New Reactors Programme

GDA close-out for the AP1000 reactor

Internal Hazards GDA Issues GI-AP1000-IH-01 to IH-06

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EXECUTIVE SUMMARY

Westinghouse Electric Company (Westinghouse) is the reactor design company for the **AP1000®** reactor, and the Requesting Party (RP) for a Generic Design Assessment (GDA) by the UK Office for Nuclear Regulation (ONR). The **AP1000** reactor completed Generic Design Assessment (GDA) Step 4 in 2011 and the RP paused the regulatory process. At that time the GDA had resulted in the granting of an Interim Design Acceptance Confirmation (IDAC), which had 51 GDA issues attached to it. These issues require resolution prior to the award of a Design Acceptance Confirmation (DAC) and before any nuclear safety-related construction can begin on site. The RP re-entered GDA in 2014 to close the 51 issues.

This report is the Office for Nuclear Regulation's assessment of the RP's **AP1000** reactor design in the area of internal hazards. Specifically this report addresses GDA issues:

- GI-AP1000-IH-01 GDA Issue – Internal Fire Safety Case Substantiation;
- GI-AP1000-IH-02 GDA Issue – Internal Flooding Safety Case;
- GI-AP1000-IH-03 GDA Issue – Pressure Part Failure;
- GI-AP1000-IH-04 GDA Issue – Internal Explosion Safety Case Substantiation.
- GI-AP1000-IH-05 GDA Issue – Internal Missile Safety Case;
- GI-AP1000-IH-06 GDA Issue – Substantiation and Analysis of the Consequences of Dropped Loads and Impact from Lifting Equipment Included Within the **AP1000** Design.

The GDA issues actions, which were raised in Step 4 relating to the internal hazards aspects of the **AP1000** design and safety case were due to lack of sufficient information, which limited the extent of ONR's Step 4 assessment of internal hazards, coupled by inaccurate or inconsistent information. A number of areas were identified where the internal hazards safety case presented failed to adequately address the requisite claims, arguments and evidence.

Early in the closure phase of these GDA issues, the RP recognised the need to review and re-issue all claims, arguments and evidence for the internal hazards safety cases related to each of these GDA issues. The RP's GDA issues Resolution Plans included the following:

- Development of individual Topic Reports, which summarise the claims, arguments and evidence for each GDA issue. These Topic Reports support the revised Pre-Construction Safety Report (PCSR) Chapter 11 on internal hazards.
- Development of consequences analysis for each GDA issue, the outcome of which were captured in the Topic Reports and PCSR.
- Development of a number of lower tier documents to support the individual Topic Reports.

My assessment focused on the suitability and sufficiency of the claims, arguments and evidence presented in the Topic Reports and in the revised PCSR for closing out the six GDA issues related to internal hazards. I gave particular focus on the identification of initiating events, analysis methodologies and criteria, consequences analysis, suitability of the engineering safety measures, and on the adequacy of the redundancy, segregation and separation principles adopted in the **AP1000** plant design.

In the area of internal fire the RP undertook significant fire analysis using different modelling techniques to justify the bounding fire case. This was subsequently used in the substantiation of concrete fire barriers and the steel-concrete-steel fire barriers.

The RP also undertook a gap analysis on the design of fire dampers between the UK and US codes and standards. It identified a design change proposal (DCP) that addressed a number of shortfalls which will be implemented during the site licensing phase of the UK **AP1000** reactor project. A

In the area of internal flooding, the RP presented a revised safety case on internal flooding. The RP has undertaken a systematic identification of flooding scenarios and consequences analysis based on gross failure, which culminated in the derivation of specific claims. The claims were supported by the requisite arguments and evidence.

In the area of pressure part failure, the RP responded positively to my challenge and revised its design criteria to meet ONR expectations. This now requires that they undertake additional gross failure analysis for a large number of high energy lines. The RP recognising that they could not complete the amount of analysis for all these lines within the timescales available for GDA closeout, agreed to provide examples of their analysis of some representative case studies to build ONR confidence in their approach and to demonstrate that the overall design will be fundamentally unaffected. Full implementation of the revised design criteria can only be completed post GDA and during detailed analysis. Based on qualitative arguments and the limited amount of completed analyses available for high risk scenarios, I was able to gain sufficient confidence that the full implementation of the revised design criteria should not result in major design modifications. Overall, the submissions provided information relating to the process and methodology used in the identification of pressure part failure events, characterisations of the consequences and identification of safety measures.

In the area of internal explosion, the RP significantly revised its safety case to address the GDA issue. This included revised claims, arguments, and evidence for both the battery rooms as well as the routing of hydrogen pipework.

In the area of internal missile, the RP undertook a systematic identification of all potential internal missiles and characterisation of the consequences analysis. Revised claims arguments and evidence were presented.

In the area of dropped load, the RP identified all lifting devices and undertook a systematic drop load identification study. Bounding dropped load consequences analysis presented, which aided the development of suitable claims, arguments and evidence.

I concluded that the Topic Reports and revised PCSR provide the requisite information relating to the identification of potential initiating events, consequences analysis and the identification of safety measures. Suitable and sufficient claims have been made and these were supported by the underpinning arguments and evidence.

My conclusion is based upon the following factors:

- holding regular interventions and workshops with the RP;
- challenging and influencing the RP to revise its design criteria on flooding, pressure part failure, internal missiles and dropped load in line with the relevant good practice established in the UK;
- ensuring consistency in the assessment criteria of pressure part failure between internal hazards, structural integrity and fault studies;
- achieving convergence on UK regulatory expectations on internal fire, internal flooding, pressure part failure, internal explosions, internal missiles and dropped loads;
- challenging the RP on the qualitative and quantitative consequences analysis undertaken for all areas including the computational modelling analysis;
- assessment of the Topic Reports and PCSR Chapter 11 to gain confidence that the claims are suitable and sufficient, and that they are supported by robust arguments and evidence;
- challenging the RP via targeted Regulatory Queries (RQs), which influence and improve the safety case submissions for all areas;
- adopting a positive and reactive approach by the RP in addressing the GDA issues. This led in additional documentation in response to my concerns;
- liaising with ONR specialist disciplines of civil engineering, fault studies, structural integrity and probabilistic safety assessment to maintain consistency and clarify interfaces between our assessments.

I raised 11 Assessment Findings and one minor shortfall for a licensee to consider and take forward in their site-specific safety submissions. These matters do not undermine the generic safety submission, but are at the next stage of the project and may require licensee input/decision.

I am satisfied that GDA Issues GI-AP1000-IH-01 to IH-06 and associated actions can be closed.

LIST OF ABBREVIATIONS

ADS	Automatic Depressurisation System
AF	Assessment Finding
ALARP	As Low As Reasonably Practicable
APP	American AP1000 (standard) plant
BEZ	Break Exclusion Zone
BSL	Basic Safety level
CA	(Structural Module naming convention)
CAPAL	Corrective Action, Prevention and Learning
CAS	Compressed and Instrument Air System
C&I	Control and Instrumentation (ONR naming convention)
CCS	Component Cooling Water System
CFAST	Consolidated Fire and Smoke Transport model
CFD	Computational Fluid Dynamics
cfm	Cubic feet per minute
CPP	China AP1000 plant
CMT	Core Make Up Tank
CRDM	Control Rod Drive Mechanism
CVS	Chemical and Volume Control System
DAC	Design Acceptance Confirmation
DAS	Diverse Actuation System
DBA	Design Basis Accident
DBE	Design Basis Events
DBL	Low Probability Design Basis Events
DB1	Infrequent Design Basis Events,
DB2	Frequent Design Basis Events
DN	Diameter Nominal
DCP	Design Change Proposal
DID	defence in depth
DRP	Design Reference Point
DVI	Direct Injection Line
DWS	Demineralised Water Transfer and Storage System
ECS	Main AC Power System
FDS	Fire Dynamics Simulator
FEA	Finite Element Analysis
FHS	Fuel Handling System
FPS	Fire Protection System

GDA	Generic Design Assessment
gpm	US gallons per minute
GRS	Gesellschaft für Anlagen- und Reaktorsicherheit
HE	High Energy
HEAF	High energy arc fault
HSS	High Safety Significance
HVAC	Heating, ventilation and air conditioning
IAEA	The International Atomic Energy Agency
I&C	Instrumentation and Control (RP naming convention)
IDAC	Interim Design Acceptance Confirmation
IDS	Class 1 DC and Uninterruptible Power Supply
IEEE	Institute of Electrical and Electronics Engineers.
IH	Internal Hazards
IHP	Integrated Head Package
IoF	Incredibility of Failure
IRWST	In-containment Refuelling Water Storage Tank
LBB	Leak Before Break
LOCA	Loss of Coolant Accident
LOOP	Loss of offsite power
LFL	Lower Flammability Limit
LPT	Low Pressure Turbine
MCR	Main Control Room
ME	Medium Energy
MELB	Medium Energy Line Break
MEM	Multi-Energy Method
MSIV	Main Steam Isolation Valve
mSv	Millisievert
NFPA	National Fire Protection Association
NI	Nuclear Island
NPS	Nominal Pipe Size
ONR	Office for Nuclear Regulation
PAR	Passive Autocatalytic Recombiner
PCCWST	Passive Containment Cooling Water Storage Tank
PCS	Passive Containment Cooling System
PCSR	Pre Construction Safety Report
PGS	Plant Gas System
PMS	Protection and Safety Monitoring System
PPF	Pressure Part Failure
PRHA	Pipe Rupture Hazards Assessment

PSA	Probabilistic Safety Analysis
PSS	Primary Sampling System
PWR	Pressurized Water Reactor
PWS	Potable Water System
PXS	Passive Core Cooling System
RCA	Radiologically Controlled Area
RCCA	Rod Cluster Control Assembly
RCDT	Reactor Coolant Drain Tank
RCS	Reactor Coolant System
RGP	Relevant Good Practice
RNS	Normal Residual Heat Removal System
RP	Requesting Party
RPV	Reactor Pressure Vessel
RQ	Regulatory Query
SAPs	Safety Assessment Principles
SC	Steel-Concrete-Steel Composite
scfm	Standard cubic feet per minute
SDS	Sanitary Drainage System
SFAIRP	So Far As Is Reasonably Practicable
SFP	Spent Fuel Pool
SFS	Spent Fuel Pool Cooling System
SGS	Steam Generator System
SSC	System, Structure (and) Component
TAG	Technical Assessment Guide
TR	Topic Report
TSC	Technical Support Contractor
UKP	UK AP1000 plant
VAS	Radiologically Controlled Area Ventilation System
VBS	Nuclear Island Non-Radioactive Ventilation System
VCS	Containment Recirculation Cooling System
VES	Main Control Room Emergency Habitability System
VFS	Containment Air Filtration System
VLS	Containment Hydrogen Control System
VWS	Central Chilled Water System
WENRA	The Western European Nuclear Regulators' Association
WGS	Gaseous Radwaste System
WLS	Liquid Radwaste System
ZOI	Zone of Influence

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Annex(es)

1. INTRODUCTION

1.1 BACKGROUND

1. Westinghouse completed GDA Step 4 in 2011 and paused the regulatory process. It achieved an IDAC which had 51 GDA issues attached to it.

2. During Step 4 of the GDA, ONR reviewed the safety aspects of the proposed reactor designs by examining the evidence, supporting the claims and arguments made in the safety documentation, building on the assessments already carried out for Steps 2 and 3, and to make a judgement on the adequacy of the internal hazards information contained within the Pre-construction Safety Report (PCSR) and supporting documentation.

3. During Step 4, the 2009 PCSR was found to have significant shortfalls in terms of content and quality. Recognising the shortfalls with the 2009 PCSR, the RP submitted a replacement draft PCSR in December 2010, which extensively restructured and enhanced the 2009 PCSR in order to address ONR's concerns. The RP then submitted an approved PCSR in March 2011, but this was too late for a meaningful assessment during Step 4 (Internal Hazards Step 4 Assessment Report, Ref. 13).

4. The Internal Hazards Step 4 Assessment Report concluded that "*there are areas where the safety case presented for internal hazards fails to adequately address the requisite claims, arguments, and evidence which has resulted in the generation of 6 GDA Issues comprising of a total of 9 GDA Issue Actions*" (Ref. 13).

5. In summary these relate to:

- Substantiation of the barriers in place to prevent fire spread affecting more than one train or division and the need to substantiate fire damper provision (GI-AP1000-IH-01).
- Provision of a revised safety case for internal flooding. The RP identified shortfalls in the claims, arguments and evidence included within the PCSR issued previously (GI-AP1000-IH-02).
- Identification and substantiation of all nuclear significant pipe whip restraints, barriers and shields claimed for the protection of redundant trains against the effects of pressure part failure (GI-AP1000-IH-03).
- Provision of substantiation to support claims and arguments made within the area of internal explosion, specifically associated with hydrogen generation within battery rooms and the distribution of hydrogen within areas containing Class 1 Structures, Systems and Components (SSC) (GI-AP1000-IH-04).
- Identification and substantiation of the claims, arguments and evidence that constitute the internal missile aspects of the internal hazards safety case (GI-AP1000-IH-05).
- Substantiation including supporting analyses of the consequences of dropped loads and impact from lifting equipment included within the **AP1000** design (GI-AP1000-IH-06).

6. The internal hazards Step 4 Assessment Report also concluded that "*there are no fundamental reasons for believing that a satisfactory safety case cannot be made for the generic **AP1000** reactor design, subject to satisfactory progression and resolution of GDA Issues to be addressed during the forward work programme for this reactor. It must also be recognised that some of these GDA Issues may ultimately require changes to the plant design*" (Ref. 13).

7. These issues, therefore, require resolution prior to the award of a DAC and before any nuclear safety-related construction can begin on site. The RP re-entered GDA in 2014 to close the 51 issues.

8. The ONR Step 4 Assessment Report (Ref. 13) also identified a number of Assessment Findings that will be carried forward with a future licensee as part of normal regulatory business. Resolution of these Assessment Findings are outside the scope of GDA.

9. This report is the ONR's assessment of the RP's **AP1000** reactor design in the area of internal hazards. Specifically this report addresses the internal hazards GDA Issues GI-AP1000-IH-01 to IH-06.

10. The related GDA Step 4 report is published on our website (**www.onr.org.uk/new-reactors/ap1000/reports.htm**), and this provides the assessment underpinning the GDA issue. Further information on the GDA process in general is also available on our website (**www.onr.org.uk/new-reactors/index.htm**).

1.2 SCOPE

11. The scope of this assessment is detailed in my assessment plan (Ref. 3).

12. As stated previously, ONR's Step 4 Assessment Report did not assess the RP's approved PCSR in March 2011. The RP recognised that given the scope of the 6 GDA issues it will be prudent to revise the claims, arguments and evidence and submit a revised PCSR section and new individual Topic Reports for each GDA issue.

13. The RP provided a variety of safety and design material to address the GDA issues. This included the PCSR, Topic Reports and lower tier supporting documents covering the six GDA issues.

14. The scope included the examination of the evidence, supporting arguments, and claims made in the relevant new or previously un-assessed safety and design material. This included a review of relevant Design Change Proposal (DCPs) since 2010 that may have an impact on the PCSR or on internal hazards issues.

15. The scope of assessment focused on:

- GI-AP1000-IH-01 GDA Issue – Internal Fire Safety Case Substantiation;
- GI-AP1000-IH-02 GDA Issue – Internal Flooding Safety Case;
- GI-AP1000-IH-03 GDA Issue – Pressure Part Failure;
- GI-AP1000-IH-04 GDA Issue – Internal Explosion Safety Case Substantiation;
- GI-AP1000-IH-05 GDA Issue – Internal Missile Safety Case;
- GI-AP1000-IH-06 GDA Issue – Substantiation and Analysis of the Consequences of Dropped Loads and Impact from Lifting Equipment Included Within the **AP1000** Design.

16. My assessment is therefore focused on GI-AP1000-IH-01 to GI-AP1000-IH-06 and associated actions. This assessment complies with internal guidance on the mechanics of assessment within ONR (NS-PER-GD-014, Ref. 1).

1.3 SAMPLING STRATEGY

17. ONR adopts an assessment strategy of sampling. Sampling is used to limit the areas scrutinised, to improve the overall efficiency of the assessment process, and to reveal generic weaknesses in the safety case as a whole. Samples are drawn from areas of high safety relevance since weaknesses in these areas are potentially very serious, but also from lower significance areas to check for possible oversight by the RP.

18. The scope of my assessment to close out the GDA issues is focused on the specific actions raised in the Resolution Plans (Refs 18 to 23) as addressed in the PCSR, Topic Reports and supporting documents.

19. My sampling strategy for this assessment is focused on the following:

- on the suitability and sufficiency of the claims arguments and evidence as captured in Chapter 11 of the PCSR and associated Topic Reports;
- on the passive Class 1 SSCs delivering Category A safety functions, located within the Nuclear Island (NI);
- adequacy of internal hazards identification, consequences analysis, and adequacy of principles of redundancy, segregation and separation within the **AP1000** design.

20. I also undertook some limited sampling on the following:

- Class 2 systems delivery Category B safety functions; mainly the Normal Residual Heat Removal System (RNS) and the Diverse Actuation System (DAS) systems. The DAS and RNS systems provide diverse means of safeguarding against frequent internal hazards;
- the postulated events and consequences analysis for the Turbine Building focusing on fire and turbine disintegration. Such an event may compromise delivery of Category A safety functions;
- internal explosion generated in areas outside of the NI.

1.4 STRUCTURE OF ONR ASSESSMENT REPORT

More details and assessment of RP’s safety case on each of the internal hazards GDA issues are provided in the following sections:

- section 4 – internal fire;
- section 5 – internal flood;
- section 6 – pressure part failure;
- section 7 – internal explosion;
- section 8 – internal missile;
- section 9 – dropped loads.

2. ASSESSMENT STRATEGY

2.1 PRE-CONSTRUCTION SAFETY REPORT (PCSR)

21. ONR's GDA Guidance to Requesting Parties (thewww.onr.org.uk/new-reactors/ngn03.pdf) states that the information required for GDA may be in the form of a PCSR, and Technical Assessment Guide (TAG) 051 sets out the regulatory expectations for a PCSR (www.onr.org.uk/operational/tech_asst_guides/ns-tast-gd-051.pdf).

22. At the end of Step 4, ONR and the Environment Agency raised GDA Issue CC-02 (www.onr.org.uk/new-reactors/reports/step-four/westinghouse-gda-issues/gi-ap1000-cc-02.pdf) requiring that the RP submit a consolidated PCSR and associated references to provide the claims, arguments and evidence to substantiate the adequacy of the **AP1000** Design Reference Point (DRP).

23. A separate regulatory assessment report is provided to consider the adequacy of the PCSR and closure of GDA Issue CC-02.

2.2 STANDARDS AND CRITERIA

24. The standards and criteria adopted within this assessment are principally the Safety Assessment Principles (SAPs) (Ref. 2), internal TAGs (Refs 3 and 4), relevant national and international standards and relevant good practice informed from existing practices adopted on UK nuclear licensed sites.

2.3 SAFETY ASSESSMENT PRINCIPLES

25. The key SAPs applied within the assessment are included in Table 1.

Table 1 – SAPs used in the assessment

SAP ref.	Title
AV.1	Fault analysis: assurance of validity of data and models. Theoretical models.
AV.2	Fault analysis: assurance of validity of data and models. Calculation methods.
AV.3	Fault analysis: assurance of validity of data and models. Use of data.
AV.4	Fault analysis: assurance of validity of data and models. Computer models.
AV.5	Fault analysis: assurance of validity of data and models. Documentation.
AV.6	Fault analysis: assurance of validity of data and models. Sensitivity studies.
AV.7	Fault analysis: assurance of validity of data and models. Data collection.
AV.8	Fault analysis: assurance of validity of data and models. Update and review.
ECE.1	Engineering principles: civil engineering. Functional performance.

ECE.6	Engineering principles: civil engineering. Functional performance
ECS.2	Engineering principles: civil engineering. Loadings.
ECS.3	Engineering principles: safety classification and standards. Safety categorisation.
EDR.2	Engineering principles: design for reliability. Redundancy, diversity and segregation.
EDR.4	Engineering principles: design for reliability. Single failure criterion.
EHA.1	Engineering principles: external and internal hazards. Identification and characterisation
EHA.3	Engineering principles: external and internal hazards. Design Basis Events.
EHA.6	Engineering principles: external and internal hazards. Analysis.
EHA.7	Engineering principles: external and internal hazards. Cliff-edge effects.
EHA.12	Engineering principles: external and internal hazards. Flooding.
EHA.13	Engineering principles: external and internal hazards. Use, storage and generation of hazardous materials.
EHA.14	Engineering principles: external and internal hazards. Fire, explosion, missiles, toxic, gases etc – sources of harm.
EHA.15	Engineering principles: external and internal hazards. Hazards due to water.
EHA.16	Engineering principles: external and internal hazards. Fire detection and fighting.
EHA. 19	Engineering principles: external and internal hazards. Screening.
EKP.1	Engineering principles: key principles. Inherent safety.
EKP.2	Engineering principles: key principles. Fault tolerance.
EKP.3	Engineering principles: key principles. Defence in depth.
EKP.4	Engineering principles: key principles. Safety function.
EKP.5	Engineering principles: key principles. Safety measures.
ELO. 4	Engineering principles: layout. Minimisation of the effects of incidents.
EMT.4	Engineering principles: maintenance, inspection and testing. Validity of equipment qualification.
EMT.5	Engineering principles: maintenance, inspection and testing. Procedures.
EQU.1	Engineering principles: equipment qualification. Qualification procedures.

ERL. 3	Engineering principles: reliability claims. Engineered safety measures.
ERL. 4	Engineering principles: reliability claims. Margins of conservatism.
ESS.1	Engineering principles: safety systems. Provision of safety systems.
FA.3	Fault analysis: general. Fault sequences.
FA.7	Fault analysis: design basis analysis. Consequences.
FA.8	Fault analysis: design basis analysis. Linking of initiating faults, fault sequences and safety measures.
EHF.1	Engineering principles: human factors. Integration within design, assessment and management.
EHF.7	Engineering principles: human factors. User interfaces.
NT.2	Numerical targets and legal limits. Time at risk.
SC.4	The regulatory assessment of safety cases. Safety case characteristics.

2.4 TECHNICAL ASSESSMENT GUIDES

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

Table 2 – TAGs used in the assessment

TAG Ref.	Title
NS-TAST-GD-005	Guidance on the Demonstration of ALARP, Revision 7
NS-TAST-GD-014	Internal Hazards, Revision 4

2.5 NATIONAL AND INTERNATIONAL STANDARDS AND GUIDANCE

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

Table 3 – International guidance and standards used in the assessment

Organisation	Title
Western European Nuclear Regulators' Association (WENRA)	Reactor Harmonisation Working Group: Safety Reference Levels for Existing Reactors. September 2014 www.wenra.org/media/filer_public/2014/09/19/wenra_safety_reference_level_for_existing_reactors_september_2014.pdf
International Atomic Energy Authority (IAEA)	Safety Guide No. NS-G-1.7. Protection against Internal Fires and Explosions in the Design of Nuclear Power Plants.
International Atomic	Safety Guide No. NS-G-1.11. Protection against Internal

Energy Authority (IAEA)	Hazards other than Fire and Explosions in the Design of Nuclear Power Plants.
International Atomic Energy Authority (IAEA)	Safety Guide No. NS-G-2.1. Fire Safety in the Operation of Nuclear Power Plants.

2.6 USE OF TECHNICAL SUPPORT CONTRACTORS (TSC)

28. It is usual in GDA for ONR to use technical support, for example to provide additional capacity to optimise the assessment process, enable access to independent advice and experience, analysis techniques and models, and to enable ONR‘s inspectors to focus on regulatory decision making.

29. A TSC was contracted to:

- independently assess the methodologies for each internal hazards GDA issue;
- independently assess the Topic Reports and support information;
- capture TSC technical comments from their reviews in a number of Regulatory Queries.

30. Gesellschaft für Anlagen und Reaktorsicherheit (GRS) was selected. The TSC provided a team of experts with significant relevant experience in the assessment of safety cases and internal hazards analysis. The TSC is experienced in supporting ONR on other contracts within the GDA process.

31. Using a TSC allowed ONR’s internal hazards resources to focus on significant issues such as analysis methodologies on pressure part failure and overall convergence between UK and US regulatory expectations.

32. The assessment report (Ref. 17) produced by the TSC was used to inform my regulatory judgements.

2.7 INTEGRATION WITH OTHER ASSESSMENT TOPICS

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

- **Civil engineering**: **Substantiation of barriers against internal hazards.** There were significant updates in several internal hazard areas. Barriers were reviewed to confirm whether the updates in internal hazards will affect the civil engineering design.
- **Control and instrumentation: Internal hazards affecting C&I Rooms**. Interactions centred around the room containing the DAS cabinets and the Main Control Room (MCR).
- **Fault Studies: Fault schedule and links to the hazard schedules. Internal hazards were not adequately represented on the fault schedule at the end of** GDA Step 4. Interactions were held to ensure the safety case was in alignment in between fault studies and internal hazards; and that suitable information was reflected in both schedules.
- **Human Factors: Protection via administrative procedures**. Where operator actions were argued to protect from or mitigate a fault sequence, the suitability was discussed with human factors.
- **Mechanical engineering: Review on squib valves.** The internal fire safety case in the Internal Hazards area did not appear to align with mechanical

engineering. Cross-cutting interactions were held to ensure hazard identification and consequence analysis was adequate.

- **Probabilistic Safety Analysis (PSA): Internal fire.** The PSA analysis provided information in the consideration of the internal fire safety case.
- **Structural integrity: Categorisation and classification, and pressure part failure.** The pressure part failure safety case in the Internal Hazards area did not align with structural integrity. Partial failure was used in the analysis which was not in line with my expectations. Cross-cutting interactions were held to ensure alignment.

2.8 OUT OF SCOPE ITEMS

34. The following items have been agreed with the RP as being outside the scope of this GDA and remain outside the scope of this assessment:

- Resolution of any assessment findings (AFs) identified within either the Step 4 GDA reports, or identified within the assessment reports produced to support closure of GDA issues. Suitable closure of AFs shall be the responsibility of a licensee and assessment of these will be undertaken post GDA in site-specific activities by ONR.
- Site-specific elements of the **AP1000** design. These will be assessed by ONR as part of any future site-specific activities.
- Postulated internal hazards resulting in non-nuclear safety consequences / conventional safety consequences impacting on persons either within the site or outside of the site boundary. These shall be the responsibility of a licensee and assessment of these may be undertaken post GDA in site-specific activities by ONR.
- Aspects in regards to the scope of environmental impacts. Where necessary, I will discuss any concerns directly with the Environment Agency Inspectors to take forward.

3. REQUESTING PARTY'S SAFETY CASE

35. The sections below are provided for background purposes to aid the reader of this assessment report.

36. The **AP1000** plant consists of the following five principal structures. Each of these buildings is constructed on an individual basemat:

- Nuclear Island;
- Turbine Building;
- Annex Building;
- Diesel Generator Building;
- Radwaste Building.

37. The structures that make up the Nuclear Island (NI) are:

- Containment Building;
- Shield Building;
- Auxiliary Building.

3.1 THE AP1000 PLANT'S KEY SAFETY SYSTEMS

38. Chapter 5 of PCSR (Refs. 65 and 188) describes the RP's approach to safety categorisation and classification. It has adopted a three-tier approach, categorising safety functions A to C based on their importance to nuclear safety and classified SSCs 1 to 3 based on their prominence in delivering the safety functions.

- Category A safety function is a principal means of maintaining nuclear safety. Category A safety functions are those utilised to achieve and maintain the reactor in a non-hazardous, stable state for at least 72-hour following an initiating event. Failure to maintain a Category A safety function has the potential to result in significant core damage, radiation exposure >20mSv to onsite personnel, or radiation exposure >1mSv to the offsite population. Crucially, it also defines a Category A safety function as "*Protecting SSCs from internal hazards that would directly and inevitably result in loss of a principal means of fulfilling a Category A safety function*."

- Category B safety function is a significant contributor to nuclear safety. Category B safety functions are utilised to do the following:

    - maintain the non-hazardous stable state after 72-hour following an accident;
    - prevent radiological exposure to onsite personnel and the offsite population from exceeding the design basis limits;
    - mitigate beyond Design Basis Accident (DBA), and crucially;
    - protecting against internal hazards that could, as part of a sequence of failures, result in the loss of one of the Category B safety functions, such as preventing the spread of fire.

- Category C safety functions are those safety functions that may make a contribution to nuclear safety, but are not categorised as Category A or Category B. Since the removal of nuclear heat during normal operation prevents reactor trips and the actuation of Category A and Category B functions, these normally operating duty systems are recognised as being important to safety.

39. The design philosophy of the **AP1000** plant is to use passive Class 1 means of delivering the following safety functions without a requirement for alternating current (ac) power:

- shutting down the nuclear reaction;
- removing decay heat, which uses only natural mechanisms such as natural circulation, conduction, convection, evaporation, and condensation;
- maintaining the reactor coolant water inventory;
- Containment isolation.

40. All Class 1 systems are located in the NI.

41. The **AP1000** reactor does have active SSCs, similar to those utilised on “traditional” Pressurized Water Reactors (PWR) designs, which do make a contribution to safety. However, they are generally only claimed in the design basis safety case after 72-hour to maintain the non-hazardous stable state following an accident, or as a Defence-in-Depth (DiD) backup to the main Class 1 SSCs. As a result, they are classified as Class 2.

42. The RP’s fault schedule in the PCSR set to demonstrate that there is at least one Class 1 means of delivering Category A functions for the first 72-hour following a design basis fault. For frequent faults ($<1x10^{-3}$ per year), it demonstrated that there is also a second means of delivering Category A safety functions, also mainly through Class 1 SSCs. However, there are two notable exceptions to this approach:

- many frequent faults place a claim on the Class 2 DAS as a diverse means of actuating the Class 1 SSCs;
- for small break Loss of Coolant Accident (LOCA) faults, a claim is placed on the RNS and its support systems, to provide long term low pressure safety injection.

3.2 THE AP1000 INTERNAL HAZARDS SAFETY CASE

43. Internal hazards are those hazards to plant and structures that originate within the site boundary, but that are external to the nuclear or active systems. In Chapter 11 of the PCSR, the RP considered the following hazards (Ref. 69):

- internal fire;
- internal flood;
- pressure part failure;
- internal explosion;
- internal missiles;
- release of toxic, corrosive, or flammable material;
- dropped loads and load mishandling;
- biological agents;
- onsite transport;
- electromagnetic interference;
- combinations of hazards.

44. Release of toxic, corrosive, or flammable material, biological agents, onsite transport and electromagnetic interference were addressed at high level during Step 4 of GDA. No Regulatory Issues have been raised.

