their safety significance. These matters relate to licensee operational choices, therefore I raise the following Assessment Finding:

CP-AF- AP1000-SI-17 - The licensee shall complete the refined review of UK Class 2 pressure equipment (including pumps and valves) and storage tanks and demonstrate that the codes and standards (including EIMT) applied are commensurate with the safety classification and UK regulatory expectations.

GDA Issue GI-AP1000-SI-06.A2

169. ONR expects the classification of SSC to be informed by the delivery of safety functions and to take cognisance of the significance to safety and other factors (ECS.2). The classification should consider both the delivery of functions to protect against faults and the duty function in this case maintaining the pressure boundary. In my opinion for the accumulator subsystem, the safety functional requirements, in terms of fault protection, are similar to plant with active protection systems.

170. I am satisfied that for the **AP1000** plant loss of the PXS delivery function during faults is likely to be dominated by the operability of the check valves. However, I consider this also to be the case for plant where protection against faults is delivered by active protection systems.

171. Westinghouse identified a difference in the duty function for the PXS compared to older PWR designs. In particular, the PXS is located within containment. I regard this as a safety improvement for the PXS as a whole, , but also identify the similarities in the accumulator fault protection functional requirements when compared to modern designs of PWR plant.

172. Overall, my view is that the safety functional requirements for the **AP1000** plant accumulators are similar to modern PWR plant with active protection systems. However, their ECCSs are partially outside containment such as legacy PWRs, unlike the **AP1000** reactor design.

173. Westinghouse has demonstrated the design and QA provisions to either ASME III Class 2 or 3 as specifically applied to the accumulators in the **AP1000** plant are broadly equivalent. The main implication associated with the assignment of the lower ASME III Class is that the commensurate ISI provisions under ASME XI are less stringent; with ISI limited to periodic visual inspection (ASME III Class 3, IWD ASME XI) rather than volumetric inspection (ASME III Class 2, IWC ASME XI).

174. I consider that full compliance with provisions of a recognised nuclear design code which includes the associated ISI provisions is fundamental to inferring that the level of structural integrity demonstration accords with the implied reliability for the safety case (ECS.3, EMC.3, and EMC.27). In UK civil nuclear practice the emphasis is placed on ISI to provide forewarning of failure with leakage monitoring and the safety management arrangements (tech specs) providing useful support to the safety case. Westinghouse has provided evidence in Ref. 40 that direct and indirect consequences of gross failure are not a concern for Category A functions to be met per the fault studies and internal hazards areas.

175. Notwithstanding my view that the Westinghouse ISI proposals fall short of meeting UK expectations, I am not convinced that the Westinghouse ALARP case presents a balanced approach with a considered view of the benefits and detriments of retaining or reducing the ASME nuclear provisions. I also consider other options e.g. the use of ultrasonic inspection may provide a more effective and efficient means to both meet UK expectations and demonstrate that risks are reduced ALARP. My assessment also identified that further evidence was needed to justify that the ASME provisions for the other Class C systems were commensurate with reducing risks ALARP.

176. The main shortcoming relates to the through-life ISI provision; this is an operational matter for the licensee to address. I consider UK expectations for the **AP1000** plant

Class C systems are not onerous and are readily achievable without significant design change. I therefore raise the following assessment finding for the Licensee to provide additional evidence during licensing.

CP-AF-AP1000- SI-18 – The licensee shall apply ISI for the accumulator subsystem and other Class C systems, structures and components that are commensurate with UK expectations and relevant good practice.

GDA Issue GI-AP1000-SI-06.A3

177. Based on my assessment of the relevant aspects of the PCSR (Ref. 11) and Reference 13, I am content that Westinghouse has provided sufficient evidence that the effectiveness of the SG vertical support will not be challenged by the failure of the pump bowl in order to support the assignment of a Standard Class 1 structural integrity classification for the pump bowl. The key regulatory judgements underpinning my conclusion are the following:

- Westinghouse has determined the most damaging [redacted] fragment size [redacted] is most conservative following R3, with bounding velocity [redacted] and kinetic energy [redacted] .
- Westinghouse has demonstrated that the integrity of the hot and cold piping connected to the SG would be maintained under the postulated fault of failure of the vertical support column following impact from the "bounding" RCP fragment.
- Westinghouse has indicated that the design modification to the SG vertical support column would benefit from providing a more robust design for the support to withstand the impact from the RCP casing fragment, while not impacting the configuration of the interfacing components. The improved flexural capacity of the column would potentially reduce plastic strains developing in the column following impact and further minimise the risks of any potential failure of the SG vertical support column to ALARP, as informed by ONR SAPs.