45. The overarching high level safety claim addressing internal hazard challenges within the **AP1000** design basis is defined in the PCSR as:

Claim IH-0: An internal hazard within the design basis does not prevent delivery of the Category A safety functions and supporting post 72-hour Category B safety functions necessary to respond to the postulated event.

46. Infrequent hazards are the most severe accidents within the normal design basis of the plant that could occur on site, taking such things as fire loading or water inventories into account. An SSC not designed or evaluated to survive a Design Basis hazard is assumed to fail and it is therefore a requirement of the safety case to demonstrate that there are sufficient remaining SSCs to provide all Category A and supporting post 72-hour Category B safety functions safety functions.

47. Despite this claimed resilience to the consequences of hazard, the RP identified the following sub-claims which its safety case sets out to demonstrate:

- Prevention of the internal hazard fault. Where claims of this nature are made, minimisation and elimination of hazards so far as is reasonably practicable from the **AP1000** design have been used in the prevention of hazard initiation.
- Protection from the internal hazard fault. Where claims of this nature are made, protective measures have been incorporated into the design to safeguard the delivery of Category A safety functions from the effects of an internal hazard.
- Mitigation of the internal hazard fault. Where claims of this nature are made, mitigation of the resulting faulted conditions occurs through crediting the SSC design, selection of materials, limiting inventories, or use of redundant divisions of Class 1 SSCs.

48. In Chapter 11 of the PCSR (Ref. 69) and associated Topic Report, the RP detailed how it the **AP1000** design delivers the capability to support these claims. Specific arguments are made for each of the considered hazards.

49. These hazard specific safety case arguments are described in Sections 4 to 9 respectively, ahead of my assessment.

4. ONR ASSESSMENT OF GDA ISSUE GI-AP1000-IH-01 – INTERNAL FIRE

4.1 RESOLUTION PLAN ACTIONS AND DELIVERABLES

50. During GDA Step 4, a GDA issue was raised relevant to internal fire safety case substantiation of the fire barriers and fire damper provisions (GI-AP1000-IH-01) (ONR's Step 4 Internal Hazards Assessment Report GI-AP1000-IH-01 (Ref. 13). This GDA issue was comprised of the following actions:

- GI-AP1000-IH-01.A1: Provide substantiation of the nuclear significant hazard barriers claimed to provide the level of fire resistance stated within the PCSR for integrity, insulation and load bearing capacity (where applicable);
- GI-AP1000-IH-01.A2: Provide the substantiation of the approach taken to the design and installation of fire dampers claimed within the **AP1000** PCSR.

51. The RP recognised that in order to address the two actions relevant to the fire substantiation, there was a need to re-present the claims and arguments in the PCSR. The RP proactively decided to submit a Topic Report in this area capturing the claims arguments and evidence. The outcome of the Topic Report is captured in the revised PCSR.

52. The RP's Resolution Plan (Ref. 18) identifies specific deliverables associated with the above actions:

- Internal Fire Road Map;
- Finite Element Analysis Calculation;
- Fire Barrier Similarity Report;
- Internal Fire Topic Report;
- Fire Damper Report;
- Combined Consequential Hazards.

53. The Internal Fire Road Map (Ref. 173) was initially issued to show the link between the existing claims, arguments and evidence. The document has been superseded and captured in the Fire Protection Topic Report (Ref. 82).

54. In addition to the list above, the RP issued a number of draft documents, multiple revisions of the Topic Report and PCSR and a number of supporting documents.

55. During this phase of the GDA, I raised regulatory queries RQ-AP1000-1301, RQ-AP1000-1340 and RQ-AP1000-1529 (Refs. 24, 31 and 44) aiming to seek clarity on the scope of the submissions and the timescales given in the Resolution Plan.

56. In the following sub-sections, I will cover the following:

- the RP's safety case on internal fire;
- my assessment of GI-AP1000-IH-01.A1, which includes:
    - assessment of the claims and arguments relevant to fire barriers, which also includes assessment of combined consequential hazards;
    - assessment of the fire analysis undertaken to support the substantiation of the fire barriers;
    - assessment of substantiation of the concrete fire barriers and steel-concrete-steel composite structures.
- my assessment of GI-AP1000-IH-01.A2;
- my conclusions and assessment findings.

4.2 REQUESTING PARTY'S SAFETY CASE ON INTERNAL FIRE

57. The RP's top level generic PCSR is a summary document (Ref. 188). This includes summarising all the systems that underpin the requirements of the safety case. Internal hazards are described in Chapter 11 (Ref. 69), and are addressed from a deterministic approach. Underpinning the PCSR are the Topic Reports on the internal hazards GDA issues. There is also a suite of supporting documents, discussed in my assessment. The assessed draft Chapter 11 of the PCSR (Ref. 69) was subsequently included in the final submission of the full PCSR (Ref. 188).

58. Key document submissions for internal fire are:

- UKP-GW-GL-793, Revision 0D – **AP1000** Pre-Construction Safety Report – Chapter 11 Internal Hazards (Ref. 69);
- UKP-GW-GLR-111, Revision 1 – UK **AP1000** Internal Hazards Topic Report – Fire Protections, (Ref. 82).

59. The **AP1000** reactor internal fire safety case uses the following approaches to ensure that the Class 1 SSCs will continue to provide their Category A safety functions following the worst case postulated internal fire (Pre-Construction Safety Report – Chapter 11 and Topic Report - Fire Protection Refs 69 and 82):

- minimise combustible loads and ensure that no significant combustible loads exist that could cause failure of Class 1 SSCs;
- ensure that a fire is prevented from spreading between redundant trains of equipment using appropriate combinations of physical separation and Class 1 nuclear fire barriers;
- provide sufficient redundancy in the design such that if one train of protection fails as a result of fire, coincidental with an unrelated single active failure elsewhere, the safety functions continue to be provided by the equipment that remains unaffected;
- provide a ventilation system that minimises the spread of fire between fire compartments and the damage to electrical systems from smoke;
- maintain the habitability of control room areas.

4.2.1 AP1000 INTERNAL FIRE CLAIMS

60. The RP made the following claims in the area of internal fire (Pre-Construction Safety Report – Chapter 11 and Topic Report - Fire Protection Refs 69 and 82):

61. The overarching high level safety claim addressing internal hazard challenges within the **AP1000** design basis is summarised as:

- **Claim IH-1.0**: Postulated fire events within the design basis do not prevent the delivery of the Category A safety functions and the supporting post 72- hour Category B safety functions necessary to respond to postulated events.

62. Note that the numbering terminology between the PCSR and Topic Report differs. For clarity in this assessment report, the numbering from the PCSR has been used. This avoids confusion with the numbering used in the other internal hazards GDA issues.

63. Specific internal fire claims are:

- **Claim IH-1.1**: The internal fire hazard has been minimised so far as is reasonably practicable.

- **Claim IH-1.2**: Class 1 SSCs will be protected from the direct or indirect effects of a fire by isolating the source of the fire.
- **Sub-claim IH-1.2.1**: A postulated fire outside Containment will not propagate beyond the fire area of inception; therefore, redundant Class 1 SSCs will not be disabled by a postulated fire event.
- **Sub-claim IH-1.2.2**: A postulated fire within Containment will not propagate to the extent that it damages redundant safe shutdown components.
- **Sub-claim IH-1.2.3**: Penetrations through barriers do not degrade the fire withstand capability of the barrier itself.

64. The **AP1000** design uses two principles to inhibit the spread of fire during a Design Basis Event:

- Separation of SSCs inside Containment. Within the Containment, fire spread is prevented by providing adequate separation of equipment by distance or height (particularly for redundant Class 1 SSCs), and by the use of non-combustible structural barriers acting as passive fire-protection features to partially segregate redundant SSCs. The passive fire-protection features include steel, steel composite, and steel-concrete-steel (SC) composite modular structures. No formal claims on the SC composite modules, inside Containment have been made; however, these structures have been substantiated against three hours fire resistance.
- Segregation of SSCs outside Containment. In the NI areas outside Containment, the plant is segregated into fire compartments by fire resistant reinforced concrete, steel composite (for example, Durasteel), and steel-composite-steel fire barriers. Fire compartment barriers consist of three-hour fire rated barriers along with two-hour fire rated barriers provided for selected stairwell and elevator shafts. In addition, SC composite structures of three hours' fire resistance are used in the Auxiliary Building (and also in Containment and in the cylindrical wall of the Shield Building). The reinforced concrete, the steel composite fire barriers and the SC modules (including the CA modules within the Auxiliary Building and the in-Containment structures) could withstand the total burn of all combustible material within the fire area. These structures are Class 1 SSCs. The CA modules are of the same SC design.

4.3 ASSESSMENT OF GDA ISSUE GI-AP1000-IH-01

4.3.1 SCOPE OF THE ASSESSMENT

65. The assessment strategy in Section 2 was used to formulate the scope of my assessment.

66. My assessment covers the deliverables used in addressing the two actions (GI-AP1000-IH-01.A1 and GI-AP1000-IH-01.A2). I assessed the Topic Report and the PCSR on internal fire. I also sampled supporting documents to obtain confidence on the requisite evidence and substantiation of the claims made.

67. The areas chosen to review the internal fire safety case were limited to:

- inside Containment;
- outside Containment – mainly focused on the Auxiliary Building. Some areas of high combustible inventory such as the Turbine Building were also assessed.

68. The sections below cover the areas of my assessment.

4.3.2 ASSESSMENT OF CLAIMS AND ARGUMENTS

69. I assessed the RP's claims and arguments, relevant to these actions, and I observed the following:

Inside Containment

70. Initially, the RP made implicit claims (in an earlier revision of the Topic Report - Fire Protection, Ref. 81) on the SC composite structures for a number of rooms inside Containment including 11300, 11400, 11500, 11504 and 11306.

71. I challenged the RP on the implicit claims made on the SC structures as this claim was not supported by the arguments presented. This was not in line with ONRs SAPs SC.4.

72. The RP in response undertook a review of the fire safety case inside Containment and submitted a "Review of Claims for Internal Fires within Containment" (Ref. 111). This review resulted in a revised claim inside Containment as given above (Sub-claim IH-1.2.2).

73. The revised claim inside Containment is based on a combination of low combustible inventory, distance or height between redundant SSCs and by partial physical structures that inhibit fire propagation between fire zones. The RP argued that surface ignition is precluded by the presence of walls, ceilings and floors. No formal claim on civil structures inside Containment has been made.

74. I further challenged the RP as to why no claims have been made on the walls, ceilings and floors (SC modules) inside Containments given that a number of these structures have been substantiated against more onerous fires.

75. The RP argued that the function of the barriers inside Containment is to reduce the potential for the flames from a fire in one fire zone to directly impinge on the redundant SSCs in an adjacent fire zone, and to provide a physical separation distance between combustible materials in adjacent fire zones. This argument is supported by the low combustible loading inside Containment.

76. The RP also argued that complete segregation cannot be achieved due to the need to maintain the free exchange of gases and liquids for passive containment cooling.

77. The RP also provided some qualitative discussions on fire spread through flame impingement, thermal radiation or due to smoke exposure.

78. The RP proactively identified eight areas where redundant divisions of Class 1 SSCs are located in adjacent fire zones. The RP argued that a combination of low combustible inventory, passive separation between fire zones (by either distance, by a 30 cm wall, or by SC composite structures, and minimum penetrations) prohibits the spread of fires between adjacent fire zones.

79. The RP recognised that inside Containment cable design including insulation, routing and management has a significant influence on fire safety and could affect the internal fire safety claims inside Containment. However, these aspects were identified as assessment findings during Step 4 of the GDA (AF-AP1000-IH-02, AF-AP1000-IH-03 and AF-AP1000-IH-04) to be addressed during the licensing stage (ONR's Step 4 Internal Hazards Assessment Report, Ref. 13). I concur with the approach taken.

80. Overall, the arguments presented inside Containment are reasonable.

81. I sampled the UK **AP1000** Barrier Matrix document (Ref. 104) and noticed that this document identifies barriers inside Containment. Therefore the documents are not consistent and I raised CP-AF-AP1000-IH-01.

CP-AF-AP1000-IH-01 – The licensee shall update the Barriers Matrix document to clearly identify all claimed Class 1 and 2 Barriers (walls, floors and ceilings) for each room, and align it with all internal hazards Topic Reports. The document shall clearly state the internal hazards imposed loads as well as the design loads that the barriers are designed to withstand.

Outside Containment

82. The SSCs delivering the safety claim (claim IH-1.2.1) are the Class 1 three-hour reinforced concrete fire barriers and the steel-concrete-steel SC composite structures in the Auxiliary Building. The three-hour fire barriers are claimed to provide the segregation between redundant Class 1 SSCs. These are listed in the hazard schedule of the Topic Report.

83. The claim on the three-hour reinforced concrete fire barriers and on the steel-concrete-steel SC modules is in line with my expectations and ONR's SAPs EKP.5, ECS.2 and EHA.16. It is also in line with international guidance (Refs 5 and 6).

84. Early submissions of References 68 and 81 listed a number of areas where one and two-hour fire barriers have been claimed.

85. I queried with the RP the claims made on one and two-hour fire barriers and in particular in Rooms 12105, 12101, 12205, 12204, 12202 and 12201 (Ref. 112). In response to my query, the RP undertook a further review and identified a number of two-hour fire barriers that need to be claimed. These surround the stairs and elevator shafts in the Auxiliary Building. These do not separate redundant Class1 equipment. The RP argued that as the fire loading for stairs and shafts is limited, the two-hour fire barriers should be sufficient. These will be constructed by fire-rated steel composite assemblies.

86. During my assessment, I challenged the RP on the suitability and sufficiency of Class 1 fire barriers which incorporate a door within them (RQ-AP1000-1797 Ref. 58). These doors are Class 1 three-hour fire-rated doors.

87. The RP explained that outside Containment, and with the exception of five cases, the **AP1000** design incorporates two doors between fire areas containing redundant Class 1 SSCs. The five locations are:

- the division 'A' and/or division 'C' cables are separated from fire areas with division 'B' and/or division 'D' cables with a single fire door. This is the case with corridor 12300 in fire area 1230 AF 01 which is separated from Rooms 12303, 12304 and 12305 in different fire areas with a single fire door.
- the reactor trip switchgear rooms (12422 and 12423) have all four divisions and are separated from Room 12421 in separate three-hour fire areas with a single fire door.

88. For these five cases cited in the above paragraph the doors are fitted with Portal Access Controllers which monitor the doors. A prolonged opening of the door will sound an alarm to MCR. The Portal Access Controllers and alarm are Class 2. The RP identified the doors separating potentially redundant SSCs with a single fire door, which are fitted with the Portal Access Controllers in the PCSR (Ref. 69).

89. The approach taken on the Class 1 doors is in line with my expectations and relevant good practice established in the UK. However, this aspect should be considered during detailed design, to ensure that the criteria are supported and that relevant good practice is followed.

90. The RP's detailed analysis identified and listed all relevant barriers in the hazard schedule of the Topic Report. The hazard schedule also listed the redundant SSCs delivering the Category A safety function. This is in line with my expectations.

91. The UK **AP1000** Barrier Matrix document (Ref. 104) also graphically illustrated the barriers claimed in the Topic Report. The document, however, is not aligned with the Topic Report as it firstly refers to one-hour fire barriers, and secondly it does not differentiate between the various resistance ratings (see CP-AF-AP1000-IH-01).

92. I also sampled the safety case arguments on the Diverse Actuation System (DAS) processing cabinets and Squib Valve Controller Cabinet. The DAS is claimed in the UK safety case for diverse mitigation of multiple faults and therefore it is an essential Class 2 SSC. It provides a diverse backup to the Class 1 Protection and Safety Monitoring System (PMS) for selected functions.

93. All DAS cabinets are located on the radiological side of the Auxiliary Building within Room 12554. This room is separated from the Containment Personnel Access area by a closed door and is open by stairwell to the 24-hour continuously-manned Security Centre at 110.7 m (135' 3") elevation.

94. The co-location of all DAS cabinets is not in line with my expectations and SAPs EDR.2. I raised regulatory query RQ-AP1000-1431 and RQ-AP1000-1641 to gain clarity on the internal fire safety case relevant to the DAS system, including redundancy and spurious operations (Refs 35 and 51).

95. The RP's internal fire analysis conservatively assumed that a fire will destroy all cabinets. However, a fire in Room 12554 cannot disable all 4 PMS trains and the DAS simultaneously. The DAS and the PMS are independent and separate. The PMS cabinets are located in the elevation 100 m (100' 0") divisions 'A', 'B', 'C' and 'D' C&I Rooms on the non-radiological side of the Auxiliary Building. In addition, Room 12554 has a smoke detection and alarm system.

96. The RP also undertook an As Low As Reasonably Practicable (ALARP) study related to the internal hazards impacts on the DAS and concluded that the current DAS cabinet location represents an ALARP design and that changes to this design would not provide significant safety benefit (Ref. 35).

97. The DAS cabinets have also been considered by an ONR specialist PSA assessor in his assessment report for the PSA for the UK **AP1000** reactor (Reference 125).

98. From the internal hazards point of view, I am therefore satisfied with the arguments presented on the location of the DAS panels.

99. I am also satisfied with the identification of the redundant Class 1 SSCs and the link between initiating faults, consequences and safety measures presented (ONR's SAPs EDR.2, ELO.4 and FA.8).

100. The RP also made a claim on the penetrations through barriers that do not degrade the fire withstand capability of the barrier itself – sub-claim IH-1.2.3.

101. Penetrations in fire barriers including ventilation ductwork, cables and pipework are fire-stopped to the same fire resistance as the barrier penetrated (three hours' fire resistance) to reduce the potential for the spread of fire. The penetrations will be qualified to meet the requirements of BS EN 1366 Part 3 (Internal Hazards Topic Report - Fire Protection, Ref. 82).

102. The RP also developed penetration seal schedules, which provide details of all penetrations passing through walls and floors.

103. The RP in Reference 82 summarised its design criteria, for minimising penetrations. These were developed for the AP600, predecessor to the **AP1000** reactor, and the design philosophy was carried forward into the **AP1000** design.

104. I am broadly satisfied with the information presented on the design of penetrations against internal fires.

105. Fire dampers are considered further below in this assessment report.

106. Overall, I am satisfied that during the GDA the RP provided sufficient information on the claims and arguments presented in the Topic Report.

4.3.3 COMBINED CONSEQUENTIAL HAZARDS

107. Combined events and their associated combined consequential loads have the potential to compromise the safety measures in place against internal fire, in this instance the fire barriers. Because such events could affect the barrier substantiation it was necessary for me to assess this aspect in closing out this GDA issue.

108. The RP proactively undertook a study to identify all credible consequential, correlated or independent hazards relevant to internal fire and captured them in a hazard schedule combinations of hazards (Internal hazards topic report – combined hazards, Ref. 71).

109. I assessed this document and I noted the following:

- The RP in general considered that the combined consequences of internal fires, missiles and flooding do not result in the loss of all divisions of Class 1 SSCs delivering Category A safety functions. It identified a case where a fire in 1250 AF 01 could consequentially result in flooding in Room 12501. The flooding sources are from the Demineralised Water Transfer and Storage System (DWS), Fire Protection System (FPS) and Central Chilled Water System (VWS). During plant modes 5 and 6; the FPS is aligned to the yard tanks and as such represents a significantly larger flooding source such that all four divisions of Class 1 DC and Uninterruptible Power Supply (IDS) batteries may be lost if the flooding is not isolated. The RP proactively captured resolution to this issue in 'Design Reference Point document' (DRP) (Ref. 64) via 'UK AP1000 Internal Hazards Design Change Proposal' APP-GW-GEE-5401 'Limitation of FPS Supply to Non-RCA Auxiliary Building'.
- The RP considered consequential internal explosion, but argued that a release of flammable material is expected to burn rather than generate an explosion. In the Battery Rooms, the loss of heating, ventilation and air conditioning (HVAC) being detected by low exhaust flow sensors and the presence of hydrogen detectors would preclude an explosion. Some evidence underpinning this was provided in the 'Internal Hazards Topic Report – Explosion' (Ref. 77) and in a subsequent new document, '*Unmitigated explosion hazard analysis for **AP1000** Division 'B' Battery Room 1 (Room 12104)'* (Ref. 110). However, further

evidence is required to fully substantiate these arguments. This is discussed further in section 7.4.2.1.

- The RP argued that the potential for 'arc flash' which could lead to a High Energy Arc Fault (HEAF) event, as a result of flooding, is extremely unlikely. However, from operating experience, HEAF induced fires are not negligible (Refs. 114, 115, 116 and 170). Therefore, further consideration of these events is required.
- With regard to independent hazards (two or more completely independent internal hazards occurring at the same time), the RP identified fire as being a frequent event, but qualitatively dismissed any consideration of independent events from further analysis. This was based on consideration given within PSA in terms of core damage frequency being below the Basic Safety Level (BSL). This is not in line with my expectations and the RP should consider, relevant good practice and operating experience worldwide (for example the topical reports to the OECD committee on the safety of nuclear installations from the fire incidents records exchange project, Ref. 117), which indicate that delayed consequential hazards can occur.
- With regard to the external hazards inducing internal fires the RP considered a number of events including seismic and aircraft impact, but the overall high level qualitative discussion does not provide the requisite evidence.
- Fire initiated by a seismic event has been identified to have the potential to give an 'arc flash' fire if cable voltage exceeds 480V. However, no justification has been provided for the basis for selection of the 480V threshold. In addition no events or rooms have been identified.
- Turbine disintegration inducing fire and flood has not been considered in this report. However, the NI exterior structure was demonstrated to remain unaffected by the turbine missile event.
- The RP concluded that the Category A safety functions will continue to be delivered following various design basis combined hazards.

110. Based on the above, there is a need to further consider and provide appropriate justification for all consequential, correlated or independent hazards post GDA and during detailed design, where consideration of site-specific requirements would be made available. This will satisfy ONR SAP EHA.1 and EHA.19.

CP-AF-AP1000-IH-02 – The licensee shall use site-specific information to:

- **Complete the screening assessment of hazard combinations and provide justification for those screened out.**
- **Fully analyse all credible external and internal hazards combinations.**
- **Justify the adequacy of the barriers.**

111. This finding is applicable to all internal hazards assessed in this report.

4.4 ASSESSMENT OF GI-AP1000-IH-01.A1 – INTERNAL FIRE SAFETY CASE SUBSTANTIATION

4.4.1 FIRE ANALYSIS METHODOLOGY AND RESULTS

112. The RP undertook a room-by-room deterministic fire analysis for all relevant buildings (Refs. 82 and 118). The fire analysis included identification of combustible loads, calculations of fire severity identification of Class 1 SSCs, identification of segregation and redundant SSCs, control of fire and smoke spread via the ventilation system, and identification of defence in depth measures.

113. The fire analysis approach is in line with my expectations and ONR's SAPs EHA.1, EHA.6, EHA.14 and, EHA.16. It is also in line with IAEA guidance (Safety Guides. NS-G-1.7 and NS-G-2.1; Refs. 6 and 8, respectively.

114. The fire severity analysis is based on estimates of the time-temperature profiles to which the walls and ceilings could be exposed and then compared to the ASTM E119 standard exposure profile, which is used in North America for qualifying the fire resistance rating of the fire barriers.

115. In principle, this approach is reasonable. However, the fire analysis undertaken was not in line with my expectations and relevant good practice and so I raised regulatory queries RQ-AP1000-1429, RQ-AP1000-1515 and RQ-AP1000-1591(Refs. 33, 42 and 48). These regulatory queries cover the following aspects of the fire analysis:

- the methodology given in National Fire Protection Association (NFPA) Handbook 16th edition 1986 is outdated and not in line with relevant good practice;
- the fire equivalent approach does not directly address a “realistic fire” which may lead to conditions that could exceed the predictions, which support the ASTM E119 curve;
- the chosen categories of fire severity curves used in the analysis were not conservative;
- ventilation effects have not been considered;
- protected loads such as cables contained within enclosed conduits or heavy sheet metal cabinets have been excluded from the analysis, and therefore the results were not conservative;
- impact of the localised fire effects on barriers.

116. In response to the above regulatory queries, the RP undertook a revised analysis for selected rooms, using the fire equivalent approach described in EN 1991-1-2:2002 (Ref. 119).

117. The results of the revised analysis are presented in References 120 and 121.

118. The RP concluded that while the revised methodology resulted in longer equivalent fire durations, than those initially determined, in all instances, within the NI, the equivalent fire duration remains within the prescribed fire duration rating of the Class 1 fire barriers with significant margins.

119. A quantitative uncertainty analysis was also performed, which indicated that there is a very high probability (99.7%) that the barriers will confine the fire to the fire area of inception.

120. The RP in response to RQ-AP1000-1515 confirmed that the cable trays will have a cable fill of 40% and this figure has been used in the analysis whether the tray is empty or filled to its maximum allowed per design. The AP1000 Fire Protection Design Criteria and Guidelines has been revised to reflect the maximum 40 % fill for all cable trays. The 40% fill requirement was also reflected in the maximum allowed fill in the electrical design criteria.

121. In addition, and in order to demonstrate that the use of equivalent fire durations is conservative and bounding, the RP undertook realistic fire scenarios modelling for three plant areas using diverse fire modelling techniques (Fire modelling report for selected rooms, Ref. 122). The three areas modelled were the following:

- the RNS Pump Room ‘A’, using algebraic calculations;
- the I&C Division ‘C’ Room, using the Consolidated Fire and Smoke Transport (CFAST) software (zone fire model);
- the Steam Generator Compartment 1, using the Fire Dynamic Simulator (FDS) (a Computational Fluid Dynamics (CFD) model).

122. The RP concluded that the resulting temperature profiles do not exceed the standard time temperature, which the barriers are qualified at (ASTM E119).

123. I subjected the analysis undertaken in References 120, 121 and 122 to a detailed assessment and I satisfied myself that the results overall are reasonable.

124. The RP proactively identified five areas (two in the Turbine Building, one in the Annex Building and two in the Diesel Generator Room) where the three-hour credited fire barriers are insufficient against fires (Fire modelling report for selected rooms, Ref. 122).

125. I raised regulatory query RQ-AP1000-1641 (Ref. 51) to seek clarity on the implication of the above five fire scenarios. The RP argued that these barriers are not required to segregate safe shutdown equipment and additionally fire detection and suppression capability has been provided. Furthermore, in the Annex Building and in the Diesel Generator building there are no Class 1 safe shutdown components.

126. With regard to the Turbine Building, no Class 1 safe shutdown SSCs are present; however, a number of SSCs are located in the building that provide redundancy.

127. The RP also identified that due to the change in turbine frequency from 60Hz to 50Hz, to be compatible with the UK national grid, the proposed turbine will need to change. This may result in consequential changes to the size and layout of the Turbine Building. Therefore, the current fire assessment and definition of the type and location of fire, smoke and combination fire/smoke dampers in the Turbine Building will need to be revisited post GDA.

128. The RP is confident that based on the design principles currently adopted, the final design of the Turbine Building, when completed, will show that a fire in the Turbine Building does not prevent delivery of the Category A safety functions from the Class 1 SSCs located in the NI.

129. Given the severity of the fires presented in these five locations, it will be prudent to minimise the combustible inventory in these areas in line with claim IH-1.1 and in line with ONR's SAPs EHA.13, and to provide a robust justification.

CP-AF-AP1000-IH-03 – The licensee shall:

- **Develop controls and procedures to minimise the combustible inventory held in the Turbine Building, the Annex Building and in the Diesel Generator Room and provide adequate fire resistance barriers where required.**
- **Consider the impact of the turbine selected on the design of the Turbine Building, including fire barriers and penetrations.**

130. Despite the need to review the effects of localised fires, this is detail matter that in my judgement is unlikely to lead to need for significant change to civil layout.

131. With regard to localised effects and in response to regulatory query RQ-AP1000-1429 (Ref. 33) the RP undertook a qualitative assessment on localised fire effect (Fire modelling report for selected rooms, Ref. 122). The RP identified areas where localised fire may be present.

132. The RP concluded that no localised effects could impact on the safe shutdown function of the **AP1000** design.

133. I subjected Reference 122 to an assessment and concluded that the RP's qualitative assessment doesn't provide the requisite justification to support the conclusion. There is a need, therefore, to undertake a quantitative analysis of the localised effects, post GDA when detailed design is available, to support the conclusion reached in the Topic Report.

CP-AF-AP1000-IH-04 – The licensee shall use site-specific information to undertake a quantitative analysis of localised fire effects on fire barriers.

134. Overall, the fire analysis undertaken is in line with my expectations and ONR's SAPs FA.7, AV.2 and AV.6.

135. I am largely satisfied that during the GDA the internal fires have been subjected to detailed quantitative consequence analysis. With the exception of localised effects the RP demonstrated that the fire barriers resistance provisions are sufficient. Localised effects shall be considered post GDA, when detailed design is available.

4.4.2 SUBSTANTIATION OF REINFORCED CONCRETE FIRE BARRIERS

136. References 82 and 123 determined the reinforced concrete walls and floors which are required to be fire resistant.

137. The design of **AP1000** NI concrete is based on ACI 349 and ACI 318. ACI-318 provides direction for fire protection guidance to ACI-216.1, which follows the ASTM E119 (Refs 82 and 123).

138. I raised regulatory query RQ-AP1000-1429 (Ref. 33) to gain clarity and confidence on the approach taken to fire barriers substantiation and on the scope of the work undertaken.

139. The approach to fire barriers substantiation was discussed in **AP1000** Plant fire resistance rating for nuclear island reinforced concrete structure (Ref. 123). This document states the minimum thickness and cover required to satisfy ACI 216.1, for a given fire resistance requirement, and then compares that to the requirements of BS EN1992-1-2. In this context:

- concrete barrier thickness is the face to face thickness of the structure;
- concrete cover relates to the distance from the face to the reinforcing steel (rebar).

140. The plant fire resistance rating document (Ref. 123) concludes that the minimum thickness of concrete required by ACI216.1 and by BS EN 1992-1-2 (157mm and 210mm respectively) is exceeded by the **AP1000** design for standard three-hour reinforced concrete structures. The minimum cover for three-hour fire resistance is given as 32 mm by ACI 216.1.

141. I sampled an example in drawing APP-1220-CR-931 for the section view through wall 7.3, which is the south wall of Room 12112 (fire zone 1212 AF 12112) and the three-hour fire area boundary for fire area 1210 AF 01 (captured in regulatory query RQ-AP1000-1797, Ref. 58). The cover dimensions on the drawings were shown as 38 mm from the exterior surface of the concrete wall to the outermost layer of the rebar.

142. The RP indicated that the ACI 216.1 rebar cover requirements are shown from face of cover to face of rebar which differs from BS EN 1992.1.2. The latter shows face of cover to centerline of rebar. The RP clarified that from APP-1220-CR-931 and the concrete reinforcement drawings the minimum concrete cover to

the centreline of the first reinforcing element is 43mm. This is sufficient to meet the requirements of BS EN 1992.1.2.

143. The RP also proactively identified the need for the fire modelling report for selected rooms (Ref. 122) to be updated to demonstrate compliance with the requirements of BS EN 1992-1-2 for load-bearing capacity and to report the actual concrete cover.

144. Overall the submission provides the requisite information relating to the substantiation of the reinforced concrete barriers claimed in the Topic Report.

145. The results presented when coupled with the overall conservatism in the analysis, gives me confidence that sufficient margins are available in the design of the reinforced concrete fire barriers.

146. I am, therefore, satisfied that the fire resistance of the three hours' reinforced concrete barriers has been subjected to detailed substantiation and has been demonstrated to be adequate.

4.4.3 SUBSTANTIATION OF THE STEEL-CONCRETE-STEEL (SC AND CA) COMPOSITE MODULES

147. The civil engineering structures, other than reinforced concrete, have been designed for rapid construction by making maximum use of offsite fabricated steel modules, which are subsequently filled with concrete when located onsite.

148. The SC composite modules are constructed from concrete and structural steelwork.

149. Substantiation of the SC structures were largely captured in the Resolution Plan (Ref. 124) for GI-AP1000-CE-01.A7 – Justification of the ability of SC to withstand fire. This action is concerned with the structural stability of all the CA modules following a potential fire. Substantiation of the SC, including the CA modules is therefore captured in ONR's assessment report for the closeout of civil engineering GDA issues (Ref. 126).

150. In addition, and in support of GI-AP1000-IH-01.A7, the RP undertook analysis and submitted the following documents:

- Reference 127, which presents the finite element analysis of heat transfer of typical CA walls and floor and a typical shield wall section in a three-hour standard fire. The analysis concluded that the SC structures are capable of behaving as three-hour fire barriers.
- Reference 128, which presents the heat transfer analysis of postulated fires on exposed Auxiliary Building structural steel floors. The analysis excluded the CA20 module floors, floors with a precast panel on the underside and floors containing embedded steel inside the concrete; for example, in Rooms 12255 and 12251. The report concluded that with the exception of CA63 in Room 12362 all CA models can be considered as acceptable fire barriers. Fireproofing material were added to protect CA20 submodule 63. A Corrective Action (CAPAL 100071674) is being used to track the required design change.

151. Reference 128 also identified four Rooms (12112, 12113, 12302 and 12301) where in the structural floors one of the rebars exceeds the temperature threshold of 538 ^{o}C. These were considered in Reference 129 and the conclusion was that the concrete sufficiently insulates the rebar, which provides adequate support under the considered loads.

152. Reference 130 presented the global finite element analysis results for the overall behaviour of the structure under the combined loading of dead load and fire loading. The document concluded that the walls and floors can maintain structural integrity without collapse. The analysis showed the outer temperature from a fire in Room 12264 to be 48.9 ^{o}C. This topic is further considered by the ONR's civil engineering assessment inspector as part of the overall closure of the GI-AP1000-CE-01.A7 (see ONR's civil engineering assessment report, Ref. 126).

153. Overall the submission provided the requisite information relating to the substantiation of the SC composite structures claimed in the Topic Report.

154. I am, therefore, satisfied that the claimed three hours fire resistance of the SC composite structures has been subjected to detailed substantiation and the RP demonstrated that they are suitably rated.

4.4.4 GDA ISSUE ACTION GI-AP1000-IH-01.A1 CONCLUSION

155. Overall the submission provides the requisite information in relation to the identification of internal fires, consequences analysis and substantiation of the nuclear significant hazard barriers claimed in the safety case.

156. I am satisfied that during GDA the claimed fire barriers have been subjected to detailed review and substantiation.

157. I am, therefore, satisfied that action GI-AP1000-IH-01.A1 can be closed.

4.5 ASSESSMENT OF GI-AP1000-IH-01.A2

158. In the **AP1000** plant, the HVAC system penetrates barriers that are required for the protection of redundant divisions of Class 1 SSCs. In a number of cases the fire dampers (115 fusible link, 18 smoke and 65 combination dampers) have to maintain the fire withstand capability of the barrier – three hours fire resistance (see sub-claim IH-1.2.3 above).

159. In response to GI-AP1000-IH-01.A2, the RP issued the AP1000 Fire Protection Dampers – UK Compliance Report (two revisions of this report, Refs. 131 and 132).

160. I assessed the above References and issued RQ-AP1000-1430 and RQ-AP1000-1590 to gain clarity on the following (Refs. 34 and 47):

- suitability of US codes and standards, and compliance with relevant UK law, codes and standards;
- classification of dampers;
- compliance with single failure criterion;
- fire rating of dampers;
- choice of dampers and installation;
- reliability of fusible link dampers.

161. The **AP1000** design generally utilised US codes and standards. During the review, the RP identified relevant UK codes and standards (such as BS 9999, BS EN15650, BS EN 13501-3, BS EN 1366-2 and BS EN 15882-2) and undertook a gap analysis between the US and UK codes and standards.

162. The UK fire protection dampers compliance report (Ref. 133) presents the outcome of the review of all dampers focusing on equipment specification, single failure criterion, control and logic, installation details and maintenance aspects.

The UK fire protection dampers compliance report and the Topic Report (Ref. 82) identified the following shortfalls or requirements for the detailed design:

- Fire dampers (fusible link, smoke and combination dampers) for the UK application of the **AP1000** design will be specified to comply with the legislative requirements in the UK, and to meet UK codes and standards.
- Class 1 fire barriers will be fitted with redundant combination fire/smoke dampers to address a single failure causing loss of function for redundant SSCs. This change applies to 16 locations in the non-radiologically-controlled side of the Auxiliary Building. It is also noted that five locations had existing fusible link fire dampers that will be replaced by combination fire/smoke dampers. This will take the total number of dampers on the **AP1000** plant to 214 from the existing total of 198 fire, smoke and fire/smoke dampers. This modification will satisfy ONR's SAP EDR.4.
- All fire dampers serving ventilation systems for the NI need to have position indication.
- In the radiologically-controlled side of the Auxiliary Building fire dampers in the exhaust sections of the HVAC system will be placed adjacent to the fire barriers to reduce the possibility of operators receiving high doses during maintenance.

163. The RP raised a Design Change Proposal (DCP) to capture the above modifications (APP-GW-GEE-5406 – UK AP1000 Fire Damper Changes) (Ref. 64).

164. The RP also provided examples of generic damper installation: through concrete wall fire damper; and out of the wall combination fire and smoke damper. The details of the arrangement, however, will be specified during the licensing phase when detailed design and supplier selection will take place.

165. I raised a regulatory query RQ-AP1000-1641 (Ref. 51) seeking clarity on the selection and installation of dual damper design on the Class 1 fire barrier between Rooms 12300 (Class 1 Division 'A' and 'C' Essential Electrical Supply System cables trays) and 12421 (Class 1 Division 'A' and 'C' Essential Electrical Supply System cable trays).

166. While the proposed installation will differ from the examples given in the UK fire protection dampers compliance report (Ref. 133), I am largely content with the submissions. It should be stated here that the specific detailed selection of dampers and final detailed design will be carried out post GDA and subject to the detailed requirements of the licensee.

167. The proposed change in turbine frequency from 60Hz to at 50Hz, stated above, may result in consequential changes to the size and layout of the Turbine Building and the type and location of fire, smoke and combination fire/smoke dampers in the Turbine Building, see **CP-AF-AP1000-IH-03**.

168. Overall, I am satisfied that the RP during the GDA provided sufficient information on the design and installation of the fire dampers.

169. I have confidence that the licensee would be able to implement the design proposal, identified above, during detailed design to meet all relevant UK legislative requirements and codes and standards.

4.5.1 GDA ISUE ACTION GI-AP1000-IH-01.A2 CONCLUSION

170. Overall the submission provides the requisite information relating to the substantiation of the approach taken to the design and installation of fire dampers claimed within the PCSR.

171. I am, therefore, satisfied that during GDA the dampers have been subjected to detailed review. I acknowledge, however, that the specific selection of dampers and final detailed design and substantiation will need to be carried out during the UK licensing phase.
172. I am, therefore, satisfied that action GI-AP1000-IH-01.A2 can be closed.

4.6 ASSESSMENT FINDINGS

173. During my assessment four assessment findings were identified for a licensee to take forward in their site-specific safety submissions. Details of these are contained in Annex 2.
174. 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.
175. Residual matters are recorded as assessment findings if one or more of the following apply:

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

4.7 MINOR SHORTFALLS

176. There are no minor shortfalls identified from my assessment of internal fire.

4.8 GDA ISSUE GI-AP1000-IH-01 CONCLUSION

177. Based on the conclusions above issue GI-AP1000-IH-01 can be closed.

5. ONR ASSESSMENT OF GDA ISSUE GI-AP1000-IH-02 – INTERNAL FLOOD

5.1 RESOLUTION PLAN ACTIONS AND DELIVERABLES

178. In GDA Step 4, GDA Issue GI-AP1000-IH-02 was raised (ONR Step 4 assessment report for internal hazards, Ref. 13) requiring the RP to provide an updated internal flooding safety case. This was due to the inconsistencies associated with the claims made on barriers, drains and sumps, and flood calculations. The GDA issue included the following action:

- GI-AP1000-IH-02.A1: Provide an updated internal flooding safety case that considers the claims, arguments and evidence associated with internal flooding.

179. The RP's Resolution Plan (Ref. 19) identifies specific deliverables associated with the above action:

- Internal Flooding Roadmap;
- Internal Flooding Topic Report;
- PCSR Chapter 11.

180. I raised regulatory queries RQ-AP1000-1302, RQ-AP1000-1336, RQ-AP1000-1443 and RQ-AP1000-1516 (Refs. 25, 30, 38 and 43) aiming to seek clarity on the scope of the submissions and the timescales given in the Resolution Plan.

181. In the following sub-sections, I will cover the following:

- the RP's safety case on internal flooding;
- my assessment GI-AP1000-IH-02, which includes:
  - assessment of claims, arguments and evidence including assessment of combined consequential hazards.
- my conclusions and assessment findings.

5.2 REQUESTING PARTY'S SAFETY CASE

182. The internal flooding safety case concludes that Category A safety functions can continue to be delivered by Class 1 passive SSCs. This is following a design basis flood through combination of claims made on structural barriers, equipment qualification, passive flood relief systems and administrative procedures.

183. Key document submissions for internal flooding are:

- UKP-GW-GL-793, Revision 0D – **AP1000** Pre-Construction Safety Report – Chapter 11 Internal Hazards (Ref.69);
- UKP-GW-GLR-107, Revision 1 – UK **AP1000** Internal Hazards – Flooding Topic Report (Ref.73).

5.3 AP1000 INTERNAL FLOODING CLAIMS

184. There are no specific claims made on the prevention of flooding. Flooding is assumed to occur as a result of gross failure of pipes, vessels and components containing fluids. Sources of flooding have been identified in the Auxiliary Building under Table 11.3-1 of the PCSR (Ref. 69) and in the Containment under PCSR Table 11.3-2 (Ref. 69).

185. The overarching high level safety claim addressing the internal flooding challenges within the **AP1000** design basis is summarised as (Ref. 73):

- **Claim IH-2.0**: Postulated internal flooding events within the design basis will not prevent the delivery of the Category A safety functions by Class 1 SSCs and supporting post 72-hour Category B safety functions necessary to respond to the postulated event.

186. The following key claims and sub-claims underpin the high level safety claim:

- **Claim IH-2.1:** Class 1 SSCs required for delivery of Category A Safety Functions are protected from sources of internal flooding by civil/structural barriers.
- **Sub-claim IH-2.1.1**: PXS Room A and PXS Room B will not flood concurrently.
- **Sub-claim IH-2.1.2**: Flooding in the non-RCA Auxiliary Building mechanical areas will not spread to the electrical areas of the Auxiliary Building.
- **Claim IH-2.2:** Where Class 1 SSCs are not qualified to operate in a submerged state, sources of flooding will be isolated prior to exposing Class 1 SSC to a flooded environment.
- **Claim IH-2.3**: Where required, Class 1 SSC will be capable of delivering the Category A safety function when submerged.
- **Claim IH-2.4:** The available inventory of flood sources has been minimised so far as is reasonably practicable.
- **Claim IH-2.5:** Flooding will be alleviated by passive flood relief measures.
- **Sub-claim IH-2.5.1**: Flood heights in Division A, B, C & D I & C Rooms, 12301, 12302, 12304 & 12305, will not exceed 0.076 m.
- **Sub-claim IH-2.5.2:** Flood height in the Valve/Pipe Penetration Room, 12306, will not exceed 0.533 m.
- **Sub-claim IH-2.5.3:** Flood height in Middle Annulus, 12341, will not exceed 2.36 m.
- **Sub-Claim IH-2.5.4:** Flood heights in the Main Steam Isolation Rooms, 12404 and 12406, will not exceed 0.91 m.
- **Sub-claim IH-2.5.5:** Flood heights in the Truck Bay / Filter Storage Area and RNS HXs Rooms, 12371, 12372 and 12362 will not exceed 1.22m
- **Sub-claim IH-2.5.6:** Flood heights in the VBS MCR / A&C Equipment Room, 12501, will not exceed 0.152m.
- **Sub-claim IH-2.5.7:** Internal doors will not retain the full volume of fluid within the affected room.

187. The RP identified the following, including SSCs, which deliver the above claims:

- minimising flood sources and volumes;
- withstand capacity of barriers to retain water in flood areas to prevent flood propagation;
- providing engineering discharge routes to alleviate the effects of flooding. These include doors, drains and pressure relief panels and are discussed in detail below;
- location of Class 1 SSCs at an elevation above the maximum flood height;
- separation of Class 1 SSCs within containment;
- qualification of Class 1 SSCs against submergence;
- administrative procedures to isolate internal flooding.

5.4 ASSESSMENT OF GDA ISSUE GI-AP1000-IH-02

5.4.1 SCOPE OF THE ASSESSMENT

188. The assessment strategy in Section 2 was used to formulate the scope. My assessment covers the deliverables used in addressing the actions for the Resolution Plan for GDA Issue GI-AP1000-IH-02.

189. I assessed the Topic Report and the internal flooding section of the PCSR. I sampled supporting documents to obtain confidence on the requisite evidence and substantiation of the claims made.

190. The areas chosen to review the internal flooding safety case were limited to:

- inside Containment;
- outside Containment – Auxiliary Building only.

5.5 ASSESSMENT OF CLAIMS, ARGUMENTS AND EVIDENCE

191. The RP undertook a systematic identification of flooding scenarios and consequences analysis which culminated in the derivation of specific claims as given above. An internal flood is deterministically assumed to occur. There are no specific claims made on the prevention of internal flooding hazards.

192. I assessed the RP's "internal flooding safety case roadmap" (Ref. 99) along with responses to regulatory query RQ-AP1000-1302 /1336 (Refs. 25 and 30) during the initial stages of the assessment process. This was to aid my understanding of the safety case structure. As the safety case documentation developed, the roadmap was superseded. Up-to-date information is provided in the PCSR and the internal flooding Topic Report (Refs. 69 and 73, respectively).

193. The NI consists of a free-standing steel Containment Vessel, a concrete Shield Building and the Auxiliary Building outside Containment. These are considered as principal flood areas.

Inside Containment

194. Containment is a largely open space to maintain the free movement of gases and therefore does not contain full-height internal walls. The four key areas in Containment are the Chemical and Volume Control System (CVS) Room (11209), the Passive Core Cooling System (PXS) Room B (11207 and 11208), the PXS Room A (11206) and the general Containment area. These compartments are segregated by partial walls and floors (UK **AP1000** barrier matrix, Ref. 104).The partial walls, described in the internal flooding Topic Report (Ref.73), have been designed to heights of:

- refuelling cavity – 109.28ft (102.8m);
- CVS room – 110.00ft (103.05m);
- PXS room B – 110.08ft (103.07m);
- PXS room A – 110.17ft (103.10m).

195. Curbs around the top of each compartment determine the sequence in which each one floods. The argument is that partial walls separate each key area and flooding in one area will not propagate to another area unless the height of the flood exceeds the height of the separating partial wall identified in Figure 10-1 of the flooding Topic Report (Ref. 73). Therefore, for a postulated flood in the general Containment area, fluid will flood up through the vast volume to a height of 109.28ft before overflowing into the refuelling cavity. The cavity will fill up before the fluid rises and over-spills into the CVS room, then the PXS Room B and finally the PXS Room. The drain line from each of these areas contains two backflow preventers in series.

196. Based on a bounding flood source, these partial walls, curbs and backflow preventers deliver sub-claim IH-2.1.1.The RP substantiated this sub-claim through identifying bounding sources in the general Containment space, the CVS room and the PXS rooms under regulatory query RQ-AP1000-1676 (Ref. 54).

Flood volumes and the flood heights have been analysed and presented in Table 3 of the flooding Topic Report (Ref. 73). The supporting analysis is provided in APP-SSAR-GSC-743 (Ref. 94) with LOCA and non-LOCA events compared in APP-PXS-M3C-034 (Containment Flood-Up Level, Ref. 93). The partial walls were assessed for hydrostatic load withstand capability under APP-1000-CCC-017 (Structural acceptance of PRHA internal flood in auxiliary and shield buildings, Ref. 97). These key areas along with the partial walls are identified in the Barrier Matrix (Ref. 104).

197. The general Containment space, up to the curb height of the CVS room has a volume of 2911m^3 (102,789ft^3). A double end guillotine failure of the RCS or PXS pipework would result in 2,540m^3 (89,538 ft^3) of fluid leaking into the general Containment space. Therefore, the flooding would be retained in the general Containment space without affecting the CVS room or the PXS rooms. However based on the actuation of the Automatic Depressurisation System (ADS), the bounding flood in the general containment space is from the Central Chilled Water System (VWS) pipework and the volume becomes 2859m^3 (100,960ft^3). Therefore, the general Containment would flood up to the 110.08ft level and fill the CVS room before an additional 150m^3 of fluid would cascade into PXS Room B. PXS Room A would be unaffected by this release and would have sufficient SSCs to mitigate against the effects of the flooding scenario.

198. However, both PXS Room A and PXS Room B contain SSCs which maintain reactor coolant inventory. These are valves which are identified on the hazard schedule (Flooding topic report, Ref. 73, Hazard Refs. FL2-3). If this flooding scenario were to occur, valves in both PXS rooms are qualified to operate in a submerged environment. This delivers claim IH-2.3. The environmental conditions are detailed in '**AP1000** environment conditions for equipment qualification' (Ref. 92). The equipment qualification methodology is covered in '**AP1000** equipment qualification methodology' (Ref. 89).

199. A CCS pipe failure in the CVS room, assuming Automatic Depressurisation System (ADS) actuation would release 2770m3 (97,960 ft^3) of water resulting in flooding of the general Containment up to the 110.08ft level, complete submergence of the CVS room and the remaining 65m^3 cascading into the PXS valve Room B. PXS Room A would be unaffected by this release. The valves in both PXS rooms are qualified to operate in a submerged environment (the **AP1000** equipment conditions report, Ref. 92).

200. The bounding release in the PXS valve Room B is from the CVS. Assuming ADS actuation, a total of 2,716 m^3 (95,925 ft^3) would be released. The result would be flooding of the PXS valve Room B to the 99.8m (99.35ft) level, the general Containment space to the 110ft level and the CVS room to the 99.1m (97.06ft) level. PXS valve Room A would be unaffected by this release. As referenced above, the valves in both PXS rooms are qualified to operate in a submerged environment.

201. It was also noted that Direct Injection Line (DVI) LOCA and non-DVI LOCA hazard scenarios were analysed in the Containment flood-up level report, (Ref. 93). Pressure part failure effects as a result of a DVI line break are discussed in Section 6 of this report. The maximum flood height was identified as 110.12ft from the non-DVI LOCA case (Containment flood-up level report, Ref. 93, Case 8) and therefore does not affect PXS Room A.

202. The bounding release in PXS valve Room A is failure of the Spent Fuel Pool Cooling System (SFS) and would result in flooding of PXS Valve Room A to the 97.8ft level. Assuming actuation of the ADS, the general Containment space would flood to the 110ft level, with the remaining 207m^3 (7,307ft^3) spilling into the

CVS room. PXS Valve Room B would be unaffected by this release. As referenced above, the valves in both PXS rooms are qualified to operate in a submerged environment.

203. The refuelling cavity is flooded during refuelling operations and therefore the SSCs present in these areas are qualified to operate in a submerged environment. The environmental conditions are detailed in (the **AP1000** equipment conditions report, (Ref. 92). The equipment qualification methodology is covered in ‘**AP1000** environment conditions for equipment qualification’ (Ref. 89).

204. Environmental conditions are detailed in the **AP1000** equipment conditions report (Ref. 92). The RP stated that this reference is not applicable to United Kingdom Standard Plants (UKP) on the basis of radiological and thermal conditions. However, these conditions are considered to have no significant impact on a postulated flooding event. The calculated maximum flood inventory and key Class 1 barriers are expected to remain as per the standard **AP1000** design. The Design Reference Point (Ref. 64) already identifies that APP-GW-VP-030 needs to be created as a UKP document and capture specific UKP changes. Impacts of changes should be reviewed by the licensee. Therefore I consider it proportionate that the **AP1000** reactor equipment conditions report may be used for the assessment of internal flooding in GDA.

205. For the scenarios analysed, I consider that the evidence and arguments to underpin sub-claim IH-2.1.1 and claim IH-2.3 inside Containment are reasonable and are line with SAPs EHA.1, EHA.6, EHA.7 and EHA.15.

206. I noted that the margin between the heights of the partial walls inside Containment appeared to be minimal from Figure 10-1 of the flooding Topic Report (Ref. 73). For example, PXS Room B has a partial wall of 110.08ft compared with PXS Room A which has a wall height of 110.17ft. The layout of the PXS Rooms is identified in the flooding Topic Report under Figure 10-14 which shows elevation 92’-6” and the associated partial walls are identified in Figure 10-15 on elevation 100’ 0” (Ref. 73).

207. I sought clarification on the layout and flood path inside Containment. From my discussions with the RP, it was observed that Figure 10-1 (flooding topic report, Ref. 73) is slightly misleading. I created Figure 1 (Annex 1) which shows a simplified elevation schematic of the PXS Room A and Room B. This is to aid my understanding and readers of the flow path inside Containment.

208. I sampled the pipe rupture scenario inside PXS Room B. The water level would be channelled up the partial walls up to a wall height of 110.08ft. From there the water transitions over the curbs and drops into the general Containment area which is at an elevation of 107’-2”. Therefore, the height difference is approximately three feet but there is a vast volume to fill. I consider that the distance between PXS Room A and Room B is at a sufficient distance such that no inadvertent filling of PXS Room A will occur if there is a leak in PXS Room B, and vice versa (refer to Figures 10-14 and 10-15 of flooding topic report, Ref. 73). Therefore, the separation distance between the rooms, in combination with the partial walls, would reduce the likelihood of both rooms flooding concurrently and deliver sub-claim IH-2.1.1. I consider the evidence to take cognisance of SAPs EHA.1, EHA.6, EHA.7 and EHA.15.

Outside Containment

209. In the NI, the Auxiliary Building is located outside Containment. The Auxiliary Building comprises the Radiologically Controlled Area (RCA) and a non-RCA.

Adjoining buildings are also physically segregated from the Auxiliary Building by walls, floors and ceilings. Access to either the RCA or non-RCA portions of the Auxiliary Building is from the adjoining buildings, which are themselves sub-divided into RCA and non-RCA sections (as appropriate) by walls, floors and ceilings.

210. The RP used the following methodology to carry out a deterministic analysis of internal flooding:

- identifying all sources resulting in internal flooding on a room-by-room basis in the Auxiliary Building;
- calculation of the maximum inventory;
- identification of flood paths and areas affected;
- the maximum flood height in affected rooms or areas;
- barriers subjected to postulated floods assessed for hydrostatic load withstand capability against Reference 97;
- output compared against the design criteria (Refs. 91 and 95) and against the location of Class 1 SSCs.

211. The above approach is broadly in line with my expectations and I consider in line with the IAEA Safety Guide NS-G-1.11 (Ref. 7).

212. The Auxiliary Building RCA and non-RCA are two principal flood areas. The Auxiliary Building RCA is segregated from the non-RCA by 2 ft thick concrete walls and floors as a minimum (flooding topic report, Ref. 73). Therefore a flood which initiates on the RCA side cannot propagate to the non-RCA side of the Nuclear Island, and vice-versa. All sources of flooding are identified in Table 11.3-2 of the PCSR Chapter 11 (Ref. 69). The identified flood barriers are designed to withstand the loading based on a maximum flood height and were assessed for hydrostatic load withstand capability against APP-1000-CCC-017 (Structural acceptance of PRHA internal flood in auxiliary and shield buildings, Ref. 97). The barriers are identified in the Barrier Matrix (Ref. 104). Penetrations in the flood retaining barriers were either minimised below the maximum flood height or sealed to eliminate flow paths (Pipe rupture protection design criteria for the **AP1000** plant, Ref. 90). Therefore, these Class 1 barriers support claim IH-2.1 and are line with SAP EKP.3.

213. However, the RP did not make any claims on barriers to prevent flooding from reaching the different divisions. Instead the overarching claim used for internal flooding is that Class 1 SSCs will be available to deliver Category A safety functions and supporting post 72- hour Category B safety functions. Therefore, there are 0.5-inch (1.27 cm) door gaps in place acting as passive flood relief measures. These gaps alleviate flood heights in a room or compartment to prevent the flood levels from reaching the Class 1 SSCs. These gaps partly support sub-claim IH-2.5.7 which addresses SAP EKP.3.

214. However, it is not clear how the door installation will affect the flow dynamics through the door. I consider that a 0.5inch gap is a tight tolerance and my expectation is that ‘cliff-edge’ effects should be adequately addressed in line with SAP EHA.7. In addition, as the door gaps are crucial in distributing flood levels, door installation, long term monitoring and inspection should be addressed by the licensee. This should take cognisance of SAPs EMT.5 and EMT.6 to ensure that the door gap remains as per design throughout the plant lifecycle. Therefore, I am raising an assessment finding to ensure that sensitivity analysis is carried out to assess any impacts on Class 1 SSCs as a result of reduced door gaps.

CP-AF-AP1000-IH-05 – The licensee shall carry out flooding sensitivity analysis on the reduced door gap heights, and on the assumed blockage and redundancy of the drains, and assess any impacts on Class 1 SSCs delivering or contributing to Category A functions.

215. An assessment of the rooms where SSCs are located and the maximum flood height have been determined Flooding topic report, Ref. 73, Table 3). SSCs in the Auxiliary Building are summarised in '**AP1000** PRHA Safety-related equipment flooding target identification' (Ref. 95) and inside containment are summarised in 'Safety-related equipment flooding target identification in containment' (Ref. 169). To deliver sub-claim IH-2.3, equipment will be qualified for submergence due to flooding / wetting and is captured under the equipment qualification methodology (Ref. 89). Calculation outputs and environmental conditions are summarised in the **AP1000** environment conditions report (Ref. 92).

216. To deliver sub-claim IH-2.4, pipework comprising each of the fluid systems has been routed to minimise the number of rooms through which they pass. Significant accumulations of water have been located outside of the Nuclear Island where practicable. I consider this is in line with SAP. ELO.4.

217. The FPS is present in the Auxiliary Building RCA, non-RCA and other areas outside of the Nuclear Island. During all plant modes, the Auxiliary Building Non-RCA is isolated from the tanks via a locked valve (FPS-PL-V101). The FPS instead draws water from the Passive Containment Cooling Water Storage Tank (PCCWST) via a standpipe. The standpipe reduces the maximum draw volume to 99 m^3. This is in line with SAP EKP.3. Within containment, the FPS is operational when the reactor is shut down (plant modes 5-6) but isolated at other times (plant modes 1-4). Plant modes are described in detail in the PCSR Chapter 6 (Ref. 66). In all plant modes, the FPS inventory is bounded by Hazard Ref. FL6 confirmed in regulatory query RQ-AP1000-1796 (Ref. 59).

Auxiliary Building Non-RCA

218. The non-RCA side has one flood compartment from the basemat up to elevation 97.714m (92'-6"). At elevation 100m (100'-0") and above, more compartments (two or more) have been identified which separate out the areas containing piping or mechanical equipment from the electrical areas, and including separate compartment for the MCR area.

219. In the non-RCA-side, rooms containing mechanical equipment are 12306, 12404, 12405 and 12406. These rooms are individual flood compartments. The identified flood barriers are designed to withstand the loading based on a maximum flood height and were assessed for hydrostatic load withstand capability against Reference 97. The barriers are identified in the Hazard Barrier Matrix (Ref. 104). Penetrations in the flood-retaining barriers were either minimised below the maximum flood height or sealed to eliminate flow paths (Flooding topic report, Ref. 73). Therefore, the Class 1 barriers deliver sub-claim IH-2.1.2 and support claim IH-2.1 which are in line with SAP EKP.3.

220. To support claim IH-2.2, differential pressure level sensors located in the Auxiliary Building RCA and Auxiliary Building non-RCA will alert operators, via the PMS. Protection is via procedure and requires operators to isolate of the affected system.

221. There are two redundant differential pressure level sensors (WLS-LT-400A and WLS-LT-400B) within the Auxiliary Building RCA Sump Room (12154), located at 0.30 m (1 ft) above the 89.8m (66'-6") level. These differential pressure level

sensors are provided to ensure that all sources of flooding can be isolated prior to the 95.58 m (85'-6") criteria flood-up level being exceeded; the criteria flood-up level being the point at which Class 1 SSC would become submerged and therefore cease to function.

222. The level detection does not automatically close isolation valves. SAP ERL.3 states that automatically initiated measures should be provided where "*reliable and rapid protective action is required*." The time to reach the first alarm point is just over 5 minutes. This alarm would alert the operators initially of a problem. Operators would then have approximately 7.5 hours to carry out an isolation before there would be a potential impact to a SSC. This is documented in regulatory query RQ-AP1000-1796 (Ref. 59).

223. I discussed this with the human factors assessor and conclude that this is an adequate amount of time for an operator response. However additional analysis post-GDA on human performance is required to incorporate emergency and site impacts and should address SAPs EHF. 1 and EHF.7. This is captured under by ONR's human factors specialist assessment inspector in his assessment report (Ref. 176) and therefore no additional internal hazard assessment findings will be required here.