4.3 Comparison with Standards, Guidance and Relevant Good Practice

178. Section 2.2 of this report identifies standards, guidance and RGP that has informed my assessment, which is described in Section 0. In particular, my assessment has been guided by ONR's SAPs, see Table 1, and TAGs, see Table 2. Notable example of RGP adopted by Westinghouse include application of the R3 procedure (Ref.15) for dynamic analysis of the effect of postulated gross failure of the RCP bowl casing on the SG vertical support, and application of Section III of the ASME Code (Ref. 16) for stress analysis of RCL piping.

4.4 Assessment findings

179. During my assessment 2 items were identified for a future licensee to take forward in their site-specific safety submissions. These are collated in Annex 1.

180. 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.

181. 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; and
- to resolve this matter the plant needs to be at some stage of construction / commissioning.

5 CONCLUSIONS

182. This report presents the findings of the assessment of GDA Issue GI-AP1000-SI-06 relating to the **AP1000** plant GDA closure phase.

183. To conclude:

- Westinghouse has provided sufficient evidence that a categorisation and classification process has been developed which should ensure that those SSCs with an important structural integrity claim are assigned a code or standard in line with ONR's expectations.
- Westinghouse has provided sufficient evidence that components in **AP1000** plant Equipment Class C have been assigned a class that is consistent with their duty and implied reliability.
- Westinghouse has provided sufficient evidence that catastrophic failure of a reactor coolant pump bowl would not challenge the effectiveness of the vertical support for the steam generator.
- I am satisfied that GDA Issue GI-AP1000-SI-06 can be closed.

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

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

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37. UKP-GW-GL-106, AP1000 Plant Safety Class B and C and Accumulated Design ALARP Assessment, Rev. 1, 17th March 2016, TRIM Ref. 2016/119738.

38. RQ-AP1000-1785, Plant Safety Class B & C – Accumulator Design ALARP Assessment (SI06 A2), Full Response, 20th January 2017, TRIM Ref. 2017/29358

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43. LS Dyna – General Purpose Built Finite Element Simulation Software (http://www.lstc.com/products/ls-dyna)