224. The RP claimed that flooding will be alleviated by passive flood relief measures. The sub-claims supporting this key claim are all based on the non-RCA side of the Auxiliary Building. Sub-claim 2.5.7 was examined above.

225. The following paragraphs examine sub-claims IH-2.5.1, IH-2.5.2, IH-2.5.4 and IH-2.5.6. For calculation purposes, drains in rooms containing High Energy (HE) pipework are assumed to be 100% blocked. This is considered as the conservative case for a resulting flood height. In standard operation, it is expected that the drains would be at least partially available and therefore the flood height would be less than the figures calculated. In rooms subject to a Medium Energy Line Break (MELB), the drains are assumed to be 25% blocked. The latter lines were argued to result in less debris and hence the partial availability of drains for hazard scenarios containing MELB.

226. In Room 12300, there are five 4inch drains (WWS-D212 to WWS-D216), which have a combined capacity of 1250 gpm. These feed into a common 6 inch header, which has a capacity of 312 gpm. However, the bounding pipe rupture in room 12300 is 295 gpm from the Fire Protection System pipe FPS-PL-L159. This has the potential to flood Rooms 12301, 12302, 12034 and 12305. The five 4inch drains require to discharge at 25% of their combined capacity to ensure that flooding in Room 12300 and surrounding areas remains negligible.

227. This could be achieved by two drains being 100% available with the other three being blocked or by all five drains providing a fraction of their design capacity. The drains discharge to the Auxiliary Building North Sump. Once the sump was full, the level would backflow into the 12111 corridor and potentially reach the Class 1 battery rooms. However, as the source is from the PCCWST, the inventory is limited to 99 m^3. Furthermore the flood paths preferentially drain to the sump first, and therefore any accumulation of FPS water on this floor is insufficient to challenge Class 1 SSCs present. Therefore the water level would not exceed 0.076m which would meet sub-claim IH-2.5.1 (hazard schedule Ref. FL9 in the flooding topic report, Ref. 73). This is in line with SAP EKP.3.

228. In Room 12306, there are two 4-inch drains (WWS-D104 and WWS-D235) which have a combined design capacity of 500gpm. This feeds into a common 4-inch header with a capacity of 250gpm. However, the bounding pipe rupture in Room 12306 is 670gpm from the FPS pipe FPS-PL-L148. Therefore, the header is the

limiting factor for discharge rate and water accumulation in Room 12306 will occur. The RP gave credit to the 0.5-inch door gap connecting through to the Turbine Building to act as an additional passive relief measure (sub-claim IH-2.5.7). Any fluid would drain to the Turbine Building Sump. The Turbine Building is outside of the scope of my assessment. Furthermore, flooding of the Turbine Building is not a concern, during GDA, as no Class 1 SSCs are located therein.

229. I sampled the analysis for the flood height determination under Reference 101. The combination of the limited inventory from the PCCWST, the 0.5inch door gap and a drainage rate of 250gpm limits the maximum flood height to 0.533m (1.75 ft) from the floor. The drainage rate could be achieved by two drains being 100% open with the other two drains blocked with debris or by all drains providing 50% of their design capacity. The maximum flood level is below the point at which SSCs would become submerged, which is 0.92m (3ft). This delivers sub-claim IH-2.5.2 (hazard schedule ref. FL11 in the Flooding topic report, Ref. 73). This is in line with SAP EKP.3. Although I am largely satisfied with the case presented here, it will be prudent to undertake sensitivity studies, during detailed design, on the assumptions made on the blockage and redundancy of the drains to confirm that the current drainage provisions are suitable and sufficient and that no additional passive flood control systems are reasonably practicable (see assessment finding CP-AF-AP1000-IH-05).

230. Steam Generator System (SGS) line breaks in Rooms 12404 and 12406 have been identified as the bounding cases for the Auxiliary Building non-RCA. The flood heights following a failure of the SGS are 2.05 ft in Room 12404 and 2.03 ft in Room 12406. The flood height is limited by Class 1 pressure relief panels (12404-AY-P01 and 12406-AY-P01 respectively) in each of the rooms which discharge flooding to the Turbine Building. The flood heights are within the design criteria to limit the flood height in Rooms 12404 and 12406 to less than 3 ft (0.91m) and have been designed accordingly (Design Specification for the relief panels, Ref. 88). These pressure relief panels deliver sub-claim IH-2.5.4. This is in line with SAP EKP.3.

231. As mentioned inside Containment above, environmental conditions are detailed in the **AP1000** equipment conditions report (Reference Ref. 92). The RP stated that this reference is not applicable to UKP on the basis of radiological and thermal conditions. However, these conditions are considered to have no significant impact on a postulated flooding event. The calculated maximum flood inventory and key Class 1 barriers are expected to remain as per the standard **AP1000** design. The Design Reference Point (Ref. 64) already identifies that APP-GW-VP-030 needs to be created as a UKP document and capture specific UKP changes. Therefore, I consider it proportionate that the **AP1000** reactor equipment conditions report may be used for the assessment of internal flooding in the GDA.

232. ONR was notified on 18 January 2017 of a new internal flooding sub-claim, IH-2.5.6, "*Flood heights in the VBS MCR / A&C Equipment Room, 12501, will not exceed 0.152m*."

233. The hazard schedule shows a flooding source in Room 12501 from the DWS (DWS-PL-L400). This has a maximum capacity of 477m^3 (Hazard ref. FL16 in the Flooding topic report, Ref. 73). The DWS pipework (DWS-PL-L400) incorporates an orifice plate located in the Annex Building to limit the flow to less than 1.89x10^{-02} m^3/s (300gpm) as part of the design change proposal covering changes due to these flooding scenarios APP-GW-GEE-4568 (Ref. 183).

234. I sampled the Data Sheet Report (Ref. 184) and the Orifice Data Sheet (Ref. 185) supplied as part of regulatory query RQ-AP1000-1796 (Ref. 59). The

orifice maximum design flow is designed to 250gpm (Ref. 185). The references to the orifice plate currently only have applicability in **AP1000** China Plants (CPP), but their design change proposal commits the RP to implementing an equivalent one for the UK. Therefore, I consider it proportionate that the Data Sheet Report and the Orifice Data Sheet may be used for the assessment of flooding in GDA. As the DWS has a limited inventory, the RP did not considered this to be the bounding case.

235. There are three floor drains (WWS-D251/D229/D230) which discharge water to the Turbine Building Sump. The Turbine Building is outside the scope of my assessment. If drains are credited and allowing for blockage, the design of 75% availability is 312gpm. Therefore, I consider that the drains are an adequate DiD measure for this scenario. See also assessment finding CP-AF-AP1000-IH-05.

236. The bounding case in Room 12501 is from the unlimited supply of Potable Water System (PWS) (Hazard ref. FL7 in the Flooding topic report, Ref. 73). The PWS flowrate is 44gpm which is within the capacity of the floor drains, as discussed above.

237. A consequence analysis for UK **AP1000** reactor was provided in a supporting report, 'UK **AP1000** flooding variations' (Ref. 101), which I sampled. No credit was provided for the drains (WWS-D229/D230/D251, described above). It would take in the order of 3.53 hours before water would reach the door gap and distribute to other areas (UK **AP1000** flooding variations, Ref. 101). If the water depth exceeded 0.152m, then the RP advised in the response to regulatory query RQ-AP1000-1796 (Ref. 59) that water would potentially flow downwards into the Main Control Room. There is also Class 1 systems in the MCR but these are not required for safe shutdown caused by an internal flood.

238. There is a curb located in front of the staircase to prevent water flow from travelling down the stairwell which was advised in regulatory query RQ-AP1000-1796 (Ref. 59). In addition the door between 12411 and 12400 vestibule has been designed as a water tight door. This is identified on the Barrier Matrix (Ref. 104). Therefore, the combination of the passive relief measures in Room 12501 deliver sub-claim IH-2.5.6. This is in line with SAPs EKP. 2 and EKP.3.

239. Due to the MCR containing four safety divisions. I sampled the DiD arguments further. If the above measures failed, the consideration of water in the MCR is bounded by a similar PWS failure scenario. The Operator Break Room 12401 is supplied via a 1-inch unlimited PWS line. The pipework is confined to the northwest corner of the MCR breakout area in the kitchen and restroom areas. Failure of this pipework would initially pool on the kitchen floor and restroom areas before spreading out beneath the suspended floor of the operator breakout area and main control area.

240. Cable trays containing 'A', 'B', 'C' and 'D' of the Class 1E DC and Uninterruptible Power Supply System (IDS) are present in the space beneath the MCR suspended floor. As such, these would become wetted. However, the cabling is a continuous run with no connections and is qualified for harsh environments, including submergence in water (Ref. 92). Therefore, the RP argues that they would function as normal.

241. The PWS flowrate is limited to 2.8E-03 m^3/s (44gpm) as noted in Reference 101. No credit has been given to drains in the area. After 390 minutes (about 6.5 hours), the space beneath the floor would become completely submerged and water would pool on the floor's surface. Flooding would have to continue for a further 25 minutes before the critical flood height in the MCR of 2.54 cm (1-inch)

is reached (**AP1000** PRHA safety-related equipment flooding target identification, Ref. 95).

242. The MCR is continuously manned and operators would be expected to frequently pass into the Operator Break Room. With the audible indication of running water and likely pooling of water in the kitchen / restroom areas, it is considered likely that the failure would be detected. Protection would be via administrative procedure. The kitchen and restroom floor areas are fitted with an additional four drains connected to the Sanitary Drainage System (SDS) which would provide additional defence-in-depth. The drains are capable of accommodating the full flow rate of a PWS pipe break. Therefore, I consider that the DiD arguments for the MCR are adequate and are in line with SAP ELO.4.

243. In considering the Auxiliary Building non-RCA, I judged that further sensitivity analysis is required to underpin claim IH-2.5.7. However, I am satisfied that the evidence and arguments to underpin sub-claims IH-2.1.2, IH-2.5.1, IH-2.5.2, IH-2.5.4, IH-2.5.6 and claim IH-2.1, IH-2.2 and IH-2.4 in the Auxiliary Building non-RCA outside Containment are suitable and sufficient.

Auxiliary Building RCA

244. The RCA side contains the Spent Fuel Storage Pit (12563) and associated areas such as the Waste Hold Up Tank Rooms (12166 and 12167) and the Cask Loading Pit (12463) which have flood barriers. These rooms are designed to be flooded and therefore barriers are not in place for nuclear safety purposes. Excluding those areas, the Auxiliary Building RCA has one flood compartment from the basemat up to elevation 97.714m (92’-6”). At elevation 100m (100’-0”), Room 12564 becomes its own flood compartment but is designed to be flooded as part of the spent fuel handling system. Room 12373 also becomes a separate compartment. Above this elevation, excluding the spent fuel area, the RCA become a single flood compartment.

245. The Auxiliary Building RCA is segregated from the non-RCA by 2 ft-thick concrete walls and floors as a minimum (Flooding topic report, Ref. 73). Therefore a flood which initiates on the RCA side cannot propagate to the non-RCA side of the NI, and vice-versa. All sources of flooding are identified in Table 11.3-2 (PCSR Chapter 11 Ref. 69). The identified flood barriers are designed to withstand the loading based on a maximum flood height and were assessed for hydrostatic load withstand capability against ‘Structural acceptance of PRHA internal flood in auxiliary and shield buildings’ (Ref. 97. The barriers are identified in the Hazard Barrier Matrix (Ref. 104). Penetrations in the flood retaining barriers were either minimised below the maximum flood height or sealed to eliminate flow paths (“Blockouts and Barriers - Penetrations, Seals and Fire Stops”, Ref 87). Therefore, these Class 1 barriers support claim IH-2.1 and are in line with SAP EKP.3.

246. Within the RCA side of the Auxiliary Building, the RP considered the break of the CVS pipe in room 12255, leading to a consequential failure of the Component Cooling Water System (CCS), FPS and VWS systems as the bounding case (Flooding topic report, Ref. 73). The CVS Room runs through 12156, 12255, 12258 and 12259. The affected systems have a combined volume of 2,960m^3 (780, 643 US gallons). This volume will be discharged at a rate of 0.378 m^3/s (5986 gpm).

247. There are no Class 1 SSCs located on Level 1 of the Auxiliary Building (elevation 89.8m or 66’ 6’). However, Class 1 SSCs would become submerged once flooding progresses to Level 2 (elevation 84.7m or 82’ 6”). It would take

approximately 263 minutes (approximately 4 hours) for flooding to reach the 94.67m (82'-6") level, at which point the CCS and VWS would be depleted.

248. There are two redundant differential pressure level sensors (WLS-JE-400A and WLS-JE-400B) within the Auxiliary Building RCA Sump Room (12154), located at 0.30 m (1 ft) above the 89.8 m (66'-6") level. These differential pressure level sensors are provided to ensure that all sources of flooding can be isolated prior to the 95.6m (85'-6") criteria flood-up level being exceeded; the criteria flood-up level is the point at which Class 1 SSC would become submerged and therefore cease to function.

249. The level detection does not automatically close isolation valves. SAP ERL.3 states that automatically-initiated measures should be provided where "*reliable and rapid protective action is required*." The time to reach the first alarm point is just over 5 minutes. This alarm would alert the operators initially of a problem. Operators would then have approximately four hours to carry out an isolation before there would be a potential impact to a SSC.

250. The time available for operators to take action is nearly half that required in comparison with the Auxiliary Building non-RCA. Although a rapid response does not appear to be required, four hours may be considered a tight timescale if there are difficulties in locating and isolating a flood. I discussed this with the human factors assessor and concluded this to be an adequate amount of time for an operator response for GDA considerations. Additional human factor analysis is required post-GDA on human performance to incorporate emergency and site impacts and should address SAPs EHF. 1 and EHF.7. This is covered by ONR's human factors specialist assessment inspector in his assessment report (Ref. 176) and therefore no additional internal hazard assessment findings will be required here.

251. The bounding pipe rupture in the Middle Annulus 12341 is 1227.5 gpm from the FPS pipe FPS-PL-L167 in Room 12351. Room 12341, along with other rooms, is located in the same flood compartment as Room 12351 on elevation 100m (100'0"). These are identified on the Barrier Matrix (Ref. 104) and the hazard schedule (within the Flooding topic report, Ref. 73). Therefore, there are a number of doors each with a 0.5-inch door gap which acts as an additional passive flood relief measure (sub-claim IH-2.5.7). The critical flood height at which Class 1 SSC would become submerged within Room 12341 is 2.36m (7.75ft).

252. To mitigate against internal flooding, the Middle Annulus 12341 contains seven 4-inch drains (WRS-D318/D319/D321/D324/D328/D330/D333) which have a combined design capacity of 1750gpm feeding into a common 4-inch header (WRS-PI-L218) with a capacity of 250gpm. In Room 12351, the four 4-inch drains (WRS-D313 to WRS-D316) have a combined design capacity of 1000gpm which feed into a separate common 4-inch header (WRS-PI-L313) with a capacity of 250gpm. Therefore, the common headers are the limiting factor for the discharge rate. The combined discharge rate from Rooms 12341 and 12351 is therefore 500gpm. This will result in water accumulation in Room 12341.

253. I sampled the calculation for the determination of the flood height in Rooms 12341 and 12351 under Reference 96. The combination of the limited inventory from the PCCWST, the 0.5-inch door gaps and a drainage rate of 500gpm limits the maximum flood height to 2.31m (7.58ft). The drainage rate would be achieved with a combination of one drain in Room 12341 and one drain in Room 12351 being 100% available or all 11 drains working at reduced capacity. The RP argued that this delivers sub-claim IH-2.5.3 (hazard schedule ref. FL28 within the Flooding topic report, Ref. 73).

254. However, the margin between the calculated volume and the associated flood height (2.31m) versus the height at which Class 1 SSCs would become submerged (2.36m) is minimal. SAP ERL.4 requires that a margin of conservatism be considered to allow for uncertainties. Considering an unmitigated consequence if drains were blocked, flood waters would distribute into various rooms identified in Figure 10-15 (Flooding topic report, Ref. 73). The hazard schedule indicates a number of SSCs would be potentially affected. On further examination via regulatory query RQ-AP1000-1796 (Ref. 59), the RP advised that only the SFS level transmitter SFS-JE-LT019B would become submerged, but argued that it was qualified for this flooding condition. No other impacts were advised under the regulatory query RQ-AP1000-1796 (Ref. 59). Therefore, this is considered as in line with SAPs EKP.2 and EKP.3.

255. ONR was notified on 18 January 2017 of a new internal flooding sub-claim, IH-2.5.5 'Flood heights in the Truck Bay / Filter Storage Area and RNS HXs Rooms, 12371, 12372 and 12362 will not exceed 1.22m.' (Flooding topic report, Ref. 73).

256. In these rooms, the failure of the FPS line FPS -PL-L054 has been considered. Unmitigated flood levels would have the potential to release radioactivity from the stored CVS filters (Ref. 73, Hazard Ref. FL31). DCP APP-GW-GEE-4568 (Design change proposal – flooding in auxiliary building etc., Ref. 183) added a Class 1 pressure relief flapper (12362-AD-D02) at the floor elevation of Room 12362. This alleviates the flood height of the area and distributes the water via drains. The inventory is limited to 2006m^3 (530,000 US gallons) which was determined in the PRHA auxiliary building analysis of internal flooding etc. (Ref. 96). I surface-sampled the analysis and flood height outputs. The flood heights are within the design criteria to limit the flood height in these rooms to 4ft (1.219m).

257. Due to time constraints, I had not sampled the technical design specification of the pressure relief flapper, drain details and the flood calculation in detail to check for the adequacy of its design. I also did not carry out detailed sampling to compare why the FPS volume on the RCA side (2006m^3) was different to the FPS volume on the non-RCA side (99m^3) of the Auxiliary Building. However, I sampled other areas, as described above, and I have confidence in the analysis undertaken.

258. Rooms 12371, 12372 and 12362 fall into the same flooding compartment area as Rooms 12341 and 12351 as indicated in Figure 10-15 of the Flooding topic report (Ref. 73) and Figure 23 of the Barrier Matrix (Ref. 104).There were no significant impacts identified earlier from Rooms 12341/12351. From the supporting evidence reviewed, the combination of the pressure relief flapper (12362-AD-D02) and floor drains delivers sub-claim IH-2.5.5 which is in line with EKP.3.

5.6 COMBINED CONSEQUENTIAL HAZARDS

259. Combined events and their associated combined consequential loads have the potential to compromise the safety measures in place against internal flooding, such as barriers.

260. The RP considered a combination of internal hazards postulated to initiate plant level faults. (Internal hazards topic report – combined hazards, Ref. 71). I sampled the initiating events or event combinations that would result in an internal flooding. These are captured on the hazard schedule. The RP argued that these events are bound by the internal flooding hazards.

261. I raised regulatory query RQ-AP1000-1516 (Ref. 43) to seek more clarification of the consideration of water spray, steam generation and wave formation. The RP stated that water spray and steam generation effects are addressed as part of the Environmental Qualification programme. The RP also stated that steam releases are addressed where fluid systems are at 100°C or above. These are considered under the Pressure Part Failure assessment (Section 6 of this assessment report). The RP argued that wave formation within the NI is not identified as a relevant internal hazard effect that impacts on the design basis of the **AP1000** plant (regulatory query RQ-AP1000-1516, Ref. 43).

262. However, I sampled the analysis performed under the UK **AP1000** flooding variations report (Ref. 101). The RP considered:

- the flood height differential between adjacent rooms where a postulated break may cause flood water to pass from one room to another;
- a review of various tanks and their sizes and pressures to determine the potential to cause a possible wave during rupture.

263. The RP identified tanks with 'low potential' for wave formation but it is not clear, particularly in the case of pressurised tanks why this is the case. The evidence to substantiate these arguments was not available. I considered that wave formation as a result of a catastrophic tank failure was inadequately substantiated and was not in line with SAPs SC.4, EHA.1, EHA.6 and EHA.12.

264. However, the RP acknowledged that '*site specific considerations will be required to determine if the maximum probable flood height is bounded by the maximum design probable flood height*' (IAEA Safety Guide NS-G-1.11, Ref. 7 p.24). However, it was not apparent that effects from a wave resulting from a catastrophic tank breach were considered. This example is given in the IAEA Safety Guide NS-G-1.11 (Ref. 7). I refer to the earlier assessment finding CP-AF-AP1000-IH-02 that required the consideration of all credible internal and external hazard combinations (consequential, correlated and independent). Therefore the effects of a wave resulting from a catastrophic tank breach should be fully considered as part of this assessment finding.

5.7 ASSESSMENT FINDINGS

265. During my assessment, one assessment finding was identified for the licensee to take forward in its site-specific safety submissions. This is summarised in Annex 2. The matter does not undermine the generic safety submission.

5.8 MINOR SHORTFALLS

266. There are no minor shortfalls identified from my assessment on internal flooding.

5.9 GDA ISSUE GI-AP1000-IH-02 CONCLUSION

267. The submission provides the requisite information relating to the updated internal flooding safety case. Suitable and sufficient claims have been made and these were generally supported by the requisite arguments and evidence. I identified an internal flooding assessment finding that requires site-specific actions to take forward in site licensing. Further consideration of all postulated events, consequence analysis and adequacy of safety measures is required post-GDA. This includes addressing the internal flooding assessment finding CP-AF-AP1000-IH-05 and the generic assessment findings which are summarised in Annex 2.

268. I am satisfied that during the GDA internal flooding has been subjected to an adequate review and substantiation. To conclude, I recommend that GDA Issue GI-AP10000-IH-02 be closed.

6. ONR ASSESSMENT OF GDA ISSUE GI-AP1000-IH-03 – PRESSURE PART FAILURE

6.1 RESOLUTION PLAN ACTIONS AND DELIVERABLES

269. During GDA Step 4, a GDA issue was raised relevant to identification and substantiation of all nuclear significant pipewhip restraints, barriers and shields claimed for the protection of redundant trains against the effects of pressure part failure (GI-AP1000-IH-03) (ONR's Step 4 internal hazards Assessment report, Internal Hazards Report, Ref. 13). This GDA issue was comprised of the following actions:

- GI-AP1000-IH-03.A1: Identify and substantiate all nuclear significant pipe whip restraints, barriers and shields claimed for the protection of redundant trains against the effects of pressure part failure.
- GI-AP1000-IH-03.A2: Provide the updated safety case that details the identification and substantiation of all claims in relation to Main Steam Isolation Compartments associated with pressure part failure.

270. In order to address the two actions it was prudent for the RP to re-present the claims and arguments for the pressure part failure safety case in the PCSR. These are covered in the specific Topic Report and in the revised PCSR.

271. The RP's Resolution Plan (Ref. 20) identifies specific deliverables associated with the above actions:

- Pressure Part Failure Road Map;
- Main Steam Isolation Valve (MSIV) Summary Report;
- Barrier Matrix;
- Break Location Criteria;
- Pressure Part Failure Topic Report.

272. The MSIV Summary Report and the Break Location Criteria document have been encompassed within the Pressure Part Failure Topic Report (Ref. 86). These are discussed in the sections below.

273. The Pressure Part Failure Road Map (Ref. 175) was initially issued to show the link between the existing claims, arguments and evidence. The document has been superseded by the Pressure Part Failure Topic Report (Ref. 86).

274. In addition to the list above, the RP issued a number of draft documents, multiple revisions of the Topic Report and PCSR, and a number of supporting documents.

275. During this phase of the GDA I raised regulatory queries RQ-AP1000-1303 and RQ-AP1000-1529 (Refs 26 and 44) aiming to seek clarity on the scope of submissions and the timescales given in the Resolution Plan.

276. In the following sub-sections, I will cover the following:

- the RP's safety case on pressure part failure;
- my assessment of GI-AP1000-IH-03.A1 and A2, which includes:
  - assessment of claims and arguments including combined consequential hazards;
  - assessment of analysis methodology and criteria;
  - assessment of substantiation of pressure part failure barriers;

- assessment of substantiation of restraints, shield, guard pipes and relief devices;
- conclusions and assessment findings.

6.2 REQUESTING PARTY'S PRESSURE PART FAILURE SAFETY CASE

277. Key document submissions for pressure part failure are:

- UKP-GW-GL-793, Revision 0D – **AP1000** Pre-Construction Safety Report – Chapter 11 Internal Hazards (Ref. 69);
- UKP-GW-GLR-114, Revision 1– UK **AP1000** Plant Internal Hazards Topic Report – Pressure Part Failure (Ref. 86).

278. The RP considered HE systems (operating pressures greater than 1.896 MPa (g), or operating temperature greater than 93.3°C) and Medium-Energy systems (ME). The HE systems have the potential to result in dynamic and environmental effects that can challenge plant SSCs. Dynamic effects include pipe whip, jet impingement, jet spray, sub-compartment pressurisation, asymmetric pressurisation of large equipment, fluid decompression (for example, water hammer) and environmental effects (temperature, pressure, radiation, humidity, spray wetting including chemistry and submergence).

279. ME systems do not have appreciable thermal or pressure energy levels to create substantial dynamic effects and are therefore limited to environmental consequences.

280. The RP developed design criteria and these are assessed in the sections below.

281. The safety design of the **AP1000** plant for pressure part failure has been achieved by:

- removing pressure part failure events that challenge the tolerability of the plant safety case through the assessment of structural integrity;
- ensuring redundant essential Class 1 SSCs are protected from postulated pressure part failure events through:
  - Class 1 passive barriers;
  - limiting the consequences of the dynamic effects by placing SSCs outside the 'Zone of Influence' (ZOI);
  - Class 1 restraints, shields and guards;
  - Class 1 relief devices.
- providing sufficient redundancy in the design such that the consequences of postulated pressure part failures, coincident with an unrelated single active failure, do not adversely affect the delivery of Category A and post 72-hour Category B safety functions;
- equipment qualification of Class 1 SSCs.

6.2.1 A1000 PRESSURE PART FAILURE CLAIMS

282. The RP made the following claims in the area of pressure part failure (Refs. 69 and 86):

- **Claim IH-3.1**: Pressure part failures are deterministically assumed to occur as a gross failure initiating event, except those justified by the Structural Integrity Classification as Highest Safety Significance.

- **Sub-claim IH-3.1.1**: Piping equal to or less than Diametre Nominal 25 (DN25) is considered bounded by failure of larger piping systems when they are present within the same room or compartment.
- **Sub-claim IH-3.1.2**: Failure of fluid systems of moderate-energy cannot result in pipe whip, jet impingement, compartment pressurisation, or decompression transients.
- **Claim IH-3.2:** Passive protective measures have been incorporated in the **AP1000** design to protect SSCs that deliver Category A and post-72 hour Category B safety functions from pressure part failures.
- **Sub-claim IH-3.2.1**: Consequences of pressure part failure will be contained through the design of barriers or physical separation as to not cause the loss of a Category A or post-72 hour Category B safety function.
- **Sub-claim IH-3.2.1.1**: Class 1 civil/structural relief devices are used to control subcompartment pressure where unmitigated effects may challenge the integrity of claimed Class 1 barriers.
- **Sub-claim IH-3.2.2**: Consequences of pressure part failure will be restricted through the design of shields, restraints, barriers, or physical separation so as to not cause the loss of a Category A or post-72 hour Category B safety function.
- **Claim IH-3.3**: SSCs that deliver required Category A and post-72 hour Category B safety functions will operate in conditions following a pressure part failure event.

283. The RP identified the following SSCs to deliver the above claims:

- Class 1 barriers are the principal protection feature to protect essential SSCs from the dynamic effects of a pressure part failure event (IH-3.2.1);
- Class 1 pipe restraints to restrict the failure consequences (IH-3.2.2);
- Class 1 shields to protect against jet impingement (IH-3.2.2);
- Class 1 guard pipes to protect against the effects of postulated gross failures within or onto Containment penetration (IH-3.2.2);
- Class 1 relief devices to maintain the function of the Class 1 barriers from the effects of compartment pressurisation (IH.3.2.1.1).

6.2.2 ASSESSMENT OF GDA ISSUE GI-AP1000-IH-03

6.2.3 SCOPE OF THE ASSESSMENT

284. The assessment strategy in Section 2 was used to formulate the scope.

285. My assessment covers the deliverables used in addressing the two actions (GI-AP1000-IH-02.A1 and GI-AP1000-IH-02.A2). I assessed the Topic Report and the PCSR on pressure part failure, and I also sampled supporting documents to obtain confidence on the requisite evidence and substantiation of the claims made.

286. The areas chosen to review the pressure part failure case were limited to:

- inside Containment;
- outside Containment – mainly the Auxiliary Building. High level consideration of the Turbine Building was also given.

287. The sections below cover the areas of my assessment.

6.2.4 ASSESSMENT OF CLAIMS AND ARGUMENTS

288. The RP undertook a systematic identification of pressure part failure events and consequences analysis which culminated in the derivation of specific claims, as given above (Chapter 11, Ref. 69 and pressure part failure topic report, Ref. 86).

289. Claims IH-3.1 and sub-claims IH-3.1.1 and IH-3.1.2 are relevant to the methodology and assumptions used in the analysis. These are in line with the relevant good practice established in the UK. Further discussion on the analysis methodology and in particular on claim IH-3.1 is given in the sections below.

290. The **AP1000** plant arrangement is based on segregation of redundant Class 1 SSCs to protect against the dynamic effects of postulated pipe failures. Segregation is achieved by suitably rated reinforced concrete barriers or composite steel structures. These Class 1 barriers are designed to withstand the loads imposed by postulated pressure part failure events (IH-3.2.1).

291. In addition and where there are no barriers, separation of equipment by distance or height using ‘ZOI’ is used. The extent of the ‘ZOI’ is a function of the system process condition and the size of the failure. Redundant Class 1 SSCs are located outside of the pipe whip and jet ‘ZOI’ (IH-3.2.1). Class 1 restraints are used to restrict the ‘ZOI’ (IH-3.2.2).

292. Inside Containment a combination of Class 1 barriers formed by the walls and floors of compartments, and separation by restricting the ZOI is utilised (IH-3.2.1).

293. Outside Containment, the RCA and non-RCA areas of the Auxiliary Building are physically segregated by Class 1 structural walls and floor slabs. These structural barriers are designed to prevent the effects of postulated pressure part failures within one part of the Auxiliary Building from damaging Class 1 SSCs contained in the other part (IH-3.2.1).

294. To restrict the consequences of pressure part failure Class 1 pipe restraints, shields, and guard pipes are used to protect the essential Class 1 SSCs including barriers (IH-3.2.2).

295. All credited pipe restraints, shields and guard pipes are listed in the hazard schedule of the pressure part failure Topic Report (Ref. 86).

296. Class 1 relief devices are used to relieve the overpressure as a result of subcompartment pressurisation to protect the Class 1 barriers (IH.3.2.1.1).