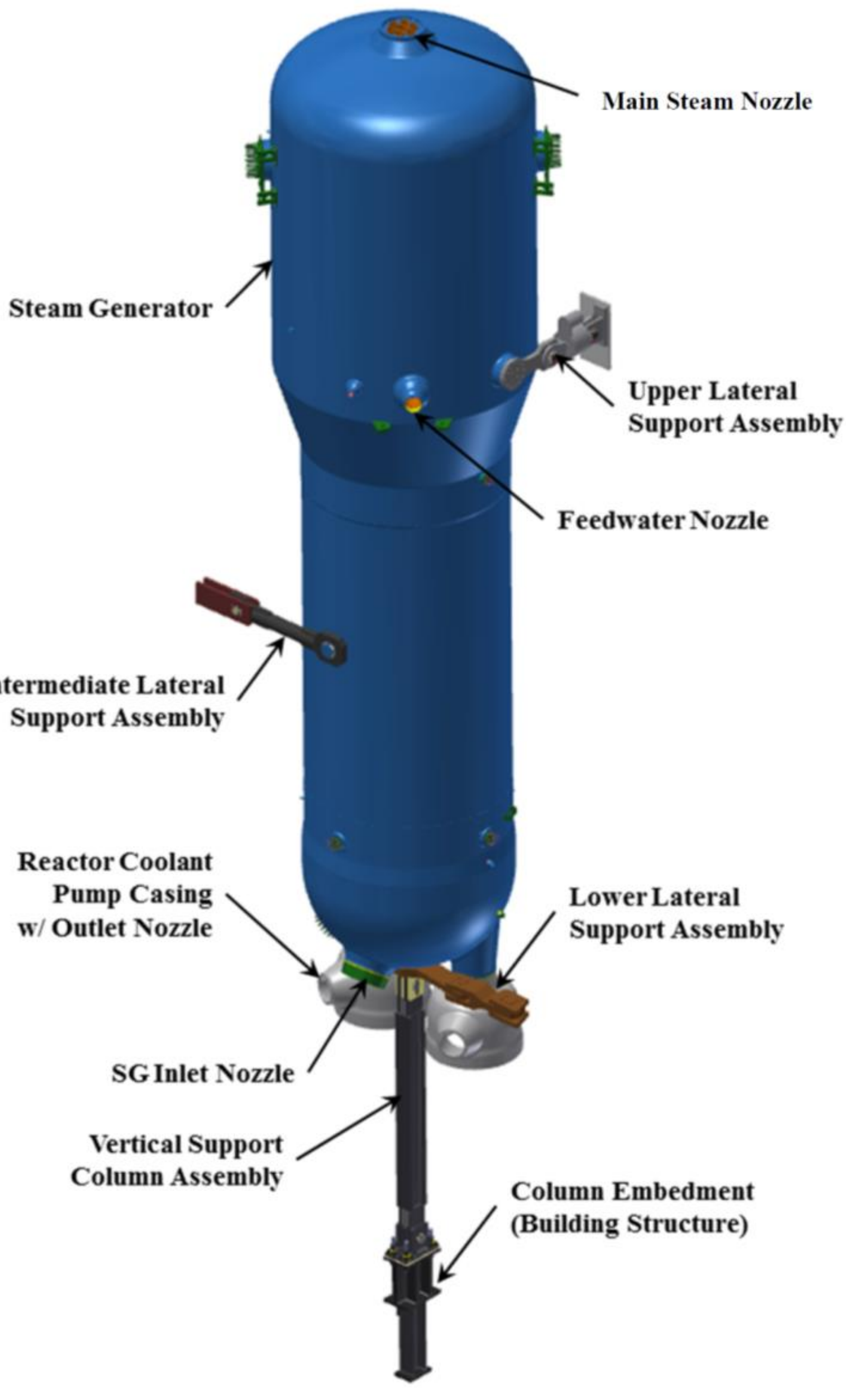

Figure 1: AP1000 Plant Steam Generator Support System General Assembly

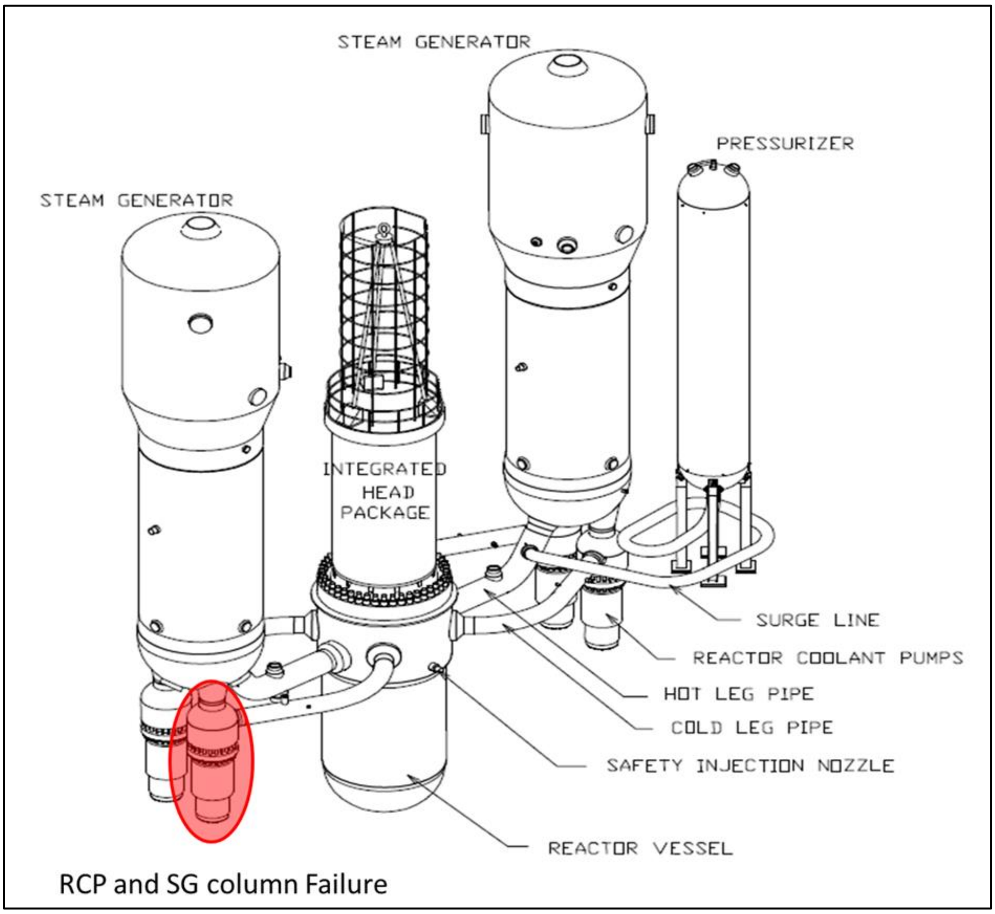

Figure 2: Reactor Coolant Loop Isometric View

[redacted]