297. Inside Containment the Containment Vessel is claimed for retention of pressurisation effects.

298. Outside Containment, in the Auxiliary Building failure of the main steam or feedwater lines associated with the SGS could pressurise a number of Rooms (for example, 12306, 12404, 12406, 12504, and 12506) and cause damage to the Class 1 barriers. Pressure relief devices are in the form of blow-out panels or in the form of dual-acting passages such as doorways or hatchways. These are listed in the hazard schedule of the Reference 86.

299. To satisfy myself that the RP identified suitable and sufficient safety measures, I sampled the following areas.

Inside Containment

300. Within the Containment a number of HE systems are present including the CVS, PXS, Reactor Coolant System (RCS), Normal Residual Heat Removal System (RNS) and SGS.

301. The RCS and PXS are fully contained within the Containment Vessel. Failure of these pipes could result in pipe whip, jet impingement and over pressurisation within the Containment. The remaining HE systems have components within the Auxiliary Building RCA area (CVS and RNS) and the non-RCA area (SGS).

302. In addition, a number of ME systems are present and these are listed in the hazard schedule of the Topic Report.

303. I sampled the claims made for the Steam Generator (SG) compartments. The SG1 compartment consists of Rooms 11201, 11301, 11401, 11501, and 11601. The SG2 compartment consists of Rooms 11202, 11302, 11402, 11502 and 11602. Failure of multiple HE lines, including the CVS purification pipe, the SG blowdown line, the Main Feed water line or the Start-up Feed water line could lead to pressurisation effects within the SG compartments.

304. The SSCs in the SG compartments have redundant trains, which are segregated between the two SG compartments by Class 1 barriers. The SG compartments have composite concrete-steel walls with a minimum thickness of 0.762m and are separated by the reactor cavity and the refuelling compartment (Pressure part failure topic report, Ref. 86).

305. The RP identified all relevant room barriers (walls, floor and Containment Vessel) against the relevant HE break and listed them in the hazard schedule.

306. In addition, to the Class 1 barriers, Class 1 pipe restraints have been identified in Rooms 11201 (RCS), 11301 (RCS), 11401 (SGS) and 11601 (SGS) for SG1, and in rooms 11402 (SGS) and in 11602 (SGS) for SG2. Furthermore, Class 1 jet shield have been provided in Room 11602 (SGS).

307. Further evidence on segregation of SSCs is provided by the design provision to include the inboard Containment isolation valves in a number of rooms in the Containment. These Containment isolation valves are provided with redundant valves outside Containment in the RCA Auxiliary and non-RCA Auxiliary Buildings (for example, Valve and Piping Penetrations Room 12306), which will remain unaffected by the possible pipe whip, steam release and water spray events inside Containment. This is in line with SAPs EDR.2 and ELO.4. Most of these isolation valves are also fail-close.

308. Further discussion and assessment of the claims and arguments is given below for Rooms 11205 (Case 1) and 11206 (Case 2).

309. Although the hazard schedule lists all relevant barriers, it does not differentiate against the dynamic effect (pipe whip, jet impact or overpressurisation) that the barriers are claimed against.

310. The Barrier Matrix report (Ref. 104) also presents a graphical presentation of the rooms and identifies all relevant barriers (walls and floors). I assessed this document and concluded that the information presented was insufficient in terms of identifying all relevant barriers for each room. In addition, the document did not identify the barriers claimed against pipe whip and jet impact, nor did it list the imposed loads. Therefore, there is a need to update the Barrier Matrix report

(Ref. 104) to provide all barriers (walls, floors and ceilings) against all identified dynamic effects, see CP-AF-AP1000-IH-01.

Outside Containment

311. In the non-RCA section of the Auxiliary Building HE break locations are confined to the MSIV compartments (Rooms 12404/12504 and Rooms 12406/12506), and the Valve/Piping Penetration Room (12306).

312. In the RCA section of the Auxiliary Building HE break locations are confined to CVS pipework located in Rooms 12156, 12255, 12258, and 12259.

313. In the RCA section of the Auxiliary Building a number of ME systems are also present. These are listed in the hazard schedule of the Topic Report.

314. I sampled the claims made in the Valve/Piping Penetration Room 12306. This room contains automatically actuated Containment isolation valves for the VWS, SGS, Passive Containment Cooling System (PCS), DWS, and PCS Passive Containment Cooling Water Storage Tank (PCCWST) recirculation equipment piping and valves. Dynamic effects from the gross failure of piping in these areas are limited to the SGS. The isolation valves are fail safe. Barriers (walls, floor ceilings) and containment penetration guard pipes have been identified and listed in the hazard schedule. However, neither the hazard schedule of the Topic Report nor the Barrier Matrix identified the specific dynamic load requirements that the barriers have been identified to provide protection against, see CP-AF-AP1000-IH-01.

315. With regard to MSIV compartments Rooms 12404/12504 and Rooms 12406/12506, the RP identified and listed in the hazard schedule the barriers (walls, floor and ceiling), pipe whip restraints (in Room 12404), containment penetration guard pipes (in Rooms 12404 and 12406) and relief panels. My assessment of the MSIV compartment is further discussed below.

316. Outside the NI the RP also identified that for a postulated Main Steam line or Main Feed Water failure in the Turbine Building, beyond the ‘First Bay’, the seismically-designed Class 2 barrier (Wall 11.2) is claimed to prevent pipe whip from impacting the Class 1 Wall 11. Wall 11 is the barrier between the Turbine Building and the Auxiliary Building and is the principal mean of protecting Class 1 SSCs from an internal hazard in the ‘First Bay’. First Bay is the section between Wall 11 and Wall 11.2 of the Turbine Building and contains Class 2 defence-in-depth SSCs. Wall 11 is composed of thick concrete that is locally strengthened (thickened) at the steam and feedwater line penetration areas.

317. With regard to claim IH3-3 the RP developed a methodology for environmental equipment qualification (**AP1000** Equipment qualification methodology, Ref. 134). It has also developed core criteria for temperature, pressure, humidity, submergence, radiation, spray and chemistry activities related to the design or analysis of spaces within the **AP1000** plant (**AP1000** environment conditions for equipment qualification, Ref. 92).

318. The RP claimed that all Class 1 SSCs are qualified for abnormal and accident conditions, including the effects of postulated pressure part failure.

319. In addition, the RP identified all essential SSC components where additional qualification requirements are imposed resulting from the dynamic effects of the associated failure above and beyond the core environmental conditions. These components have been identified based on a “Zone of Influence” assessment

using the plant arrangement coupled with an operability assessment to confirm that the SSC is required.

320. Qualification will be completed, by means of design or testing in accordance with governing codes and standards, post GDA and during the detailed design, equipment selection and procurement process.

321. The **AP1000** environment conditions for equipment qualification report (Ref. 92), however states that the information is applicable to the **AP1000** standard plant design, with the exception of EPS (European Standard Plants) and UKP (United Kingdom Standard Plants). These plants are excluded on the basis that their unique design will differ, for radiological and thermal conditions, from the standard design for the **AP1000** plant Auxiliary Building. I queried this with the RP (Regulatory query RQ-AP1000-1795, Ref. 60), who proactively identified that the environmental conditions report required updating and has captured this in the Design Reference Point (Ref. 64). This aspect is also discussed in section 5 of this assessment report.

322. Overall, I am satisfied that the RP identified suitable and sufficient claims supported by suitable arguments. The safety measures against the postulated effects of pressure part failure have been adequately identified and captured within the hazard schedule. This is in line with ONR’s SAP EKP.5, ECS.2 and ESS.1.

6.2.5 COMBINED CONSEQUENTIAL HAZARDS

323. Combined events and their associated combined consequential loads have the potential to compromise the safety measures in place against pressure part failure such as barriers.

324. The RP proactively undertook a study to identify all credible consequential, correlated or independent hazards relevant to pressure part failure (Combined hazards topic report, Ref. 71).

325. Pressure part failure may be a source of internal flooding only, or a source of both internal flooding and internal missiles. The consequences of these events are assessed in Sections 5 and 8 of this report.

326. The RP did not identify any credible combinations of pressure part failure events in the combined hazard schedule.

327. The RP overall concluded that the Category A safety functions will continue to be delivered following various design basis combined hazards.

328. Assessment Finding CP-AF-AP1000-IH-02 was raised in Section 4 above which is also applicable here.

6.3 ASSESSMENT OF ANALYSIS METHODOLOGY AND CRITERIA

329. The RP’s analysis methodology consists of the following steps (Pressure part failure topic report, Ref. 86):

- identification of hazard sources;
- evaluation of the indirect effects of the pipe break (dynamic effects, environmental effects and flooding);
- evaluation of the consequences of the indirect effects on essential Class 1 equipment;
- identification of protective measures (barriers, restraints, shields, guard pipes and relief panels).

330. The above approach is in line with my expectations and ONR’s SAPs EHA.1 and EHA.6.

331. The RP submitted its design criteria for pipe rupture (Pipe rupture criteria for **AP1000** plant, Ref. 135). The design criteria included the use of partial failures such as Leak before Break (LBB) and Break Exclusion Zone (BEZ) to HE lines.

332. These criteria, however, are not in line with my expectations and the relevant good practice established in the UK. In the area of Structural Integrity classification a key element is the assumption of gross failure irrespective of any anticipated material properties and crack development behaviour. Therefore, in the area of structural integrity and fault studies gross failure has been assumed. This was not the case for the internal hazards area.

333. I raised regulatory query RQ-AP1000-1378 (Ref. 32) to seek clarity on the criteria used in the analysis including classification of restraints, time at risk arguments, on the conservatism of the analysis, and whether any sensitivity analysis has been undertaken.

334. The RP conceded that the internal hazards pressure part failure has not considered gross failure for a number of selected pipe lines. These selected ASME Section III pipes have been subjected to supplemental mechanistic and deterministic design criteria requirements to reduce the failure probability. These lines have been classified as Low Probability Design Basis Events (DBL). The potential consequences of pressure part failure from these lines could be significant. Therefore, the unmitigated effects of gross failure required assessment.

335. Given the scope of this shortfall, ONR also formally wrote to the RP to express its concerns (Ref. 136).

336. A workshop took place in January 2016 to reach convergence on the technical approach to be taken regarding the assessment of the DBL lines.

337. The RP proposed to revise its assessment criteria for pipe rupture and to align the assessment undertaken within Structural Integrity, Fault Studies and Internal Hazards areas. Gross failure will be assumed on all pressure retaining systems where an Incredibility of Failure (IoF) claim is not applicable (IH-3.1).

338. For the HE systems, gross failure in the form of a double-ended guillotine break is assumed for those lines not classified as High Safety Significance (HSS). Arbitrary intermediate breaks have also been added.

339. For the ME systems, through-wall cracks and crack exclusion have been removed and replaced by gross failure.

340. The above proposal was in line with my expectations.

341. It was also agreed that, given the scope of the task, the RP was to propose detailed analysis of a limited number of examples. The limited number of examples will also serve as the foundation in determining the ‘risk gap’ between the total population of the gross failure assessment and the examples and to demonstrate a suitable application of the pressure part failure assessment process.

342. Subsequently the RP submitted the following:

- a summary of the **AP1000** design pressure part failure methodology for mechanistically and deterministically designed piping (Ref. 137);

- the **AP1000** plant pressure part failure assumptions (Ref. 138);
- identification of assessment examples (Ref. 139).

343. I assessed the above submissions and raised regulatory query RQ-AP1000-1549 (Ref. 45). The aim of this RQ was to:

- understand the proposal to retain the LBB criteria within the overall assessment;
- whether consideration was given to single failure criterion, intermediate break locations, longitudinal splits, benefits and detriments of the use of restraints, consequential propagation criteria, cliff edge effects, and on ALARP consideration;
- examine the criteria used for the selection of the three representative examples;
- explicitly identify all remaining affected lines, not previously considered in terms of gross failure, and present the 'risk gap' between the total population of gross failure assessments and the examples identified.

344. The RP developed a 'risk matrix' for the total population of mechanistically and deterministically design piping, which was based on an expert panel's subjective ranking of the 'total risk'. The latter was not supported by sufficient arguments and evidence. A number of systems have been bounded by the three representative events.

345. The RP in response to RQ-AP1000-1549 did not provide the requisite clarity on the methodology and criteria used to identify the three representative examples. Similarly the justification of the remaining "risk gap" was not robust.

346. The RP submitted further documentation which provided some further insight (Refs 140 and 141) to support regulatory query RQ-AP1000-1549.

347. I considered these references and issued regulatory query RQ-AP1000-1585 (Ref. 46) to gain more clarity on the following:

- details on the methodology and criteria used for the selection of the representative examples.
- the expert panel terms of reference and evidence used;
- how the expert panel determined that the remaining pipe lines represent a low 'risk gap'.

348. I also wrote to the RP (Ref. 142) expressing my lack of confidence in demonstrating that the current design reduces the pressure part failures risk to So Far As Is Reasonably Practicable (SFAIRP). I requested the RP to provide further evidence to demonstrate that no major design changes (for example, compartment size and barrier strengthening) will be necessary as a result of addressing the remaining DBL lines post GDA.

349. The RP re-issued a report on 'Supplementary pressure part failure analysis cases for GDA' (Ref. 141 which documented the methodology and outcome of the expert panel in the evaluation of the mechanistically and deterministically designed piping system. A supporting analysis document provided the following (Supplemental gross failure analysis cases, Ref. 145):

- the expert panel terms of reference and review process and criteria.
- a summary of the mechanistically and deterministically designed piping systems and affected rooms within Containment and in the Auxiliary Building.
- a qualitative risk assessment matrix for the evaluation of the consequences of a postulated gross failure event. This included the probability of plant

performance against the consequences associated with the gross failure event. The analysis presented was based on qualitative ranking by the expert panel.
- identification and justification of gross failure events DBL that represent the greatest risk in the **AP1000** plant design for pressure part failure.

350. The selection of the three examples was based on systems and location which presented the highest risk for a given dynamic effect. These are as follows:

- Case 1 – Large Break LOCA of the Reactor Coolant System (RCS) – Cold Leg Break in the reactor vessel nozzle area in Room 11205;
- Case 2 – Direct Vessel Injection Line (DVI) failure in Room 11206;
- Case 3 – Main Steam Line failure in Room 12404/12504 - MSIV B compartment.

351. The RP concluded the following:

- the 'risk gap' has been minimised through a review of postulated break events and locations that represent the greatest challenge to the **AP1000** plant safety case.
- the three cases do not pose a risk of a major design change;
- the remaining population of postulated failures are largely bounded by the above three cases;
- due to timescales and the level of analysis involved, the implementation of the pressure part failure evaluation for the remaining lines will be completed post GDA;
- minor modification may be required as a result of addressing the remaining pipe lines.

352. References 140 and 141 have subsequently been captured within the pressure part failure Topic Report (Ref. 86).

353. The Topic Report presents the deterministic safety case for pressure-retaining systems and components within the **AP1000** plant. This includes infrequent Design Basis Events (DB1), Frequent Design Basis Events (DB2) and DBL events. This is in line with ONRS's SAPs EHA.3.

354. The Topic Report also presents the consequences analysis undertaken for the three representative examples, given above. These are discussed below:

Case 1 - Large Break LOCA - RCS Cold Leg Break in Room 11205 - Reactor Vessel Nozzle Area

355. Room 11205 is a significantly challenging volume within the **AP1000** design as it presents the largest compartment pressurisation potential coupled with pipe whip and jet effects from the primary coolant system. Additionally, consistent with GDA Issue FD-02, the rapid decompression of the reactor due to a primary piping gross failure and the effect on fuel performance has not been previously analysed. Therefore, a primary coolant loop failure within Room 11205 presents a significant challenge for multiple indirect effect mechanisms.

356. The RP assessed the consequences of the dynamic and environmental effects and concluded that the plant response to the postulated gross failure of a RCS cold leg is acceptable and no major design changes are required. A brief summary is given below:

- Pipe whip impact on the CA01 module will result in plastic deformation of both the cold leg pipe material and the CA01 module at the point of impact. Using Finite Element Analysis (FEA) it was demonstrated that the overall integrity of the structure is unaffected.

- Flooding of Room 11205 will occur as a direct result of the LOCA event. Management of flood height and effective cooling water availability is controlled by barriers within the Containment.
- Jet impingement and thrust reaction. The consequences of these indirect effects are the horizontal displacement of the RV and the jet impingement of the DVI line piping.
  - The reactor vessel supports are expected to exceed minimum elastic stress allowable limits, but remain functional as a result of the failure; i.e. minor deformation will occur.
  - The reactor vessel and the remainder of the RCS are adequately restrained and remain intact.
  - The DVI lines are susceptible to high stresses outside of Room 11205, however, their injection ability is not compromised using linear modelling techniques.
  - The failure of the RCS cold leg will not adversely affect the plant safety case as the post-failure RCS conditions and Class 1 injection capability are retained.
- Fluid decompression affects the reactor vessel internal and fuel assemblies. This is outside the scope of my assessment, but it was evaluated as part of GI-AP1000-FD-02.
- Asymmetric pressure effects have been included in the jet impingement analysis.
- Sub-compartment pressurisation. Room 11205 consists of walls formed by CA04 and CA01 modules and of ceiling CA31. The latter presents the weakest link with regard to this room. Failure of CA31 will terminate pressurisation of Room 11205. The RP concluded that given the various penetrations (RCS loop penetrations, DVI line ‘A’ penetrations and reactor vessel cavity shield door) the consequences are acceptable.
- Class 1 SSCs are qualified for operation in a post-accident environment.
- Restraints are not considered in this scenario, as the impact of the resulting pipe whip and jet does not unduly challenge the claimed barriers.
- All relevant claims have been captured in the hazard schedule.

Case 2 – PXS Direct Vessel Injection (DVI) Line failure in - in Room 11206 - PXS ‘A’compartment

357. Room 11206 contains valves and pipework associated with the DVI pipe line ‘A’. Failure of the DVI piping within Room 11206 presents a challenge to structural integrity by means of subcompartment pressurisation, which is a critical parameter in maintaining long-term core cooling. Furthermore, the in-containment Refuelling Water Storage Tank (IRWST) squib valves are credited in the plant safety analysis and therefore their function must be maintained considering the pipe whip and jet effects from an 8-inch Nominal Pipe Size (NPS) primary pipe break.

358. The RP assessed the consequences of the dynamic and environmental effects and concluded that no one effect, or combination of these effects, will result in the loss of a Class 1 SSC whose function supports the results of the direct effects analysis. A brief summary is given below:

- Pipe whip impact. The PXS Room walls and ceiling are claimed as barriers for the effects of pipe whip in Room 11206. However, failure does not result in an impact to a structure.
- Pipe whip will render the Class 1 SSC within the PXS ‘A’ room unavailable (for example, IRWST injection squib valve PXS-V125A). The RP provided qualitative arguments as to why the consequential loss of the IRWST injection or containment recirculation squib valve function in Room 11206 is acceptable.

- Jet impingement was qualitatively discussed. The conclusion was that the jet impingement effects resulting from the failure of the DVI ‘A’ line in the PXS ‘A’ room would not adversely affect the plant safety case.
- Fluid decompression affects the reactor vessel internal and fuel assemblies. This is outside the scope of my assessment, but it was evaluated as part of GI-AP1000-FD-02.
- Asymmetric pressurisation. Components within the PXS rooms have not been considered for asymmetric pressurisation due to their size and the resulting pressurisation rate of the volume.
- Subcompartment pressurisation was analysed in terms of differential pressure across the barrier using droplet models. The RP concluded that the communication between Room 11206 and the Containment should limit the dynamic differential pressure across structural barriers.
- Class 1 SSCs are qualified for operation in a post-accident environment.
- The addition of restraints is not considered practical for this application.
- All relevant claims have been captured in the hazard schedule.

Case 3 Main Steam Line failure in Room 12404/12504, MSIV B compartment – GI-AP1000-IH-03.A2

359. The MSIV compartments represent a challenging volume within the Auxiliary Building, outside of Containment. The limiting failures are gross failure events of the Main Steam or Main Feedwater pipes, which will result in excessive compartment pressure within close proximity to the MCR and Safety Class 1 Instrumentation and Control (I&C) functions.

360. MSIV compartment ‘B’ was chosen over compartment ‘A’ as the latter compartment is larger and does not have the layout constraints as the MSIV ‘B’.

361. The RP assessed the consequences of the dynamic and environmental effects and concluded that no one effect, or the combination of these effects, will result in the loss of a Class 1 SSC whose function supports the results of the direct effects analysis. A brief summary is given below:

- Pipe whip. Main Steam lines were assessed as not posing a pipe whip concern. Main Feedwater line poses a risk of structural impact to the floor and east wall of the respective MSIV compartment. This may cause loss of the MCR and potentially loss of one division of Class 1 C&I. Depending on the magnitude of the floor failure damage to other Class 1 C&I divisions may take place as the intervening barriers are designed only against fire. The RP identified and claimed five pipe whip restraints to protect the structures.
- Jet impingement and thrust. The failure of either the Main Steam or Main Feedwater line in the MSIV ‘B’ compartment will not adversely affect the plant safety case as the pipes are adequately restrained or experience limited displacement.
- Sub-compartment pressurisation. Based on the barrier thickness, the RP assumed that the compartment ceiling is the likely candidate for failure that would release mass and energy out of the Auxiliary Building into the atmosphere. The RP also identified the potential that subcompartment pressure may challenge the floor of the room due to differences in reinforcement. A detailed stress analysis of the MSIV ‘B’ compartment floor was performed to determine the floor response to the Main Steam line pressure profile. The RP concluded that although significant pressurisation of the compartment would occur it would not affect the barriers claimed for protection of the operators and Class 1 I&C equipment. The **AP1000** plant structure is designed to 6.5 psig based on a reduced area break. The RP made a claim on Class 1 relief panels

as well as doorways to protect the barriers. These devices will open at a pressure of no greater than 4.5 psig.

- All barriers, pipe whip restraints, containment penetrations guard pipes and relief panels are listed in the hazard schedule.
- Environmental effects. Class 1 SSCs are qualified for operation in post-accident environments to ensure that their credited safety functions are maintained. The increased pressure of a gross failure will not affect the limiting environmental profile. The effects of jets impinging within the MSIV compartments, however, will need to be considered in the qualification of the MSIVs. The RP raised a corrective action.

362. While I was broadly satisfied with the selection of the three cases identified above, I needed to obtain confidence on the following:

- that the three representative cases largely bound the remaining lines listed in the Topic Report;
- that the remaining "risk gap" has been minimised as far as is reasonably practicable;
- that any future analysis will not result in plant layout modifications or major system modifications.

363. I subjected the Topic Report (Refs 83 and 84 and 85) into a detailed assessment and raised regulatory queries RQ-AP10001677, RQ-AP1000-1702 and RQ-AP1000-1795 (Refs. 55, 56 and 60). The aim of my queries was to gain clarity and improvement on the following:

- overall cohesiveness and coherence of the safety case (in line with ONR SAPs SC.4);
- analysis assumptions (for example, intermediate break location and time at risk arguments for the RNS system);
- analysis assumptions for the DB1 and DB2 events;
- the qualitative discussions of pipe behaviour for the three cases;
- identification and selection of safety measures and claims made (barriers, restraints, shields and relief panels, in line with ONR SAPs EKP.5);
- calculated consequential loads and civil structures design criteria (in line with ONR SAPs ECE.1);
- substantiation of the claims made and the link to the Barrier Matrix;
- domino effects;
- equipment qualification against environmental criteria (in line with ONR SAPs EQU.1);
- on the identified design modifications;
- suitability and sufficiency of the 'risk matrix' and 'risk gap'; representative scenarios;
- to understand the full scope of post GDA work and its implication on the current plant design layout.

364. With regard to DB1 and DB2 events, the Topic Report refers to References 143 and 144 – Pipe rupture hazard analysis for the auxiliary building and containment building respectively. These documents capture the analysis for DB1 and DB2 events. However, the analysis presented in these documents is not aligned with the revised analysis methodology as they include partial failures. These references will require updating post GDA to reflect the revised analysis criteria.

365. The RP's analysis used stress and fatigue-based criteria for terminal end and intermediate pipe breaks. I challenged the RP on the selection of intermediate break locations and in particular on the potential vulnerabilities presented by corrosion and erosion mechanisms. In addition, and irrespective of structural integrity considerations, my expectation was that the intermediate break

locations should take cognisance of whether SSCs are present in close proximity to HE lines. The RP indicated that in HE systems, the consideration of new or revised intermediate breaks will result in revised dynamic responses which may require minor design changes (see CP-AF-AP1000-IH-06).

366. The RNS is a HE system that has been assessed as ME in the RCA side of the Auxiliary Building due to consideration of its frequency of use limits. The RP assumed that systems or portions of the systems that do not exceed the HE threshold for either 98% of its total operating time or 99% of the plant operating life are considered as ME.

367. Downgrading the RNS system to an ME energy system is not in line with my expectations and ONR's SAPs NT.2. I raised regulatory queries RQ-AP1000-1378, RQ-AP1000-1677 and RQ-AP1000-1795 to understand the potential consequences of the RNS failure with the aim of the RP providing sufficient protection (Refs 32, 54 and 60).

368. Under DCP EPS-GW-GEE-001, the RNS suction line from the RCS branches inside the Containment into two parallel trains (this DCP is incorporated within **AP1000** Design Reference Point, Ref. 64). Each train has two RCS isolation valves inside Containment, one of which is a containment isolation valve. The two RNS suction lines are separated and have separate Containment penetrations, separate outside Containment isolation valves, and separate piping to their corresponding RNS pump. The RNS pumps are located at elevation 89.79 m with each pump train provided in a separate rooms: 12162 for RNS 'A' and 12163 for RNS 'B'.

369. The RP provided the following supporting information in response to Reference 60:

- Failure of the RNS piping in the Auxiliary Building in an HE mode can occur outside or within the CA20 structural module, in and around the RNS pumps and RNS heat exchangers.
- The direct effects of the postulated failure of the RNS in the Auxiliary Building consistute a Design-Basis Event and that is terminated automatically by Class 1 SSCs, consistent with loss of coolant accidents involving RNS.
- The indirect effects of the postulated HE failure of the RNS in the Auxiliary Building include pipe whip, jet effects, fluid decompression, asymmetric pressurisation, and sub-compartment pressurisation.
- As redundant Class 1 SSCs required to terminate the associated event are located within the Containment, there is sufficient confidence that pipe whip effects within RNS equipment rooms and SSC proximity to the RNS piping will not adversely affect the **AP1000** plant response to this failure.
- Evaluation of the failure of the RNS in a HE state will confirm the adequacy of the barriers that comprise the Spent Fuel Pool and its available coolant inventory.

370. An assessment of the consequences of the RNS system is captured in CP-AF-AP1000-IH-06.

371. The analysis undertaken for the three cases is based, to some degree, on qualitative discussions on pipe behaviour which was then was used to define the consequential dynamic effect and impact on structures or SSC. The current analysis should be supported by further analysis and modelling to characterise the pipe behaviour and the consequences to barriers and SSCs. This is captured in assessment finding CP-AF-AP1000-IH-06 (discussed below).

372. In response to my RQs the RP responded positively and in a timely manner to all my queries and has updated the pressure part failure Topic Report (Ref. 86).

373. Overall, the updated Topic Report has provided much needed clarity on the cohesiveness of the safety case, the claims made, the safety measures in place and on the remaining risk gap.

374. A number of issues remain outstanding which will require addressing post GDA in an assessment finding as given below, see CP-AF-AP1000-IH-06.

6.4 OUTSTANDING ISSUES

Implementation of changes to assessment criteria

375. As part of the implementation of changes to the assessment criteria, the RP identified the following work to be undertaken post GDA (Nuclear site licensing task summary for **AP1000** plant pressure part failure assessment process, Ref. 148):

- ME Systems:
  - identify rooms/ compartments with new spray wetting or flooding conditions.
- HE Systems:
  - identify discrete break locations using stress and fatigue criteria;
  - perform indirect consequences analysis for each discrete break location; pipe whip, jet spray, fluid decompression transient analysis, subcompartment and asymmetric pressurisation and interface with flooding assessments;
  - identify affected Class 1 SSCs and determine the need for protection.
  - perform jet impingement calculations;
  - generate calculations for interfacing disciplines (input calculations to piping analysis, civil/ structural and to miscellaneous commodities.

Environmental requirements

- Reconcile modified design requirements into UK **AP1000** plant environmental requirements for design and qualification.

376. In addition, the following require addressing:

- Assessment of the consequences of the RNS system as an HE system.
- Further justification and analysis to support the qualitative discussions presented in the three cases.
- Assessment of intermediate break locations for failure. This should allow for failure mechanisms other than due to stress and fatigue criteria such as erosion and corrosion. Also consideration should be given to plant layout and whether a particular location should be assessed because essential SSCs are located in close proximity such that they could be impacted by a whipping pipe.

CP-AF-AP1000-IH-06 – The licensee shall complete the pressure part failure assessment based on gross failure to quantitatively characterise the total population of Medium Energy and High Energy systems and for all Design Basis Events. This shall include:

- **Identification and assessment of additional intermediate break locations due to stress and fatigue, erosion and corrosion, and where SSCs are in close vicinity of an HE system, assess the potential consequences.**
- **The prediction of pipe behaviour and the consequential dynamic effects and impact on structures and SSCs shall be supported by appropriate modelling.**
- **Evaluation of the consequences of analysing the RNS system as a High Energy system, and therefore evaluating the consequence of gross failure.**

377. The RP gave ONR assurance that they are confident that this assessment finding should not lead to major design modifications on the basis of their expert panel review.

378. With regard to vessel pressure part failure consequences, Reference 86 indicated that the structural integrity of the reactor vessel, pressuriser, SGs, reactor coolant loop piping, RCPs, PRHR heat exchangers, Core Make Up Tank (CMT), and accumulator are substantiated in Chapter 20 of the PCSR (Ref. 188). The reactor vessel, steam generator and the pressuriser are classified as HSS and are assumed not to fail for the purposes of the internal hazards assessment (claim IH-3.1). It also stated that failure of these components within the scope of pressure part failure is not deemed credible because the pressuriser is fitted with safety valves. Furthermore, rupture of the PRHR system is also not a credible jet impingement hazard, because the heat exchanger is normally submerged and located within the IRWST system.

379. The use of safety valves as a means of preventing component failure does not align with the structural integrity assessment contained in Chapter 20 of the PCSR (Ref. 188). Furthermore, the comment is not supported within Reference 86 by the claims, arguments, and evidence. The RP concurred that the statement is misleading and should be removed in the future to prevent confusion during licensing. Identification of the issue has been entered into the RP's corrective action process (CAPAL 100458138) with a commitment that the misleading statement be removed from the pressure part failure topic report (Ref. 86), the structural integrity classification report (Ref. 189), and the PCSR (Ref. 188).

380. The lack of unmitigated quantitative analysis of all applicable indirect consequences is not in line with my expectations and ONR SAPs EHA.6. Therefore, there is a need to quantify the unmitigated consequences of pressure part failure.

381. The effects of indirect consequences of pressure part failure on the safety classification was also assessed by the ONR structural integrity discipline during Step 4 of the GDA and the following assessment finding was raised (Step 4 Structural Integrity Assessment of the Westinghouse **AP1000** Reactor, Ref. 190).

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

382. My expectation is that through this finding all potential indirect consequences of pressure part failure should be quantitatively analysed.

383. The ONR Step 4 Structural Integrity assessment of the Westinghouse **AP1000** reactor (Ref. 190) further assessed this aspect.

384. Overall, I am satisfied that the RPs revised analysis criteria are in line with my expectations and relevant good practice established in the UK. I am also satisfied myself that the process used in the identification of postulated events, consequences analysis and identification of safety measures is in line with my expectations.

385. I acknowledge, however, that due to the implications of the revised analysis criteria, the completion of the entire scope of the analysis will be undertaken post GDA during the licensing stage. The RP provided reasonable confidence, based on the information available, in demonstrating that the full implementation of the revised analysis criteria would not result in major design modifications including plant layout changes.

6.5 SUBSTANTIATION OF THE CLAIMS

386. The hazard schedule of the Topic Report identified all claims made for each room considered. The sections below discuss the evidence provided to support the claims.

6.5.1 PRESSURE PART FAILURE BARRIERS

387. Largely the requisite evidence for the substantiation of all the barriers claimed against the dynamic effects of pressure part failure (pipe whip, jet impact and overpressurisation) has not been captured in References 69 and 86. Some analysis, however, has been presented for Case 1 (pipe whip impact on the CA01 module) and Case 3 (subcompartment pressurisation impact on barriers) discussed above.

388. I articulated my expectations to the RP in a number of interactions. The RP indicated that the information is generally available, but not necessarily consolidated in one document.

389. In order to obtain confidence that the RP has a suitable process in place to capture the requirements imposed by the dynamic effects of pressure part failure and reflect them in the relevant design criteria of civil structures, I sampled PCSR Chapter 16 (Ref. 70). I also requested the RP to provide examples to demonstrate that the barriers are sufficiently substantiated.

390. In response the RP provided two examples (Examples of **AP1000** Plant Pressure Part Failure Load Combination for DB1/DB2 Events, Ref. 149):

- Room 12258– Degasifier Column Room (including adjoining Room 12259);
- Room11403 – Pressurised Spry Valve Room (including adjoining Room 114000 – Maintenance Mezzanine).

391. In these examples the RP presented a summary of the outcome of the substantiation analysis. This included:

- the gross failure event and unmitigated consequences;
- the gross failure loads (pipewhip and thrust, jet, pressurisation and flooding loads);
- identifying the source documents where the loads generated by the hazards are evaluated;
- evaluate the capacity of each barrier (walls, floors and ceilings) against the loads generated (from the internal hazard) – a utilisation factor has been calculated;
- identifying and evaluating restraints;
- ensuring that sufficient margins have been reported.

392. I raised regulatory query RQ-AP1000-1786 (Ref. 57) to obtain an understanding of how the internal hazards requirements have been captured within the civil design criteria and how the demonstration of suitability of the barriers against the imposed loads has been achieved.

393. In response, the RP explained that all hazards are addressed in the civil engineering design, either directly by applied loading within design load combinations, or indirectly through engineering analyses performed to demonstrate that the building structures are not loaded or affected by a particular internal hazard.

394. The RP also explained that the civil engineering design explicitly accounts for internal flooding, fire, and pressure part failure in the load combinations considered. Structures acting as barriers to protect against the consequences of internal missiles, dropped loads and/or explosions are shown to be adequate by engineering analyses on a case-by-case basis, as those events are identified to demonstrate adequate hazard protection.

395. I liaised with ONR's civil engineering inspector in the assessment of this supporting document (Ref. 146). While the source documents were not requested for our assessment, the level of margins available will allay any major concerns.

396. It should also be stated here that the ONR Step 4 civil engineering assessment report for **AP1000** GDA (Ref. 14) raised assessment finding AF-AP1000-CE-09, which is relevant to the barriers: "*The licensee shall take account of any implications of the outcomes of the internal hazards GDA issues which could affect the design of civil structures, particularly the loads, load combinations and serviceability requirements applied in the design*."

397. This finding is relevant to all internal hazards requirements.

398. In addition to the above, the impact on penetrations in Class 1 barriers has not been explicitly addressed and will require addressing.

CP-AF-AP1000-IH-07 – The licensee shall justify the detailed design of all penetrations on Class 1 barriers against the potential consequences of pressure part failure.

6.5.2 RESTRAINTS, SHIELDS AND GUARD PIPES

399. The design of Class 1 restraints is given in Reference 150. The design refers to ANSI/AISC N-690-94 – Specification for the design, fabrication, and erection of steel safety-related structures and structural elements for nuclear facilities. However, this version of the standard is outdated.

400. The RP developed an equivalence/maturing study of the US codes and standards which included a review of the ANSI/AISC N-690 – 1994 (Ref. 151). It concluded that "*AISC N-690 (1994) is almost certainly the most widely used hot-rolled steel design code within the UK nuclear industry and is essentially an international code. Generally, its provisions, as applied to the **AP1000** design, are considered to provide consistent designs to BS 5950 and UK best practice. Although N690 went through a major reorganization and revision to analysis and design methodologies between the 1994 and 2006 editions, the later code edition does not have a significant effect on the current nuclear island design. The use of the 1994 edition of AISC N690 remains thus adequate*."

401. However, ANSI/AISC N-690 2012 in conjunction with ANSI/AISC N690s1-15 has replaced ANSI/AISC N690-06, whereas BS5950 has been superseded by BS EN 1993 and withdrawn. Therefore, there is a need for the RP to undertake a gap analysis of the ANSI/AISC N-690-94 against modern standards (see assessment finding CP-AF-AP1000-IH-08 below).

402. The RP submitted an example of a new jet shield analysis and qualification (Ref. 152). The design of Class 1 shields is also based on ANSI/AISC N-690-94 (see CP-AF-AP1000-IH-08).

403. The Class 1 guard pipes are also examples of shields. Guard pipes in the Containment annulus area are designed according to the rules of ASME III Division 1, subsection NE, Class MC. The RP provided some limited information on the design substantiation of the guard pipes and identified that a number of them (for example, P44/P45) will be further assessed post GDA and on completion of the implementation of change to analysis criteria (Assessment query RQ-AP1000-1795, Ref. 60).

6.5.3 RELIEF DEVICES

404. There are two types of relief devices: blow-out panels and dual-acting passages such as doorways or hatchways. Their location is listed in the hazard schedule of the Topic Report.

405. The selection of the type of assembly used is based on the plant arrangement and the minimisation of penetrations in Class 1 barriers.

406. I sampled the roof relief panels for the MSIV compartment as given in Reference 153. This reference states that the relief panels shall be designed and manufactured in accordance with the requirements of ANSI/AISC N690-1994. These shall be designed to release at a pressure of between 2.5 and 4.5 psig to prevent the Auxiliary Building from reaching an internal pressure of 6 psig (see CP-AF-AP1000-IH-08).

407. The RP also provided the design specification for a dual-acting access door in east MSIV (12504-AD-D01) and west MSIV (12506-AD-D01 and 12306-AD-D01). These doors are custom designed built to print assemblies that are designed and procured using Class 1 standards. The RP indicated that the licensee will be required to use design codes that would include ANSI/AISC N-690-94 (see CP-AF-AP1000-IH-08).

CP-AF-AP1000-IH-08 – The licensee shall substantiate the adequacy of restraints, jet shields and relief panels against modern standards.

408. Overall, I am satisfied that the RP has a suitable process to substantiate the claims made and provided me with sufficient evidence. Completion of the full scope of substantiation, however, will be completed post GDA and after completion of assessment finding CP-AF-AP1000-IH-08.

6.6 ASSESSMENT FINDINGS

409. During my assessment three items were identified for a licensee to take forward in their site-specific safety submissions. Details of these are contained in Annex 2.

410. These matters do not undermine the generic safety submission.

6.7 MINOR SHORTFALLS

411. There are no minor shortfalls identified from my assessment of pressure part failure.

6.8 GDA ISSUE GI-AP1000-IH-03 CONCLUSION

412. The RP revised its pressure part failure design criteria to reflect my expectations of relevant good practice, and also to align with the assessment criteria used in structural integrity and fault studies areas.

413. The revised design criteria resulted in an increase in the scope of the analysis, the full implementation of which can only be completed post GDA and during detailed analysis.

414. Reasonable qualitative arguments have been provided on the selection of the three representative examples and on the risk gap presented by addressing the remaining lines post GDA.

415. Based also on qualitative arguments sufficient confidence can be drawn that future analysis should not result in major design modifications.

416. I am broadly satisfied with the work presented as the submissions provide information relating to the process and methodology used in the identification of pressure part failure events, characterisations of the consequences and identification of safety measures. Suitable and sufficient claims have been made and these were supported by reasonable arguments and evidence. However, further consideration of all postulated events, evaluation of the consequences analysis and the adequacy of safety measures is required post GDA, and as a result of addressing assessment finding CP-AF-AP1000-IH-06.

417. I am, therefore, satisfied that GDA issue GI-AP1000-IH-03 can be closed.

7. ONR ASSESSMENT OF GDA ISSUE GI-AP1000-IH-04 – INTERNAL EXPLOSION

7.1 RESOLUTION PLAN ACTIONS AND DELIVERABLES

418. In GDA Step 4, GDA Issue GI-AP1000-IH-04 was raised requiring the RP to substantiate claims and arguments in the area of internal explosion specifically associated with hydrogen generation in the battery rooms and the distribution of hydrogen within areas containing Class 1 SSCs (ONR's Step 4 Internal Hazards Assessment Report, Ref. 13).

419. The key finding was that the PCSR had not adequately presented a multi-legged argument associated with the systems in place to prevent, protect and mitigate against the potential consequences of an internal explosion (ONR's Step 4 Internal Hazards Assessment Report, Ref. 13). Two actions to address the issue were identified:

- GI-AP1000-IH-04.A1, which required substantiation of the safety case for explosion within the Battery Rooms;
- GI-AP1000-IH-04.A2, which required substantiation of the safety case for the routing of the hydrogen pipework within areas containing Class 1 SSCs.

420. In order to address the two actions it was prudent for the RP to re-present the claims and arguments for internal explosions in the PCSR. These are covered in the specific Topic Report and in the revised PCSR.

421. The RP's Resolution Plan (Ref. 21) identifies specific deliverables associated with the above actions:

- Internal Explosion Roadmap;
- Battery Room Assessment;
- Hydrogen Pipeline Assessment;
- Hydrogen Management Assessment;
- Internal Explosions Topic Report;
- PCSR Chapter 11.

422. I raised regulatory queries RQ-AP1000-1304, RQ-AP1000-1439, RQ-AP1000-1642, RQ-AP1000-1506 and RQ-AP1000-1529 aiming to seek clarity on the scope of the submissions and the timescales given in the Resolution Plan (Refs. 27, 36, 40, 44 and 52).

423. In the following sub-sections, I cover the following:

- the RP's safety case on internal explosions;
- my assessment of GI-AP1000-IH-04, which includes the following:
    - assessment of claims, arguments and evidence including combined consequential hazards;
    - substantiation of the claims within the battery rooms (GI-AP1000-IH-04.A1);
    - substantiation of the claims for the routing of hydrogen pipework within areas containing Class 1 SSCs (GI-AP1000-IH-04.A2).
- conclusions and assessment findings.

7.2 REQUESTING PARTY'S SAFETY CASE

424. The RP claims that a postulated internal explosion within the design basis does not prevent the delivery of the Category A safety functions and the supporting post-72 hour Category B safety functions necessary to respond to the postulated

event. The internal explosion safety case relies on preventative measures to ensure that Class 1 SSCs delivering Category A functions are not exposed to an explosion hazard. The key measure is inventory control and therefore the management of flammable atmospheres. Passive protective measures have also been incorporated in the **AP1000** design. These are discussed in detail below.

425. Key document submissions for internal explosions are:

- UKP-GW-GL-793, Revision 0D – **AP1000** Pre-Construction Safety Report – Chapter 11 Internal Hazards (Ref.69);
- UKP-GW-GLR-109, Revision 1 – UK **AP1000** Internal Hazards Topic Report – Explosions (Ref.77);
- APP-1000-E6C-002, Revision 0 – **AP1000** Nuclear Island Hydrogen Piping, Equipment and System Assessment (Ref.98);
- UKP-1000-N4C-002, Revision 0 – UKP **AP1000** Assessment of the Potential for Hydrogen Combustion due to Leakage from Hydrogen Injection Lines in the Auxiliary Building (Ref.105);
- UKP-1000-N4C-004, Revision 0 – UK **AP1000** WLS and WGS Hydrogen Assessment (Ref.107);
- UKP-1000-N4C-005, Revision 0 – **AP1000** Hydrogen Gas Explosion Evaluation using TNO MEM (Ref.108);
- UKP-1000-N4C-006, Revision 0 – UKP **AP1000** Hydrogen Explosion Evaluation of Battery Rooms within the Annex Building (Ref. 191);
- UKP-1000-N4C-007, Revision 1 - FAI/16-1170, **AP1000** Hydrogen Migration Analysis for CVS Hydrogen Line Break in Selected Rooms of the Auxiliary Building and Containment Building (Ref. 109).
- UKP-1000-N4C-008, Revision 0 - FAI/16-1481, Unmitigated Explosion Hazard Analysis for **AP1000** Division B Battery Room 1 (Room 12104). (Ref. 110).
- UKP-GW-GL-114, Revision 0 – UKP **AP1000** Auxiliary Building Battery Rooms – Hydrogen Assessment (Ref.100).

7.3 AP1000 INTERNAL EXPLOSION CLAIMS

426. The overarching high level safety claim addressing the internal explosion challenges within the **AP1000** design basis is summarised as (Ref. 69):

- **Claim IH-4.0:** Postulated internal explosion within the design basis do not prevent the delivery of the Category A Safety functions and the supporting post-72 Category B safety functions necessary to respond to the postulated event.

427. The following key claims and sub-claims underpin the high level safety claim:

- **Claim IH-4.1**: Internal explosions which could compromise delivery of Category A safety function are prevented by controlling flammable substances such that an explosive atmosphere does not form.
- **Sub-claim IH-4.1.1**: The risk of explosion on-site from flammable materials that have the potential to generate explosive atmospheres is minimised by inventory control.
- **Sub-claim IH-4.1.2**: Under normal conditions, flammable substances which if released could form an explosive atmosphere will be adequately contained.
- **Sub-claim IH-4.1.3**: Under fault conditions, a guillotine break or a leak of the CVS hydrogen injection line in the Auxiliary Building will not result in the formation of an explosive atmosphere.
- **Sub-claim IH-4.1.4**: Under normal conditions, an explosive atmosphere is prevented from forming in the Auxiliary Building battery rooms by appropriate ventilation design.

- **Sub-claim IH-4.1.5**: Under fault conditions, an explosive atmosphere is prevented from forming in the auxiliary building battery rooms by operator action.
- **Sub-claim IH-4.1.6**: Under the faulted condition of a Loss of Offsite Power (LOOP), an explosive atmosphere is prevented from forming in the Auxiliary Building battery rooms by the inherent safety characteristics of the battery charging system.
- **Sub-claim IH-4.1.7**: Under fault conditions, a guillotine break or a leak of the CVS hydrogen injection line in the Containment Building will not result in the formation of an explosive atmosphere.
- **Sub-claim IH-4.1.8**: Explosive atmospheres will be prevented from forming in tanks / vessels by maintaining the atmosphere outside of the flammable range.
- **Sub-claim IH-4.1.9**: Hydrogen will not be present within the Liquid Radwaste System (WLS) in significant concentrations downstream of the degasifier.
- **Claim IH-4.2:** Passive protective measures have been incorporated in the **AP1000** design to protect SSCs that deliver Category A safety functions from internal explosions.
- **Sub-claim IH-4.2.1**: Safe shutdown SSCs located within the NI would not be affected by internal explosions generated in areas outside the NI.
- **Sub-claim IH-4.2.2**: Redundant safe shutdown SSCs located within the NI would not be affected by a hydrogen deflagration event on the NI battery rooms.

428. The RP identified the following, which deliver the above claims:

- minimum inventory;
- HVAC system to maintain ventilation in areas containing Class 1 SSCs outside Containment;
- hydrogen detection and alarm system to alert operational staff;
- low ventilation flow detection and alarm system to alert operational staff;
- redundancy of Class 1 SSCs;
- Battery Rooms Class 1 barriers;
- administrative procedures to isolate equipment.

7.4 ASSESSMENT OF GDA ISSUE GI-AP1000-IH-04

7.4.1 SCOPE OF ASSESSMENT

429. The assessment strategy in Section 2 was used to formulate the scope.

430. I assessed the internal explosions roadmap (Ref. 172) along with the regulatory query RQ-AP1000-1304 response (Ref. 27) during the initial stages of the assessment process. This was to aid my understanding of the safety case structure. As the safety case documentation developed, the roadmap was superseded. Up-to-date information is provided in the PCSR (Ref. 69) and internal explosions Topic Report (Ref. 77).

431. My assessment cover the deliverables used in addressing the actions for the Resolution Plan for GDA Issue GI-AP1000-IH-04. I assessed the Topic Report and the internal explosion section of the PCSR. I sampled supporting documents to check for substantiation or evidence.

432. The areas chosen to review the internal explosion safety case were limited to:

- There are a total of seven Class 1 battery rooms (12101, 12102, 12103, 12104, 12105, 12202 and 12204). Five out of the seven rooms were evaluated as I considered these to be most important to nuclear safety.

- Hydrogen pipework within areas containing Class 1 SSCs. Hydrogen pipework is routed through containment and through 25 rooms in the Auxiliary Building. Not all rooms contain Class 1 SSCs. Therefore the assessment sample is being limited to those areas containing Class 1 SSCs only. Explosions involving substances other than hydrogen are not within the scope of the Resolution Plan GI-AP1000-IH-04.

7.4.2 ASSESSMENT OF CLAIMS, ARGUMENTS AND EVIDENCE

433. The RP undertook a systematic identification of explosion sources and associated unmitigated consequences, which culminated in the derivation of specific claims as given above.

7.4.2.1 Substantiation of the safety case for explosion within Battery Rooms

434. The RP used the following methodology to analyse explosion hazards within the Battery Rooms (PCSR Chapter 11, Ref. 69):

- Hydrogen evolution rates were calculated for a battery cell following the IEEE 1184 guidance, analysing the normal float condition and equalising / recharging conditions at maximum operating temperatures. The accumulation rate was multiplied by the total number of battery cells, which comprised 60 lead-acid cells per room.
- The RP identified that at higher temperatures, hydrogen evolution rates would increase. The evolution rate at a normal operating temperature of 22.8^{o}C was compared with a maximum abnormal temperature of 48.9^{o}C.
- The time for the hydrogen concentration to reach 4% by volume (Lower Flammability Limit (LFL) and 1% by volume without HVAC being available were determined.
- Ventilation exhaust rates and background hydrogen concentrations were determined using BS EN 60079-10.
- Analysis of the results and arguments were presented (PCSR Chapter 11, Ref. 69).

435. There are four divisional Class 1 Battery Rooms (12101, 12102, 12104 and 12105) and a spare Class 1 Battery Room (12103) on elevation 89.789m (66’6”). These are situated adjacent to each other within the Auxiliary Building non-Radiologically Controlled Area (Non-RCA). In the event of a total loss of off-site and on-site AC sources, the Class 1 DC batteries constitute the sources of electrical power for operation of the required DC instruments. The batteries provide power for Class 1 equipment required for the shutdown of the plant. They also provide power to the normal and emergency lighting in the MCR and the remote shutdown workstation (PCSR Chapter 6, Ref. 66).

436. The Battery Rooms contain vented lead-acid batteries which require charging. Once fully charged, a float charge is required to maintain the batteries at 100% capacity. This will be referenced in this report as a normal condition. The batteries are on a continuous charge. In the event of a loss of power, charging will be stopped and any hydrogen generation will cease. This is the inherent safety characteristic delivering sub-claim 4.1.6.

437. The worst case condition exists when forcing maximum current into a fully charged battery. The maximum temperature of 48.9^{o}C (120^{o}F) was determined using the ‘WGOTHIC’ code. The maximum temperature calculation was reviewed in GDA Step 4. I raised regulatory query RQ-AP1000-1439 (Ref. 36 to seek further clarity of this calculation).

438. The hydrogen evolution rates were determined using the IEEE 1184 guidance and cross-checked against the EPRI Book EL-5036.These are recognised international standards. The hydrogen accumulation rate per battery bank during:

- normal float charge at 22.8 ^{o}C is 3.4 x10^{-6} m^{3}/s (0.43 ft^{3}/hr);
- normal float charge at 48.9 ^{o}C is 2.6 x10^{-5} m^{3}/s (3.30 ft^{3}/hr);
- equalising charge (worst case) at 48.9 ^{o}C is 5.67 x10^{-5} m^{3}/s (7.21 ft^{3}/hr).

439. Respectively, it takes 27.3 days, 3.6 days and at worst case, 1.6 days to reach the LFL hydrogen concentration without crediting ventilation identified in Reference 100.

440. The Battery Rooms are mechanically ventilated to maintain the concentration of evolved hydrogen below 1% by volume, which is below the LFL of 4% by volume. In order to contain hydrogen within the Battery Rooms, the ventilation system maintains a slightly negative pressure in the Battery Rooms by supplying less air than is exhausted. The remaining air is allowed into the rooms from adjacent environments through the door gaps to each respective Battery Room. Each room receives 230 cubic feet per minute, cfm (0.109 cubic metres per second, m^{3}/s) of HVAC supply air, 100cfm (0.047m^{3}/s) of transfer air under the door, and 330 cfm (0.156 m^{3}/s) HVAC exhaust air (Battery Room – Hydrogen assessment, Ref. 100). This delivers sub-claim IH-4.1.4 and is in line with SAPs EKP.4 and EKP.5.

441. If the ventilation system fails, a Class 3 ventilation low flow alarm is activated in the MCR. The set point is 75 % of the design airflow (VBS Instrument requirements, Ref. 187). Protection is via administrative procedures and requires an operator to physically check the ventilation system. This delivers part of sub-claim 4.1.5.

442. However, if the procedure has not been initiated, it takes 163 hours (normal float at 22.8^{o}C), 21.4 hours (normal float at 48.9^{o}C) and 9.8 hours (worst case condition) to reach a concentration of 1% hydrogen within the Battery Room. This is the Class 3 hydrogen detection alarm set point, and was based on the recommendation in line with BS6133:1995 (Ref. 11). However, this standard has been replaced with BS EN 50272-1:2001 (Ref. 12), which takes into consideration of the presence of personnel and therefore requires lower concentration levels.

443. The RP argued that the Battery Rooms are not in manned areas and that the batteries are not charged during periods of off-line maintenance when personnel may be present as advised in regulatory query RQ-AP1000-1506 (Ref. 40). Therefore, they have proposed that the alarm setpoint remains at 1%. The alarm is raised in the MCR. Protection is via administrative procedures and requires an operator to take air samples and stop activities in the area that may be a potential ignition source. This is required to deliver sub-Claim 4.1.5.

444. Neither the low ventilation flow alarm nor the hydrogen detection alarm cease the battery charging. SAP ERL.3 states automatically-initiated measures should be provided where "*reliable and rapid protective action is required*." If the ventilation low flow alarm and the hydrogen alarm are activated and administrative procedures are not initiated, it would take 65+ hours (normal condition) or 29+ hours (worst case condition) to reach the LFL. It would also require multiple shifts to fail to act on the alarm activation. A source of ignition is also required for an internal explosion to occur.

445. There is a significant length of time taken to reach the LFL hydrogen concentration. Therefore, I consider that an automated trip to cease battery

charging is not essential and SAP EKP.3 is adequately addressed together with SAP ERL.3. In addition, under normal modes of operation, if the Class 1 Battery Rooms fail, it does not affect the safety shutdown of the reactor. The Class 1 Battery Rooms are required only when there is a loss of on-site or off-site power. I consider the analysis adequate for nuclear safety at the GDA stage.

446. Further consideration of impacts on people and the adequacy of the 1% hydrogen concentration alarm point, in line with BS EN 50272-1:2001, should be addressed by the licensee. Additional human factor analysis is required post-GDA on human performance to incorporate emergency and site impacts and take cognisance of SAPs ECS.3, EHF.1 and EHF.7. This is covered by ONR's human factors specialist assessment inspector in his assessment report (Ref. 176). Therefore, no additional internal hazard assessment findings will be required here.

447. In GDA Step 4, explosion barriers were claimed, but it was highlighted that they were not adequately substantiated (ONR's Step 4 internal hazards Assessment Report, Ref. 13). During the GDA closure phase, the RP removed the claim on the barriers (Refs. 68 and 76). As part of the internal hazards workshop, December 2016, the RP was challenged as to why this previous claim had been retracted. Also, it was not clear for a postulated event of an explosion in a Battery Room, what the unmitigated consequence would be.

448. In addition, as part of the same workshop, cross-cutting discussions were held with Fault Studies. This was to discuss ONR's expectations on the hazard schedule and fault schedule as part of GDA issue GI-AP1000-FS-08. Both, schedules had failed to summarise the RP's safety case. Consequently, the RP issued a revised internal explosions hazard schedule in draft on the 29 December 2016. From my initial assessment, I considered that the revised hazard schedule significantly impacted on the demonstration of the safety case arguments. The narrative in the Topic Report and PCSR were updated and provided at a later date (discussed below).

449. The RP reviewed its internal explosion safety case analysis. Battery Room 12104 was analysed for the scenario of unmitigated hydrogen release and accumulation using a release rate for a sustained equalising charge.

450. ONR was notified on 18 January 2017 of a new internal explosions sub-claim IH-4.2.2, '*Redundant safe shutdown SSCs located within the NI would not be affected by a hydrogen deflagration event in the NI battery rooms.'* Evidence underpinning this sub-claim was subsequently provided in a new document, '*Unmitigated explosion hazard analysis for **AP1000** Division 'B' Battery Room 1 (Room 12104)* (Ref. 110)'. The revised Topic Report and PCSR Chapter 11 were provided on 20 January 2017.

451. The analysis confirmed that for a hydrogen concentration range of 4.5 to 6.0% v/v, a deflagration would occur, but would be handled via the venting through the HVAC ducts. The explosion pressure would remain below the withstand pressure of the Battery Room barriers of 5 psi.

452. For a hydrogen concentration range of 8.0 to 12.0% v/v an explosion would exceed the barrier withstand pressure and remain above the limit for 45 -60 seconds if venting is not credited. If venting is credited, the time for the pressure exceedance is up to 6 seconds. However, if the Battery Room doors are credited as a pressure relief defence-in-depth measure, then the peak pressure remains below 5 psi for the duration of the deflagration. The doors open out into the corridor Rooms 12111 which do not contain SSCs serving Category A functions.

453. Due to the late delivery of the submission, I did not assess the calculation for the assurance of validity of data and models in line with SAPs AV.1 - 8. It is also not clear what the unmitigated consequence is in the event of an explosion for a hydrogen concentration range of 8.0 to 12.0% with venting not credited. The current analysis states that the barrier withstand pressure is exceeded. However, this scenario has been carried out for sensitivity analysis, and it would take many more hours of high hydrogen generation with insufficient venting before such high concentrations would be achieved. At any stage after the formation of a flammable (explosive) mixture, the presence of an ignition source would cause the deflagration. This scenario therefore represents a very unlikely outcome, and has been examined to illustrate whether there is a 'cliff-edge' in consequences just beyond the more realist scenario of ignition at or near the LFL.

454. I am raising two assessment findings. The first is to ensure that justification of mathematical models will be carried out. This shall take cognisance of SAPs AV.1 to AV.8. The second assessment finding is to ensure that an unmitigated consequence analysis is fully considered and addresses SAP FA.7.

CP-AF-AP1000-IH-09 – The licensee shall justify the mathematical models which:

- **predict the hydrogen concentration;**
- **predict explosion pressures;**
- **to determine that barriers provide adequate protection to SSCs against the potential explosions.**

CP-AF-AP1000-IH-10 – The licensee shall carry out unmitigated consequence sensitivity analysis in the Battery Rooms for various hydrogen concentrations without crediting venting.

455. This analysis is also a crucial input to the civil engineering barrier design consideration. Regulatory query RQ-AP1000-1786 (Ref. 57) was raised to ensure that all relevant internal hazards inputs were captured in the civil engineering design in line with SAP ECE.6.

456. I refer to existing GDA Step 4 assessment finding AF-AP1000-CE-09 relevant to the barriers which was raised by Civil Engineering. This requires liaison with Internal hazards taking account of the revised calculation analysis. This was discussed earlier in section 6.5.1.

457. The RP advised in regulatory query RQ-AP1000-1794 (Ref. 61) that ventilation dampers are not claimed in support of the explosion safety case. The failure of HVAC dampers is included in the fault schedule under Fault ID 3.5.4. (PCSR Chapter 8, Ref. 67).

458. From the sampling that I have been able to review, I consider that limiting the maximum concentration of hydrogen to 12%v/v is a reasonable assumption for the calculation of 'cliff edge' consequences. This would require a significant time to reach the concentration (4.5+ days), and would require a number of equipment and instrument failures, along with a failure of administrative procedures, before an explosion would occur.

459. The RP determined that explosions resulting from hydrogen concentrations between 8.0 to 12.0% v/v, would in the unlikely event of a failure of the vents exceed the barrier withstand pressure. To check calculation sensitives, explosion without crediting venting shall be considered using site-specific information. These are noted in the above assessment findings CP-AF-AP1000-IH-09 and CP-AF-AP1000-IH-10. Overall, the arguments support sub-Claim IH-4.2.2 and are broadly in line with SAP FA.7.

460. Overall, I am satisfied that the evidence and arguments to underpin sub-claims IH-4.1.4, IH-4.1.5, IH-4.1.6 and IH-4.2.2 within Auxiliary Building Class 1 Battery Rooms are suitable and sufficient for GDA.

7.4.2.2 Substantiation of the safety case for the routing of the hydrogen pipework within areas containing Class 1 SSCs

461. The RP used the following methodology to analyse explosion hazards arising from hydrogen pipework (PCSR Chapter 11, Ref. 69):

- identified systems which would contain hydrogen;
- for each system identified pipe routes and postulated break locations.
- calculated leakage rates;
- ventilation exhaust rates and background hydrogen concentrations were determined using BS EN 60079-10;
- analysis of the results with arguments presented.