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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.
EAD.1	Engineering principles: ageing and degradation	Safe working life
EAD.2	Engineering principles: ageing and degradation	Lifetime margins
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.3	Engineering principles: maintenance, inspection and testing	Type-testing
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.1	Engineering principles: safety classification and standards - Safety categorisation	The safety functions to be delivered within the facility, both during normal operation and in the event of a fault or accident, should be identified and then categorised based on their significance with regard to safety.
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 their 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.3	Engineering principles: integrity of metal components and structures: highest reliability components and structures	Evidence
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.

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SAP No	SAP Title	Description
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 their 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 their constituent parts should have regard to enabling examination and to minimising the number and length of welds in the component.
EMC.11	Engineering principles: integrity of metal components and structures: design	Failure Modes
EMC.13	Engineering principles: integrity of metal components and structures: manufacture and installation	Materials
EMC.22	Engineering principles: integrity of metal components and structures: operation	Material compatibility
EMC.26	Engineering principles: integrity of metal components and structures: monitoring	Detailed assessment should be carried out where monitoring is claimed to provide forewarning of significant failure.
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.32	Engineering principles: integrity of metal components and structures: analysis	Stress Analysis
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 their 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.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.
EHA.1	Engineering principles: external and internal hazards: Identification and characterisation	An effective process should be applied to identify and characterise all external and internal hazards that could affect the safety of the facility.
EHA.2	Engineering principles: external and internal hazards: Data sources	For each type of external hazard either site-specific or, if this is not appropriate, best available relevant data should be used to determine the relationship between event magnitudes and their frequencies.

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SAP No	SAP Title	Description
EHA.3	Engineering principles: external and internal hazards: Design basis events	For each internal or external hazard which cannot be excluded on the basis of low frequency or insignificant consequence (see Principle EHA.19), a design basis event should be derived.
EHA.4	Engineering principles: external and internal hazards: Frequency of initiating event	For natural external hazards, characterised by frequency of exceedance hazard curves and internal hazards, the design basis event for an internal or external hazard should be derived to have a predicted frequency of exceedance that accords with Fault Analysis Principle FA.5. The thresholds set in Principle FA.5 for design basis events are 1 in 10 000 years for external hazards and 1 in 100 000 years for man-made external hazards and all internal hazards (see also paragraph 629).
EHA.5	Engineering principles: external and internal hazards: Design basis event operating states	Analysis of design basis events should assume the event occurs simultaneously with the facility's most adverse permitted operating state (see paragraph 631 c) and d)).
EHA.6	Engineering principles: external and internal hazards: Analysis	The effects of internal and external hazards that could affect the safety of the facility should be analysed. The analysis should take into account hazard combinations, simultaneous effects, common cause failures, defence in depth and consequential effects.
EHA.7	Engineering principles: external and internal hazards: 'Cliff-edge' effects	A small change in design basis fault or event assumptions should not lead to a disproportionate increase in radiological consequences.
EHA.14	Engineering principles: external and internal hazards: Fire, explosion, missiles, toxic gases etc. – sources of harm	Sources that could give rise to fire, explosion, missiles, toxic gas release, collapsing or falling loads, pipe failure effects, or internal and external flooding should be identified, quantified and analysed within the safety case.

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Table 2

Technical Assessment Guides Considered in the Assessment

Technical Assessment Guide No	Description
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-016	Integrity of Metal Components and Structures
NS-TAST-GD-009	Examination, Inspection, Maintenance and Testing of Items Important to Safety
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

National and International Standards and Guidance

Description
ASME B&PV Section III
Impact Assessment Procedure R3
ANSI/ANS 51.1 1983 Edition, Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants
ANSI/ANS-58.14-2011: Safety and Pressure Integrity Classification Criteria for Light Water Reactors

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Annex 1

Assessment Findings: GDA issue GI-AP1000-SI-06 – Structural Integrity Categorisation and Classification

Number	Assessment Finding	Section
CP-AF- AP1000-SI-17	The licensee shall complete the refined review of UK Class 2 pressure equipment (including pumps and valves) and storage tanks and demonstrate that the codes and standards (including EIMT) applied are commensurate with the safety classification and UK regulatory expectations.	4.2.4
CP-AF- AP1000- SI-18	The licensee shall apply ISI for the accumulator subsystem and other Class C systems, structures and components that are commensurate with UK expectations and relevant good practice.	4.2.4