462. Within the NI, the main hydrogen pipework systems considered at risk from internal explosion are:

- Chemical and Volume Control System (CVS);
- Primary Sampling System (PSS);
- Gaseous Radwaste System (WGS);
- Liquid Radwaste System (WLS).

463. **CVS:** Hydrogen is supplied from the CVS Hydrogen Injection Package (APP-CVS-MS-02) which contains four high pressure hydrogen cylinders. These are located in the Plant Gas System (PGS) in the yard. The line passes the Turbine Building via Room 21480 and enters the Auxiliary Building via Room 12306, 12341 and continues into containment. Here the line connects with the CVS purification return line before entering the reactor coolant system.

464. **PSS:** The PSS collects representative samples of fluids from the process streams of the reactor coolant system and associated auxiliary systems and from the containment atmosphere for analysis by the plant operating staff.

465. **WGS:** The WGS receives processes and discharges the radioactive waste gases during all modes of plant operation. The primary feeds to the WGS are from the Reactor Coolant Drain Tank (RCDT) and the WLS system degasifier. The degasifier receives a feed from the CVS letdown and RCDT which incorporates dissolved hydrogen. The degasifier is operated under vacuum conditions to remove the dissolved hydrogen from the CVS letdown feed for onward processing by the WGS.

466. **WLS:** The WLS receives borated and hydrogen-bearing water from the PSS, RCDT and CVS. The liquid waste is passed through a degasifier to remove dissolved hydrogen (and other radioactive gases). The degasifier is operated under vacuum conditions to remove the dissolved hydrogen from the CVS letdown feed for onward processing by the WGS.

467. My assessment of the hydrogen pipework systems in combination with the claims identified are discussed below.

CVS

468. The RP identified that it was remotely possible for an explosion hazard associated with the CVS to exist in Rooms 12306, 12406, 12506 and 12341 inside the Auxiliary Building analysed under Reference 105; and 11209 and

11300 inside Containment analysed under Reference 106. Class 1 SSCs are located in these rooms.

469. In the Auxiliary Building, the RP assumed that hydrogen moved upward and diffused outward, accumulating below the ceiling in Room 12306 (Assessment of the potential for hydrogen combustion due to leakage from the hydrogen injection line in the auxiliary building, Ref. 105). Similarly, in Containment, it was assumed that hydrogen moved upward and diffused outward.

470. Local pockets of hydrogen may accumulate at the ceiling of Room 11209 in the absence of active airflow and may pose an explosion risk due to nearby ignition sources. There are three openings in the ceiling at one end of Room 11209 for hydrogen to diffuse into Room 11300.

471. The RP argued that the hydrogen will diffuse, rise upwards and become diluted into the vast free volume of Room 11300, which includes the Containment upper dome where the hydrogen concentration is monitored. Based on the physical properties, I considered that the assumption of upward movement and diffusion was reasonable and broadly in line with SAP EHA.1.

472. However, the methodologies in the above references failed to demonstrate a full unmitigated consequence analysis in line with SAP EHA.6. Subsequently, new analysis was carried out considering the event of a double-guillotine break of a nominal one inch CVS line for a postulated event inside Containment and inside the Auxiliary Building. The preliminary report (Ref. 109) was shared with ONR on 21 November 2016. The mathematical modelling focused on the hydrogen accumulation in Room 12306 in the Auxiliary Building and Room 11209 inside Containment.

473. I examined the calculation outputs. The results show that after 14 days of continuous leakage, the upper portion of Room 12306 and Rooms 11209 / 11300 will not reach 1% v/v of hydrogen (Preliminary report, hydrogen migration analysis, Auxiliary and containment building, Ref. 109, Tables 11-1 and 11-2 respectively). In considering the Auxiliary Building, it would take the order of 54 days for the upper portion of Room 12306 to reach 1% v/v hydrogen concentration (PCSR Chapter 11, Ref. 69). However, the quantity of hydrogen is limited to four cylinders. Therefore, the RP argues that a continuous leak of 31 days or more would exceed the hydrogen cylinder supply. A similar amount of time is taken for Room 11209 inside Containment. The general Containment area 11300 shows a zero concentration after 14 days of continuous leakage.

474. I am satisfied, therefore, that there is a significant amount of time to enable operators to take action for postulated CVS pipe breaks inside Containment and the Auxiliary Building.

475. The model ‘FATE™’ was used to determine the above outputs for Rooms 12306 and 11209. Due to the late delivery of submissions, I did not sampled the calculation for the assurance of validity of data and models, in line with SAPs AV.1 to AV.8. I refer to the earlier assessment finding CF-AF-AP1000-IH-09 that requires that a justification of the mathematical models will be carried out. Therefore, no additional assessment finding is required.

476. The hydrogen-containing portion of the CVS does not have any threaded or bolted connections (**AP1000** Nuclear island hydrogen piping equipment and system assessment, Ref. 98), eliminating the possibility of leakage at the pipe connections during normal operation. In addition, all valves in the CVS hydrogen injection line are hermetically sealed as identified under Table 4-5 (Ref. 98, which references DCP APP-GW-GEE-4929. The DCP was not included in my

sampling). The RP argues that the design of the CVS piping system is compliant with ASME Section III and B31.1 standards (Refs. 9 and 10). These design measures provide additional defence-in-depth. Therefore, I consider that hydrogen should be adequately contained within the pipework and the design measures deliver sub-claim IH-4.1.2. This is in line with SAPs EKP.3, EKP.5 and ECS.3.

477. With respect to the Auxiliary Building, I am satisfied that the above arguments and evidence delivers sub-claim IH-4.1.3.

478. I consider that the case inside Containment requires additional sampling and is discussed further.

479. It was identified earlier that in Containment hydrogen will diffuse, rise upwards and become diluted into the vast free volume of Room 11300, which includes the Containment upper dome where the hydrogen concentration is monitored. I was concerned that hydrogen may accumulate in the upper dome area.

480. The hazard schedule (PCSR Chapter 11, Ref. 69) states that the Class 3 Containment Recirculation Cooling System (VCS) ventilation is a defence-in-depth measure. The VCS recirculates and cools air within the Containment during power operations and shutdown. The air recirculated by this system does not penetrate the Containment boundary and thus does not give rise to any discharges to atmosphere. In addition, the Containment Air Filtration System (VFS) purges the containment by providing fresh air from outside and exhausting containment air into the plant vent. The air exhausted by the VFS is filtered with high-efficiency filters, charcoal filters, and post-filters (PCSR Chapter 6, Ref. 66). I consider that these systems would aid the diffusion of hydrogen inside Containment.

481. An additional defence-in-depth measure is the Containment Hydrogen Control System (VLS). The VLS comprises:

- hydrogen monitoring: three hydrogen sensors located in the upper dome which trigger an alarm in the MCR when the hydrogen concentration reaches 1% v/v;
- two hydrogen Passive Autocatalytic Recombiners (PARs);
- 66 hydrogen igniters. These require to be energised by the operator when alerted to a high hydrogen concentration alarm.

482. PARs are designed to accommodate the relatively slow hydrogen production rate anticipated for a design basis LOCA. The volume of hydrogen released as a result of a LOCA is significantly greater than that from either the CVS or WLS. Therefore, the PARs will have a sufficient capacity to allow them to maintain hydrogen concentrations below the LFL. The PARs are entirely passive devices relying on the catalytic properties of the material (palladium or platinum) to induce catalytic oxidation of the surrounding hydrogen. Therefore, no power supplies or other supports are required.

483. Three hydrogen sensors are located in the upper dome, which trigger an alarm in the MCR at 3% v/v hydrogen in air (Regulatory Query RQ-AP1000-1794, Ref.61 and the alarm response for the Containment Hydrogen Control System, Ref. 192). I did not carry out detailed sampling to check for the adequacy of the alarm set point as there are no internal hazard claims on this system and this system has been implemented as a defence-in-depth measure. Therefore, I consider these additional measures in line with SAPs EKP.3 and EKP.5.

484. The RP provided arguments and evidence that sub-claim IH-4.1.7 is delivered through a combination of protective measures. The maximum number of

hydrogen cylinders is limited to four in the CVS system, as a result I am also satisfied that this system supports sub-claim IH-4.1.1.

PSS

485. The PSS is a fluid-based system. The hydrogen component is dissolved in the liquid and would not readily off-gas while contained in this system. The PSS uses flanged bolted joints which are pressure rated and tested in accordance to ASME B31.1 standards (Ref. 10). This is a recognised standard and provides additional defence-in-depth in the containment of liquids. Thus, any hydrogen would be dissolved in the liquid and contained within the pipework. Therefore, I consider that the design of this hydrogen pipework system supports sub-claim IH-4.1.2. This is in line with SAPs EKP.3, EKP.5 and ECS.3.

WGS and WLS

486. The RP carried out a room-by-room evaluation of the NI, examining hydrogen containing lines in the WLS and WGS systems. Rooms that contain hydrogen lines are identified in Table 4-1 of Reference 106. I sought additional clarification regarding the hydrogen evolution rates in the rooms identified and the rooms containing Class 1 SSCs. An updated Table 4-1 was provided as part of my assessment closure phase regulatory query RQ-AP1000-1794 (Ref. 61). I considered the characterisation process to be in line with SAPs EHA.1 and EHA.19.

487. The WLS lines are routed via 16 rooms. The hydrogen component is dissolved in the liquid and would not readily off-gas while contained in these systems. The WLS uses flanged bolted joints which are pressure rated and tested in accordance to ASME B31.1 standards (Ref. 10). This is a recognised standard and provides additional defence-in-depth. Thus, any dissolved hydrogen is contained within the pipework. I consider that the design of this hydrogen pipework system supports sub-claim IH-4.1.2. This is in line with SAPs EKP.3, EKP.5 and ECS.3.

488. Out of the 16 rooms containing WLS lines, only five rooms contain Class 1 SSCs. These are Rooms 11104, 11204, 11300, 12244 and 12341 (response to regulatory query RQ-AP1000-1794, Ref. 61). Rooms 11104 and 11204 are open to inside Containment Room 11300. In the unlikely event that there is a fluid leak in these rooms and the dissolved hydrogen off-gases, then hydrogen would rise upwards and become diluted into the vast free volume of Room 11300. The VCS, VFS and VLS provide additional defence-in-depth protection. These systems were discussed earlier under the section on the CVS system. Defence-in-depth protection in Rooms 12244 and 12341 is provided by the Class 3 Radiologically Controlled Area Ventilation System (VAS).

489. The degasifier receives influent from the CVS letdown and the Reactor Coolant Drain Tank (RCDT) at a maximum dissolved hydrogen concentration of 4.5E-05 m^3/ kg. Based on the maximum dissolved hydrogen concentration and the CVS letdown flowrate of 6.3E-03 m^3/s (100 gpm), the hydrogen content of the influent is approximately 2.84E-04 m^3/s (0.6 scfm).

490. The degasifier boils off dissolved hydrogen under vacuum conditions and discharges the hydrogen to the WGS for onward processing. The degasifier can discharge up to 2.7E-04 m^3/s (0.58 scfm) to the WGS.

491. The liquid effluent is discharged to the Class 2 WLS hold-up tanks located in Rooms 12171 and 12172 (response to regulatory query RQ-AP1000-1794, Ref. 61). The WLS effluent may contain up to 1.4E-05 m^3/kg of hydrogen. This

remains dissolved in the WLS effluent and represents 3% of the initial content. I consider that this is not a significant concentration downstream of the degasified effluent. Therefore, this delivers sub-claim IH-4.1.9 and is line with SAP EKP.2.

492. The conditions within the WLS hold-up tank would not promote boiling of the effluent. The tank is held at atmospheric conditions. Any hydrogen would be dissolved in the liquid and contained within the tank. Also, Rooms 12171 and 12172 do not contain any Class 1 SSCs. Thus these rooms do not contain hydrogen detectors. However, the RP considered the scenario that dissolved hydrogen may off-gas into the gaseous space within the WLS hold-up tank. The vent line from each of the hold-up tanks incorporates a hydrogen monitor to detect hydrogen accumulation. At 1% v/v, a Class 3 hydrogen detection alarm is activated in the MCR.

493. The hydrogen detection alarm does not initiate any automated measures. Protection is via administrative procedures and requires an operator to initiate an air purge through the tank to maintain the concentration below the flammable limit.

494. If the administrative procedure has not been initiated, it takes approximately 6.9 hours to reach the LFL (response to regulatory query RQ-AP1000-1794, Ref. 61). A source of ignition is also required for an internal explosion to occur. I discussed this with the human factors assessor and conclude this to be an adequate amount of time for an operator response. Therefore, the combination of the low hydrogen concentration in the WLS hold-up tanks, vent line hydrogen detection and administrative procedures deliver sub-claim 4.1.8. These measures are in line with SAPs EKP.3 and EKP. 5.

495. Further consideration of impacts to people should be addressed by the licensee. I refer to the human factors assessment findings which were discussed in the above section on Class 1 Battery Rooms. These require consultation with internal hazards. Therefore, no additional assessment finding regarding operator action will be raised within the internal hazards assessment.

496. The WGS lines are routed via nine rooms. WGS systems are designed to prevent leakage through use of welded pipe joints and leak-tight (hermetically sealed) valves compliant with ASME Section III and B31.1 standards (Refs. 9 and 10). These are recognised standards and provide additional defence-in-depth for the containment of gaseous fluid systems. Therefore, I consider that the design of this hydrogen pipework system supports sub-claim IH-4.1.2. This is in line with SAPs EKP.3, EKP.5 and ECS.3.

497. Out of the nine rooms that contain WGS lines, only Room 12553 contains Class 1 SSCs (Refs.61 and 106). I sampled the **AP1000** WLS and WGS hydrogen assessment (Ref. 107) and found that personnel hatch test connections (VUS-PL-V017, V018 and V019) and a personnel hatch at elevation 135’-3” (CNS-MY-Y03) form the Class 1 SSCs for Room 12553. A loss of the personnel hatch and test connections at elevation 135’-3” would not prevent safe shutdown of the plant and, therefore, I am not considering these Class 1 SSCs further.

498. However, it is noted that these Class 1 SSCs are not included in the internal explosions hazard schedule (Refs. 69 and 77). The hazard schedule should fully summarise the internal hazards safety case, summarise potentially affected Class 1 SSCs, and capture all claimed safety features and defence-in-depth measures. This was highlighted as part of the ONR cross-cutting GDA discussions on GDA Issue GI-AP1000-FS-08. It is acknowledged that the RP provided a revised internal explosions hazards schedule following the internal

hazards December 2016 workshop. However, I am raising a minor shortfall to ensure that the hazard schedule fully summarises the safety case. The update should address SAP FA.8.

CP-MS-AP1000-IH-01 – The licensee shall update the hazard schedule with site-specific information to ensure that the internal hazards safety case is fully captured. The hazard schedule shall summarise the Class 1 SSCs, as well as the Essential Safety Shutdown SSCs, and capture all claimed safety features and defence-in-depth measures.

499. Overall, I am satisfied that the evidence and arguments to underpin sub-claims IH-4.1.2, IH-4.1.3, IH-4.1.7, IH-4.1.8 and IH-4.1.9 are suitable and sufficient for the routing of hydrogen pipework containing Class 1 SSCs.

Internal Explosion generated in areas outside of the NI

500. Areas outside of the NI include the Annex Building, Turbine Building, Radwaste Building and those buildings / structures / installations not directly adjacent to the NI wider yard area. There are no Class 1 SSCs located in areas outside of the NI. Therefore, assessment of these areas falls outside of the scope of Resolution Plan GI-AP1000-IH-04. For completeness, I provided a summary discussion here.

501. Areas that accommodate sources of hydrogen or other materials that have the potential to generate explosive atmospheres outside the NI are located at safe distances from the NI. The RP defined 'safe distance' as that at which the maximum explosive overpressure resulting from an explosion is limited to 7kPa and, therefore, will not damage the NI structure (Ref. 69). Thus, the SSCs are protected by the external walls of the Containment Shield Building and Auxiliary Building.

502. The RP used a combination of the 'TNT Equivalence' methodology and "TNO Multi-Energy" method (Refs. 102 and 108). The methodologies are in line with my expectations for the scenarios considered and I consider them to fulfil SAP FA.7.

503. The external walls of the Containment Shield Building and Auxiliary Building are crucial to the protection of the Class 1 SCCs inside the NI. Therefore, the civil engineering design should take account of the outcomes of the internal explosion loadings on the external walls in line with SAP ECE.6. Refer to the Civil Engineering GDA Step 4 assessment finding CF-AP1000-CE-09 discussion earlier in this report.

504. I consider that the arguments and high-level sampling of evidence underpin sub-claim IH-4.2.1.

7.5 COMBINED CONSEQUENTIAL HAZARDS

505. Combined events and their associated combined consequential loads have the potential to compromise the safety measures in place against internal explosions such as barriers.

506. The RP considered a combination of internal hazards postulated to initiate plant level faults. (Combined hazards topic report, Ref. 71). I sampled the initiating events or event combinations which resulted in an internal explosion. Sources of explosions were identified and the necessary conditions to induce failures were presented (Regulatory query RQ-AP1000-1301, Ref. 24) and correlated (Combined hazards topic report, Ref. 71). I considered that the initiating events

and subsequent fault sequences followed a systematic approach in line with SAP FA.3.

507. However, the RP stated that it is not possible for an explosion to be the initiator of combined consequential internal hazards. The combined hazards Topic Report (Ref. 71) is dated from August 2016 (Ref. 71). The internal explosions safety case was updated with new explosion analysis provided on 20 January 2017 (Explosions Topic Report, Ref. 77). The combined consequential hazards safety case has not incorporated the new internal explosion analysis from January 2017 (PCSR Chapter 11, Ref. 69).

508. An earlier assessment finding CP-AF-AP1000-IH-02 required the consideration of all credible internal and external hazard combinations (consequential, correlated and independent). The combined consequential hazards safety case should incorporate any impacts from the internal explosions analysis and address SAPs SC.4 and EHA.1.

7.6 ASSESSMENT FINDINGS

509. During my assessment two assessment findings were identified for a licensee to take forward in their site-specific safety submissions. These are summarised in Annex 2. These matters do not undermine the generic safety submission.

7.7 MINOR SHORTFALLS

510. During my assessment one item was identified as minor shortfall in the safety case, but which is not considered serious enough to require specific action to be taken by the licensee. Details are contained in Annex 3.

511. A residual matter is recorded as a minor shortfall if it does not:

- undermine ONR’s confidence in the safety of the generic design;
- impair ONR’s ability to understand the risks associated with the generic design;
- require design modifications;
- require further substantiation to be undertaken.

7.8 GDA ISSUE GI-AP1000-IH-04 CONCLUSION

512. The submission provides the requisite information relating to the safety case for explosion in the Battery Rooms and for routing of hydrogen pipework within areas containing Class 1 SSCs. The RP significantly revised their safety case to address the internal hazards GDA Issue. There is also an improved alignment with the fault studies area.

513. Suitable and sufficient claims have been made and these were generally supported by the requisite arguments and evidence. I identified assessment findings that require to be taken forward as part of site-specific activities. Further consideration of all postulated events, consequence analysis and adequacy of safety measures is required using site specific information. This includes addressing the internal explosion assessment findings CP-AF-AP1000-IH-09 to IH-10 and the generic assessment findings which are summarised in Annex 2.

514. I am satisfied that during the GDA internal explosion has been subjected to an adequate review and substantiation. Therefore, issue GI-AP1000-IH-04 can be closed.

8. ONR ASSESSMENT OF GDA ISSUE GI-AP1000-IH-05 – INTERNAL MISSILE

8.1 RESOLUTION PLAN ACTIONS AND DELIVERABLES

515. During GDA Step 4, a GDA issue was raised relevant to internal missiles identification and substantiation of claims, arguments and evidence (GI-AP1000-IH-05) ONR's Step 4 Internal Hazards Assessment Report, Ref. 13).

516. The GDA issue included the following action:

- GI-AP1000-IH-05.A1: Identify and substantiate the claims, arguments and evidence that constitute the internal missile aspects of the internal hazards safety case.

517. In order to address the action it was prudent for the RP to re-present the claims and arguments for the internal missile safety case in the PCSR. These are covered in the specific Topic Report and in the revised PCSR.

518. The RP's Resolution Plan (Ref. 22) identifies specific deliverables associated with the above action:

- Internal Missiles Roadmap;
- Internal Missiles Identification Report;
- Internal Missiles Topic Report.

519. The Internal Missile Roadmap (Ref. 174) was initially issued to show the link between the existing claims, arguments and evidence. The document has been superseded by the Internal Missile Topic Report (Ref. 75).

520. In addition to the list above, the RP issued a number of draft documents, multiple revisions of the Topic Report and the PCSR and a number of supporting documents.

521. During this phase of the GDA, I raised regulatory query RQ-AP1000-1305 aiming to seek clarity on the scope of submissions and on the Resolution Plan (Ref. 28).

522. In the following sub-sections, I will cover the following:

- the RP's safety case on internal missiles.
- my assessment of GI-AP1000-IH-05, which includes:
    - assessment of claims and arguments including combined consequential hazards;
    - assessment of the RP's internal missile design criteria;
    - assessment of the missile barriers.
- conclusions.

8.2 REQUESTING PARTY'S SAFETY CASE

523. Key document submissions for internal missile are:

- UKP-GW-GL-793, Revision 0D – **AP1000** Pre-construction Safety Report, Chapter 11 – Internal Hazards (Ref. 69);
- UKP-GW-GLR-108, Revision 1 – UK **AP1000** Internal Hazards Topic Report - Internal Missiles (Ref. 75);

524. The **AP1000** reactor missiles safety case utilises the following approaches to ensure that the Class 1 SSCs will continue to provide their Category A safety functions (Refs. 69 and 75):

- incorporation of design features in components to prevent missiles from being generated externally to the component;
- orientation of components, such as the main turbine, to direct any missile away from Safety Class 1 SSCs;
- location of Safety Class 1 SSCs outside the zone of influence of a potential missile;
- protection where practicable using a structural barrier.

525. The RP's safety case is based on the following:

- SSCs within compartments are assumed to fail (gross failure) as a result of missile strike;
- generating an internal missile is credible in any compartment with a potential missile source;
- hazard assessments evaluate the operating plant state applicable to each missile, as appropriate, and defined by the missile source;
- structural barriers assessed as missile barriers are claimed to prevent a missile exiting its originating compartment.

8.2.1 AP1000 INTERNAL MISSILE CLAIMS

526. The RP made the following claims (Ref 69 and 75):

- **Claim IH-5.0:** An internal missile event within the design basis does not prevent delivery of the Category A safety functions and supporting post 72-hour Category B safety functions necessary to respond to the postulated event.
- **Claim IH-5.1:** Internal missiles have been eliminated from the design so far as reasonably practicable.
- **Sub-claim IH-5.1.1:** Internal missiles generated from failure of rotating equipment are eliminated by design so far as reasonably practicable.
- **Sub-claim IH-5.1.2**: Internal missiles will not be created from SSCs which are classified as HSS.
- **Sub-claim IH-5.1.3**: The failure of valve stems, bonnets and thermowells in SSCs where the stored energy is high will not lead to internal missiles.
- **Sub-claim IH-5.1.4:** Nuts, bolts and nut bolt combinations have only a small amount of stored energy and are not considered credible missiles.
- **Sub-claim IH-5.1.5:** Gross failure of control rod drive mechanism housing is not considered a credible missile source.
- **Claim IH-5.2:** SSC's required for delivery of Category A and supporting Category B safety functions are protected by barriers that will prevent missile penetration.
- **Sub-claim IH-5.2.1:** The consequences of missiles will be protected through the use of passive barriers to limit the impact to and/or loss of a Category A or post 72 hour Category B safety functions.
- **Sub-claim IH-5.2.1.1:** SSCs are protected from missiles through internal barriers such as walls, floors and ceiling structures. Gross failure is assumed for all SSCs within the barriers affected by the missile.
- **Sub-claim IH-5.2.2:** Orientation of equipment will protect delivery of Category A or post 72 hour Category B safety functions in the NI.

527. Mitigation claims are not directly claimed for internal missile events.

528. The RP identified that the Class 1 passive barriers delivers claim IH-5.2.1.

8.3 ASSESSMENT OF GDA ISSUE GI-AP1000-IH-05

8.3.1 SCOPE OF THE ASSESSMENT

529. The assessment strategy in Section 2 was used to formulate the scope.

530. My assessment covers the deliverables used in addressing action GI-AP1000-IH-05.A1.

531. I assessed the Topic Report and the PCSR on internal missile, and I also sampled supporting documents to obtain confidence on the requisite evidence and substantiation of the claims made.

532. The areas chosen to review the internal missile safety case were limited to:

- inside Containment.
- outside Containment – Mainly Auxiliary Building.

533. The sections below cover the areas of my assessment.

8.3.2 ASSESSMENT OF CLAIMS AND ARGUMENTS

534. The RP undertook a systematic identification of internal missiles events and consequences analysis which culminated in the derivation of specific claims as given above. This section provides a high level review of the claims and arguments made by the RP.

535. I am generally satisfied with the approach that the RP took in relation to the high level principles of the claims, arguments and evidence for internal missiles.

536. My judgement is based on the following:

- the RP has taken all reasonable steps to ensure that missiles of sufficient energy to damage SSCs are prevented;
- the RP conservatively assumed that equipment will fail and missiles will therefore be generated;
- all SSCs in a room are assumed to be lost;
- the RP claims passive protection by means of walls, floors and ceilings, which is supported by the requisite substantiation.

537. The prevention claims (claim IH-5.1 and sub-claims IH-5.1.1 to IH-5.1.5) exclude some SSCs as credible missile sources and support the fulfilment of the high level claim. The constraint of credible missile sources simplifies the assessment and also reduces the number of SSCs, which require a detailed assessment. This approach is reasonable, providing sufficient and suitable arguments and evidence are provided.

538. Evidence and substantiation to support the claims IH-5.1.1, IH-5.1.3 and IH-5.1.5 is provided in the following documents:

- Design Criteria for the Protection from Internally Generated Missiles (Ref. 154) This document defines the criteria to be used for the design and analysis of SSCs with respect to the effects of internally generated missiles and the strategy to prevent internally generated missiles within the design.
- **AP1000** Valve Missile Protection (Ref. 155).
  This document defines how valves in high energy systems are designed to prevent missiles as required by the design criteria for the protection from internally generated missiles (Ref. 154). The document lists the design criteria for preventing failures of valve stems, bonnets and thermowells.

- Design Specification for **AP1000** Control Rod Drive Mechanism (CRDM) Design Specification for System: Reactor System (RXS) (Ref. 156).

539. I assessed these documents and raised regulatory query RQ-AP1000-1440 (Ref. 37) to obtain clarity on the criteria and methodology used in the analysis.

540. The RP responded to my queries, and I am overall satisfied that the arguments and evidence provided in these documents, including the internal missiles Topic Report (Ref. 75) and PCSR (Ref. 69), are suitable and sufficient to fulfil the mentioned claims.

541. The arguments presented by the RP in the Topic Report and PCSR, for the prevention claims IH-5.1.2 and IH-5.1.4 are clear and in line with my expectations and relevant good practice in this area.

542. With regard to Claim IH-5.1.2, Highest Safety Significance (HSS), this is broadly equivalent in definition to the term Incredibility of Failure (IoF), used within the UK structural integrity community to describe components where the likelihood of gross failure is sufficiently low that it can be discounted.

543. The protection claims IH-5.2.1 and IH-5.2.1.1 make use of passive barriers. This is in line with my expectations and relevant good practice.

544. Within the Containment, and based on the design criteria the majority of potential missile sources were not considered credible (Design criteria for the protection from internally generated missiles, Ref. 154). The RP, however, identified some missiles sources such as the Reactor Coolant System (RCS), which could cause failure of the SG vertical column. This is discussed further below.

545. I queried the role of ‘separation’ in the internal missile hazard safety case stated within the internal missiles Topic Report (Ref. 75) which states ‘*Within the Containment there is segregation and separation of Class 1 equipment. This is provided by a combination of barriers formed by the walls and floors of compartments and the distance between systems providing redundant means of delivering the Category A functions.’*

546. The RP clarified the position via their response to regulatory query RQ-AP1000-1440 (Ref. 37), which states that a spatial separation claim has not been used in the **AP1000** internal missile safety case. Concepts such as material selection, redundancy, barriers and orientation have been used.

547. Within RQ-AP1000-1440, I queried the mechanical failure of a control rod mechanism pressure housing resulting in the ejection of a Rod Cluster Control Assembly (RCCA) from the core.

548. The RP’s response to RQ-AP1000-1440 has been captured in the Topic Report, which included the following:

- Minimisation of potential missiles from failure of the control rod drive mechanism housings is achieved in a similar manner to the Reactor Vessel and other pressure boundary components considered in the structural integrity study.
- The housings are made of Type 304 or 316 stainless steel, which exhibits excellent notch toughness.
- The control rod drive mechanism housings are hydro-tested to 125% of system design pressure during the hydro-test for the completed RCS.

- The welds in the pressure boundary of the control rod drive mechanism meet the same design, procedure, examination, and inspection requirements as the welds on other ASME (Section III) Class 1 components.
- Stress levels in the mechanism are not affected by system thermal transients at power or by thermal movement of the coolant loops.

549. In addition, the Fault Schedule presented in Appendix 8A of the **AP1000** PCSR, considers the mechanical failure of a control rod mechanism pressure housing resulting in the ejection of a RCCA from the core (PCSR Chapter 8, Ref. 67). It concluded that escalation of the fault, due to the failed control rod drive mechanism housing acting as a missile, is not considered credible. This is due to the integrated upper head structure which prevents significant upward travel, and the lateral constraint provided by the actuating coil stack. The RP concluded that failure of the housing would not cause sufficient damage to adjacent housings to prevent RCCA insertion.

550. The arguments presented above satisfy claim IH-5.1.5.

551. Outside the Containment, the principle of segregation by passive barriers has been used. The RP argues that there is sufficient equipment redundancy, segregation, and protection such that all of the equipment in a single room can be lost without preventing delivery of the Category A safety functions.

552. I sampled the hazard schedule of the Topic Report and I satisfied myself that relevant barriers have been listed in the schedule.

553. I also sampled the Barrier Matrix (Ref. 104). The document lists relevant walls inside Containment as well as walls and floors inside the Auxiliary Building. I noted, however, that no claims on barriers inside the Containment have been made. Therefore, there is a need to update this document in line with assessment finding CP-AF-AP1000-IH-01.

554. The protection claim IH-5.2.2 is relevant to the orientation of equipment to warrant the high level claim. The turbine is oriented in such a way that its shaft axis is perpendicular to the NI. The orientation of the turbine is to minimise the possibility of a missile fragment from impacting the NI. This is in line with my expectations and relevant good practice.

555. Overall, I am satisfied with the conservative approach taken by the RP in the evaluation of the internal missiles, and the claims and arguments presented.

8.3.3 COMBINED CONSEQUENTIAL HAZARDS

556. Combined events and their associated combined consequential loads have the potential to compromise the safety measures in place against internal missiles such as barriers.

557. The RP undertook a study to identify all credible consequential, correlated or independent hazards relevant to internal missile and captured them in a hazard schedule of combinations of hazards (Combined hazards topic report, Ref. 71).

558. I assessed this document and I noted the following:
    - The RP identified that internal fire can initiate an internal missile but only where the source of the missile is electrical equipment due to an over-speed. The RP identified a number of rooms and qualitatively concluded that the effects of missiles are bounded by the effects of fires and therefore an internal missile initiating consequentially to an internal fire are no more onerous than the fire.

- The RP qualitatively concluded that the combined consequences of internal fires, missiles, explosions and flooding do not result in the loss of all divisions of Class 1 SSC delivering Category A safety functions.
- The RP identified flooding causing missile, but qualitatively bound the consequences of the internal missile by either the flooding or fire event and concluded that there are no combined consequences.
- Internal missile could initiate flooding. The RP identified a number of rooms where a missile from rotating equipment could cause flooding and qualitatively bound the consequences by other flooding events.
- The RP identified a number of cases where the flood volume exceeds those analysed in the flooding Topic Report, but qualitatively bound a number of cases by the missile effect itself. In addition the RP identified a number of rooms where potential combined consequential impact associated with flooding and missiles exists; however the RP qualitatively bound the consequences by the flooding assessment.
- Missiles causing pressure part failure was qualitatively bounded by the missile consequences in the room.
- The consequences of a dropped load caused by a missile bound the missile hazard.
- Internal missile caused by a seismic event were also qualitatively discussed.
- Internal explosions are prevented within the NI and therefore explosion as an initiator for combined consequential internal hazards such as missile has been dismissed. This is further discussed in Section 7.5 above.
- Turbine disintegrations causing flooding has been considered and qualitatively dismissed. The potential combined consequences of other internal hazards such as fire have not been considered.

559. While Reference 71 presents a comprehensive identification of consequential, correlated or independent events, the justification for dismissal from further analysis as well as the consequential analysis presented is not supported by the requisite arguments or evidence. In particular the combined impact on the barriers was not presented. See assessment finding CP-AF-AP1000-IH-02 raised earlier in this report.

8.3.4 ASSESSMENT OF GI-AP1000-IH-05 – INTERNAL MISSILE SAFETY CASE SUBSTANTIATION

8.3.5 ASSESSMENT OF PR’S INTERNAL MISSILE DESIGN CRITERIA

560. The design criteria for the protection from internal missiles was initially captured in the design criteria for the protection from internally generated missiles (Ref. 154) and in a draft internal missile methodology (Ref. 157).

561. I assessed these documents and raised regulatory queries RQ-AP1000-1440, RQ-AP1000-1507 and RQ-AP1000-1607 to obtain clarity on the methodology, criteria, formulas and assumptions used in the analysis including the degree of conservatism and margins of safety (Refs 37, 41 and 49).

562. In response to the RQs the RP was required to update the analysis criteria to meet my expectations. Reference 158 captures the key changes between the standard **AP1000** plant and the UK **AP1000** plant internal missile design criteria. This include the following:

- Internal missiles are deterministically assumed to occur as a result of gross failure of safety and non-safety related turbines, rotating components and

pressurised components such as valves and tanks, except those justified by the structural integrity classification as HSS.
- Low, medium and high energy systems are capable of creating missiles.
- Rotating components that operate less than 2% of the time are considered credible sources of missiles. This includes the examination of motors on valve operators and pumps in systems that operate infrequently, such as the chemical and volume control makeup pumps.

563. The RP responded to all my queries and submitted '**AP1000** Nuclear island missile penetration calculation' (Ref. 159), which presents the updated analysis methodology for internal missiles. This has been reflected in the internal missiles Topic Report (revisions 0 and 1 are Refs. 74 and 75).

564. The revised analysis methodology includes the following steps (**AP1000** nuclear island missile penetration calculation, Ref. 159):

- Identification of potentially internal missile sources:
  - turbine disintegration;
  - rotating components (such as pumps, fans and their motors);
  - pressurised components (such as tanks, valves, and their components).
- Categorisation of the failure based on the equipment present in the each room (rooms Type I to Type III);
- Penetration analysis for each source of missile and for each category of rooms;
- Integrity calculation for the missile shield/barrier – penetration of reinforced concrete walls can be predicted.

565. References 75 lists all valves, pressure vessels and rotating equipment analysed. Reference 168 also gives a summary of the rotating equipment internal missile sources.

566. In the integrity calculations for the missile shield/barrier, spalling and scabbing effects are not mentioned. However, I do not consider this to be an area for concern as the RP assumes that all SSCs in the area of a missile generator are lost.

567. I am satisfied with the methodology used for the selection of missile sources and integrity calculations undertaken.

568. Overall I am satisfied with the approach set out in the Topic Report. This is in line with my expectations and ONR's SAPs EHA.3, EHA.6 and EHA.14.

8.3.6 ASSESSMENT OF MISSILE BARRIER

569. The evidence presented for the missile barriers is underpinned by the Nuclear Island Missile Penetration Calculation (Ref. 159), the **AP1000** Turbine Missile Assessment (Ref. 160) and the detailed description of the approach and results in the internal missiles Topic Report (Ref. 75).

570. Concrete penetrations depths generated by internal missiles from valves, tanks and rotating equipment are given in 'UK **AP1000** turbine missile assessment' (Ref. 160). This output has been used to validate that the barriers could withstand internal missiles.

571. The internal missiles Topic Report (Ref. 75) provides for each room the minimum barrier thickness, the penetration depth by the missiles and whether the barrier will survive the missile.

572. The sections below summarise my assessment.

Inside Containment

573. I focussed my assessment on the following areas:

Reactor Coolant Pumps (RCP)

574. Inside Containment, I focused my assessment on the RCP. An RCP casing failure could result in the failure of an SG vertical support column.

575. There are four reactor coolant pumps, two per steam generator. In the event of a reactor coolant pump failure, the RP acknowledges that a missile could be generated weighing in the region of 1,400 kg.

576. The RP submitted Reference 163 in the area of structural integrity. This document provides an assessment on the structural integrity classification for the RCP due to the potential impact from a catastrophic failure of the pump bowl casing on the effectiveness of the vertical support for the SG. This is relevant to GDA Issue GI-AP1000-SI-06.A3 and assessed in Reference 164.

577. In order to support the demonstration of an ALARP design for the structural integrity safety case, the RP initiated a modification to the Steam Generator Vertical Column Support, which is captured in Reference 165.

578. The RP applied the 'Hagg and Sankey' methodology to analyse the effect of missiles on the Containment heat shield part of the pressure boundary following the postulating catastrophic failure of the flywheel to show that fragments from a failed flywheel would be contained within the Containment heat shield. This analysis is part of GI-AP1000-SI-03.A2 and is assessed in References 161 and 162.

579. I queried the introduction of RCP fragments into the primary circuit following disintegration (Regulatory query RQ-AP1000-1793, Ref. 62). The RP in response confirmed that the design requirements for the RCPs are established to ensure that any failure of the rotating parts would be retained within the RCP casing at specified over-speed conditions. Flywheel disintegration evaluations have substantiated that the impeller fragments would be contained within the Containment heat shield. I am satisfied with the work undertaken in this area.

Room 11202 – SG2 Loop Compartment

580. To satisfy myself that the safety case presented is coherent and that the hazard schedule of the Topic Report is complete, I sampled Room 11202 within Containment. Room 11202 is a 'Type 2 - *A room that contains SSCs delivering Category A safety function or the post 72-hour Category B safety functions and a potential missile source*'.

581. Sources of missile in Room 11202 include:

- Compressed and Instrument Air System (CAS);
- Reactor Coolant System (RCS);
- Liquid Radwaste System (WLS).

582. This fault has the potential to impact the RCS and the SGS.

583. With regards to the missile sources there are five valves present in Room 11202, which could potentially lead to the generation of missiles.

584. Calculations have been carried out for the bounding missile source valve PV03-Z0D-175 (bounding based on the mass of the valve), which results in a penetration depth of 0.2935 inches. The calculation conservatively assumes the following (**AP1000** Nuclear island missile penetration calculation, Ref. 159):

- 1.5 times the connecting pipe nominal pipe diameter as the cross-sectional area of the valve stem;
- the internal length of all valve stems is assumed as 1ft and the internal length of the valve stem is used to calculate the total work done during ejection of the valve stem missile;
- the entire mass of the valve is used as the mass of the missile.

585. There is no claim on the ceiling of Room 11202 as it is open. Missiles could potentially affect Room 11302 above. The hazard schedule makes claims on the walls surrounding the missile sources and redundant equipment in Rooms 11201 and 11301. These rooms house identical SSCs to the ones that could be lost and which will, therefore, deliver the Category A functions.

586. I liaised with ONR civil engineering inspector (Ref. 166) who assessed the civil engineering aspects of the case presented in the **AP1000** nuclear island missile penetration calculation (Ref. 159). It was concluded that the guidance, empirical formula and sources used by the RP are considered as some of many sources of RGP in the UK when assessing impact loading, blast loading and various concrete impact mechanisms.

587. From the analysis of the room, which I sampled, I am therefore satisfied with the claims on civil engineering structures presented in the case submitted by the RP, and with the RP's approach to missiles in this area.

588. Overall, my sample has given me confidence that the case made by the RP is in line with ONR's SAPs EHA.3, EHA.6 and EHA.14.

Outside of Containment

589. I focused my assessment on the following areas:

Turbine Disintegration

590. The turbine generator presents one of the most hazardous items due to its size and rotational speed. Such a missile has the potential to impart significant energy into the turbine casing which on perforation of the casing, could impact SSCs beyond the turbine.

591. During my assessment of turbine missile case, I raised regulatory queries RQ-AP1000-1440, RQ-AP1000-1444 and RQ-AP1000-1607 to gain clarity on the following (Refs 37, 39 and 49):

- the assumed missile target angle;
- building, structures and SSCs in the missile strike zone;
- capability of structures to withstand the missile impact;
- degree of conservatism and sensitivity analysis undertaken.

592. The RP responded to all my queries and issued Reference 160, which presents the turbine missile assessment. This document assessed the turbine rotor disk failure causing turbine missiles and its effects on the **AP1000** plant NI barriers.

593. The method utilised by the RP includes the following stages (Refs 75 and 160):

- Firstly evaluating the kinetic energy of a quarter-disk fragment at design and overspeed conditions to determine the resistance the casing has against perforation.
- In the first aspect of the analysis the RP uses the 'Hagg and Sankey' method to evaluate the effect of Low Pressure Turbine (LPT) last stage disk rupture. The impact energy of a quarter disk fragment has been considered and substantiated as bounding.
- Secondary analysis is then carried out, which evaluates the missile's residual velocity after casing exit.

  - After penetrating the casing, the residual velocity of the missile is calculated at 100%, 120% and 180% of the rated synchronous speed of 1500 rpm at 50Hz. Gross failure has been assumed for all three cases. The three speed cases are analysed for impact at trajectory angles of 25^o, 45^o and 90^o from the plane of rotation, using the 'R3'analysis methodology.
  - The missile velocity is then used to validate the barrier integrity of Wall 11.2, which is the second line of defence between the missile and the nearest Class 1 SSC located in the Auxiliary Building. It was shown that a missile will not penetrate Wall 11.2 and so the principal barrier is still intact.
  - The RP demonstrated that turbine missiles have an insufficient line of sight through all credible paths through the Turbine First Bay opening in Wall 11.2 and into the Auxiliary Building.

594. The analysis carried out by the RP determined the following:

- the amount of kinetic energy generated that must be absorbed by the casing for non-perforation;
- fragment speed calculation;
- fragment energy;
- residual velocity;
- the calculated residual velocities are then inputs to the missile trajectory equation;
- concrete penetration (with outer casing considered);
- concrete penetration (with no outer casing).

595. In addition to the calculations carried out by the RP, further confidence can be obtained by the fact that the area most vulnerable to turbine missiles is along the axis of the turbine. This is not in the missile strike zone for the majority of the potential missiles that could be produced following turbine disintegration.

596. No claim on the casing has been made; however the casing would provide some degree of mitigation and, therefore, this is a defence-in-depth provision.

597. I challenged the RP on the mitigation role provided by the casing and the ability of the barriers to withstand the turbine missile without the casing. The RP in response undertook an analysis to validate the integrity of Wall 11.2 without the protective casing and at full speed. The RP confirmed that the integrity of Wall 11 is maintained even without the protective casing in place. This sensitivity analysis satisfied SAP AV.6.

598. I queried the applicability of the 'Hagg and Sankey' methodology to turbine disintegration and its comparison with relevant good practice. The RP provided the following clarification (response to regulatory query RQ-AP1000-1793, Ref. 62):

599. ‘*Relevant good practice along with the ‘R3’ methodology indicate that the prediction capabilities of the ‘Hagg and Sankey’ procedure is acceptable for failure of rotating components leading to the ejection of fragments as missiles from the casing. The ‘R3’, Volume 1 5-3, Section 5.1.4.2 references the ‘Hagg and Sankey’ method as a semi-empirical approach to analysing the containment of missiles. Once the missile is ejected from the casing, trajectory equations from the ‘R3’ are used to evaluate the distance the missile fragment travels after casing perforation, at angles of 25^o, 45^o and 90^o from the plane of rotation*.’

600. I am, therefore, satisfied that the deterministic approach is suitable and in line with ONR’s SAPs EHA.3, EHA.6 and EHA.14.

601. With regards to impact on the concrete wall separating the Turbine Building from the Auxiliary Building, I liaised with ONR’s civil engineering inspector who provided the following judgement (Ref 167).

602. ‘*The guidance document used by the RP is considered as one of many sources of RGP in the UK when assessing impact loading, blast loading and various concrete impact mechanisms. On this basis I judge that the work undertaken to assess the effects of turbine missiles on the reinforced concrete Walls 11 and 11.2 to be adequate. The RP have demonstrated that there is conservatism built into their assessment, compliance with RGP and hence that their approach is ALARP*’.

603. In addition to the above the potential for missiles to be generated from the turbine is minimised by the provision of the following:

- Turbine overspeed protection systems incorporated into the design.
- Adherence to applicable design codes, standards and criteria. Turbine rotor integrity is provided by the integrated combination of material selection, rotor design, fracture toughness requirements, tests, and inspections.
- Turbine orientation. The turbine axis of its shaft rotation is perpendicular to the NI.

604. Overall, I am satisfied with the turbine disintegration analysis undertaken by the RP and the conclusions reached. The analysis undertaken is largely in line with ONR’s SAPs AV.1, AV.6, EHA.1, EHA.14, EKP.3, EKP.4 and EKP.5.

Pressure Vessel Missile

605. The Topic Report lists all pressure vessel components considered as potential missile sources in the **AP1000** plant. It also presents a demonstration that the minimum barrier thickness is adequate to withstand the bounding penetration depths.

606. To satisfy myself that the RP adequately considered missiles generated from pressure vessels and that the provisions in place against missile are suitable and sufficient, I sampled Room 12155. This is the gaseous radwaste (WGS) equipment room outside of the Containment. There are a number of potential missile sources in this room such as the Compressed and Instrument Air System (CAS), Demineralised Water Transfer and Storage System (DWS), FPS, Plant Gas System (PGS), Central Chilled Water System (VWS), and Gaseous Radwaste System (WGS).

607. According to the hazard schedule, there are no essential SSCs in this room and it is therefore classed as a ‘Type III’ room (room that contains Class 2/3 equipment with potential missiles).

608. This purpose of this sample is to ensure that the RP considered the potential effect of missiles from this vessel on other rooms housing SSCs delivering Category A safety functions or post 72-hour Category B safety functions.

609. Pressure vessel ‘WGS-ME-01’ has been selected as the bounding vessel as the calculated penetration depth is 38 mm. This is considerably less than the wall, ceiling and floor thickness, the most penalising being the south wall, which is 457mm thick.

610. The calculation of the missiles generated following the failure of a pressure vessel is based on the entire dry tank, which is suitably conservative.

611. The RP also assumed that all stored energy is converted into kinetic energy. The change of internal energy into kinetic energy is captured by assuming an isentropic expansion of the fluid from its operating pressure to standard atmosphere, represented by total change in enthalpy.

612. By assuming an isentropic process, no energy will be lost by heat transfer or friction, ensuring a conservative calculation.

613. The RP confirmed that the missile shape is assumed to be a sphere, but based on the mass of the entire tank.

614. While this is reasonable, the analysis did not address the potential impact on doors or any other penetrations in Room 12155. If doors are weak points in missile barrier, they do create a potential route to damage SSCs delivering Category A functions in adjacent rooms.

615. I queried this with the RP who confirmed that ‘*If a door is required to protect safety-related equipment from a postulated missile, it would result in a Class 1 classification. There are no line-of-sight of missiles that need credited, since the* ***AP1000*** *contains redundant and diverse equipment which are protected by concrete walls. Thus, there are no safety-related doors for missile protection*’, (Regulatory query RQ-AP1000-1793, Ref. 62).

616. I explored further the arguments on the bounding calculations and sampled Room 12555 – Main Control Room Emergency Habitability System (VES) air storage, which is adjacent to Room 12556 Operating Deck Staging Area. There is a door on the barrier between Rooms 12555 and 12556. Room 12555 is a ‘Type II’ room; a room that contains both Class 1 and Class 2/3 equipment and contains potential missiles. The Topic Report lists the minimum barrier thickness as 228.6 mm whereas the bounding pressure vessel penetration depth is 225.35 mm.

617. I queried with the RP the margins of safety available, the sensitivity analysis undertaken and the classification of the door between Rooms 12555 and 12556 (Regulatory query RQ-AP1000-1793, Ref. 62).

618. The RP in response explained the following:

- The bounding penetration depth in Room 12555 is based on a missile from the VES tank (VES-MS-01/04). The VES missile weight is based on a total tank mass of 2,835 kg. This is conservative as it yields the highest possible missile kinetic energy.
- Some sensitivity analyses have been conducted and referred to an example where a missile 1/10th the size of the VES missile in Room 12555 would have a penetration depth of approximately 178 mm, less than the 228.6 mm concluded in the final assessment.

- Room 12556 is a Type I Room '*a room that contains either SSCs delivering the Category A safety function or the post 72-hour Category with no potential missiles, or non-safety equipment with no potential missiles'*. There are no Class 1 SSCs in Room 12556. If a missile were to travel through the door from Room 12555 to 12556, it would not result in the loss of Category A or B safety function outside of Room 12555 and 12556. No claim on the door has been made.

619. While these assumptions are overall reasonable and conservative, the RP did not consider a smaller missile, which at higher kinetic energy could present a more onerous missile and which could challenge a claimed barrier against missiles (in line with ONR's SAPs AV.6 and FA.7). However, given the margins available from the bounding analysis undertaken an acceptable level of confidence can be drawn. This is also supported by the example given above for Room 12555.

8.4 ASSESSMENT FINDINGS

620. There are no assessment findings identified from my assessment of internal missiles.

8.5 MINOR SHORTFALLS

621. There are no minor shortfalls identified from my assessment of internal missiles.

8.6 GDA ISSUE GI-AP1000-IH-05 CONCLUSIONS

622. The submission provides the requisite information relating to the identification of internal missiles, consequences analysis and the identification of safety measures. Suitable and sufficient claims have been made and these were supported by the requisite arguments and evidence. Although my assessment has been based upon a sample, I am satisfied that during the GDA, internal missiles have been subjected to a detailed review and substantiation.

623. I am therefore satisfied that GDA Issue GI-AP1000-IH-05 can be closed.

9. ONR ASSESSMENT OF GDA ISSUE GI-AP1000-IH-06 – DROPPED LOADS

9.1 RESOLUTION PLAN ACTIONS AND DELIVERABLES

624. During GDA Step 4, a GDA issue was raised relevant to substantiation including supporting analysis of the consequences of dropped load and impact from lifting equipment included in the **AP1000** design (GI-AP1000-IH-06) (ONR's Step 4 Internal Hazards Assessment Report, Ref. 13). This included the following GDA action:

- GI-AP1000-IH-06.A1: Substantiation and analysis of the consequences of dropped loads and impact from lifting equipment included within the **AP1000** design.

625. In order to address the two actions, it was prudent for the RP to re-present the claims and arguments for the dropped load case in the PCSR. These are covered in the specific Topic Report and in the revised PCSR.

626. The RP's Resolution Plan (Ref. 23) identifies specific deliverables associated with the above action:

- Dropped Load Roadmap;
- Load Path Layout Assessment;
- Dropped Load Exclusion Assessment;
- Dropped Load Topic Report.

627. The Dropped Load Roadmap (Ref. 171) was initially issued to show the link between the existing claims, arguments and evidence. The document has been subsequently superseded by the Dropped Load Topic Report (Ref. 80).

628. In addition to the list above, the RP issued a number of draft documents, multiple revisions of the Topic Report and PCSR and a number of supporting documents.

629. During this phase of the GDA I raised regulatory query RQ-AP1000-1306 aiming to seek clarity on the scope of submissions and the timescales given in the Resolution Plan (Ref. 29).

630. In the following sub-sections, I will cover the following:

- the RP's safety case on dropped loads;
- my assessment of GI-AP1000-IH-06, which includes:
    - assessment of claims and arguments including combined consequential hazards;
    - assessment of substantiation of the claims which includes assessment of analysis methodology.
- conclusions and assessment finding.

9.2 REQUESTING PARTY'S SAFETY CASE

9.2.1 AP1000 REACTOR DROPPED LOAD CLAIMS

631. Key document submission for dropped loads are:

- UKP-GW-GL-793, Revision 0D – **AP1000** Pre-construction Safety Report, Chapter 11 – Internal Hazards (Ref.69);

- UKP-GW-GLR-110, Revision 1 – UK **AP1000** Internal Hazards Topic Report – Dropped Loads (Ref. 80).

632. The **AP1000** reactor dropped load safety case uses the following approaches to ensure that the Class 1 SSCs will continue to provide their Category A safety functions (PCSR Chapter 11 and dropped loads topic report, Refs. 69 and 80):

- Provision of redundant SSC, capable of delivering the Category A safety function, in segregated locations which cannot be affected by a single dropped load hazard.
- Protection of SSC, whose availability is required to deliver Category A safety functions, from the effects of dropped loads. This protection takes the form of a dropped load withstand capability, which is demonstrated for certain key structures, and the provision of passive safety features.
- Qualification of SSC to withstand the secondary effects of a dropped load such as vibration.

633. Specific internal dropped load claims are (Refs. 69 and 80):

- **Claim IH-7.0:** Postulated dropped loads within the design basis will not prevent the delivery of the Category A safety functions and supporting post 72 hour Category B safety functions necessary to respond to the postulated event.
- **Claim IH-7.1.1:** A dropped load on the reactor vessel will not prevent adequate cooling of the core.
- **Claim IH-7.1.2:** The floors of the SFP will not fail in the event of a dropped load onto the SFP.
- **Claim IH-7.1.3:** A dropped load from the Cask Handling Crane will not impact the SFP.
- **Claim IH-7.2.1:** A single dropped load event cannot impact all divisions of SSCs delivering a single Category A safety function.
- **Claim IH-7.2.2:** Class 1 SSC, not directly impacted by a dropped load, will be capable of delivering their Category A safety function.
- **Claim IH-7.2.3:** There will be no dropped loads within Containment in Modes 1 to 4.

634. The RP identified the following SSCs and administrative procedures to deliver the above claims:

- withstand impact capability of the Reactor Pressure Vessel (RPV) and associated supports and connections from the Integrated Head Package (IHP) dropped load (claim IH-7.1.1);
- the SFP liner can withstand the bounding drop at the capacity weight of the Fuel Handling Machine and at the maximum hook height (claim IH-7.1.2);
- passive end-stops on the Cask Handling Crane rails (claim IH-7.1.3);
- separation and segregation of the Class 1 SSCs within Containment and outside Containment (claim IH-7.2.1);
- qualification of Class 1 SSCs against vibrations following a dropped load (claim IH-7.2.2);
- administrative controlled procedures to prohibit the use of lifting devices within containment in modes 1 to 4 (claim IH-7.2.3).

635. A dropped load is deterministically assumed to occur as a result of a failure of a lifting device. There are therefore no specific claims made on the prevention of dropped load hazards. The cranes and lifting devices have been designed to appropriate standards and assessed by ONR mechanical engineering inspector during Step 4 of the GDA (Step 4 Mechanical engineering assessment report, Ref. 177).

9.3 ASSESSMENT OF GDA ISSUE GI-AP1000-IH-06

9.3.1 SCOPE OF THE ASSESSMENT

636. The assessment strategy in Section 2 was used to formulate the scope.

637. My assessment covers the deliverables used in addressing the action GI-AP1000-IH-06.A1. I assessed the Topic Report and the PCSR on dropped load, and I also sampled supporting documents to obtain confidence on the requisite evidence and substantiation of the claims made.

638. The areas chosen to review the dropped load safety case were limited to:

- inside Containment.
- outside Containment – Mainly Auxiliary Building.

639. The sections below cover the areas of my assessment.

9.3.2 ASSESSMENT OF CLAIMS AND ARGUMENTS

640. The RP undertook a systematic identification of dropped load events and consequences analysis which culminated in the derivation of specific claims.

641. In an early submission of the dropped loads Topic Report (Ref. 78 and 79) the overall presentation of the safety case was not cohesive or coherent and the arguments presented were not supported by the hazard schedule. This was not in line with my expectations (ONR’s SAP SC.4). In addition, the PCSR Chapter 11 (Ref. 68) included different claims than those listed in the Topic Report, for example, the flood doors for Rooms 12166 and 12167.

642. I challenged the RP and raised a number of queries as listed in regulatory query RQ-AP1000-1792 (Ref. 63).

643. In response to the RQ- AP1000-1792 the RP updated the dropped load Topic Report and PCSR Chapter 11, and presented revised claims (Refs. 68 and 80, respectively).

644. The RP considered dropped load from each of the lifting devices within the NI, and demonstrated that a single dropped load cannot directly impact all divisions of SSCs delivering a single Category A safety function. This is discussed below.

Inside Containment

645. Within the Containment, redundant divisions of Class 1 SSC have been separated by distance as far as is reasonably practicable.

646. The Polar Crane presents the largest hook coverage area, encompassing almost the entirety of Containment, and load capacity. The Polar Crane is only required to be used for lifting operations during plant operations modes 5 and 6 (cold shutdown and refuelling, respectively).

647. The RP undertook a review of the layout of SSCs within the Polar Crane hook coverage. The RP’s review utilised ‘Drop Zones ‘A’ to ‘D’ centred on the reactor vessel centreline to divide the footprint of the Polar Crane.

648. The RP identified the IHP as the largest single component lifted in the Containment. This presents the bounding scenario inside Containment. The RP concluded that that while a drop load of that magnitude will cause significant damage to structures and components, under the Polar Crane hook coverage

the footprint of the IHP is not large enough to damage multiple divisions of Class 1 SSCs during mode 5 and 6.

649. The RP concluded that no more than two out of the four divisions could be impacted by a drop of the IHP. Either of the two divisions, which are unaffected by the load drop are capable of delivering the Category A safety function. Therefore, redundant SSCs in other zones are available to satisfy claim IH-7.2.1.

650. The Topic Report lists all rooms and SSCs for each zone ('A' to 'D') that can be impacted by a dropped load from the Polar Crane, whereas the hazard schedule presents the safety measures in place and the redundant SSCs available. This is in line with my expectations.

651. In addition to the argument presented above, the RP defined a specific safe load path for the IHP based on the dimension of the IHP and the structures inside Containment (Refs 178 and 179). During plant operation modes 5 and 6, the SSCs with Category A safety functions that could be affected by a dropped load are the In-Containment Refuelling Water Storage Tank (IRWST) and the Reactor Pressure Vessel (RPV).

652. The RP argued that during modes 5 alternative water supplies would be available from the CMT and additional cooling, if required, would be available from the SFS, RNS or Passive Containment Cooling System (PCS) lines. Failure of the IRWST, in mode 6, when the refuelling cavity is flooded, would result in more water within the cavity, but flooding of the IRWST does not represent the bounding flood event. Flooding is covered in the section further below. Therefore, such an event will not prevent delivery of the Category A safety functions during modes 5 and 6.

653. The IHP drop from the Polar Crane on the RPV has been identified as the bounding dropped load event and quantitatively analysed. It was conducted in Reference 180 that the RPV and associated supports and connections will withstand the impact from the dropped load, see further below. This analysis provides the requisite evidence for claim IH-7.1.1.

654. I also sampled the claims made relevant to Equipment and Maintenance Hatch hoists which are used only in plant modes 5 and 6.

655. The Equipment Hatch hoist is located on the Operating Deck. Its hook coverage covers a portion of zone 'C' within Containment. A dropped load could affect the RNS suction line penetration and the PXS Core Makeup Tank 'B'. The RP argued that the Class 1 SSCs associated with the RNS system are not required for safe shutdown, as the PXS is the primary means for providing this function. The RP identified redundant SSCs including isolation valves outside Containment (Auxiliary Building) and the provision of the Core Makeup tank 'A' located in zone 'D'.

656. The Maintenance Hatch hoist is located at the Maintenance floor. Its hook coverage covers a portion of zone 'D'. A dropped load could affect the ECS electrical penetrations. However, during modes 5 and 6 the Main AC Power System (ECS) electrical penetration to power the Reactor Coolant Pumps is not required.

657. In regulatory query RQ-AP1000-1792 (Ref. 63), I also queried the suitability of separation of the two PXS Valve Rooms inside Containment. The RP in response argued the following.

658. The Maintenance Hatch and PXS Valve Room 'B' are both located in the South East quadrant (zone 'D'), while the Equipment Hatch and PXS valve room 'A' are both located in the north east quadrant of the Containment (zone 'A'). The two quadrants are separated mostly by the civil structure surrounding Steam Generator 2 and its associated components. At their closest approach, the two PXS Valve Rooms are 6.7 m apart, while both Hatches have a diameter of 5.2m. Therefore, it is unlikely for the Equipment Hatch to impact both rooms. This also satisfies claim IH 7.2.1.

Outside Containment

659. Outside the Containment, I focused my assessment on the claims made for the Cask Handling Crane. The function of the Cask Handling Crane is to move fuel casks between the Rail Car/ Truck Bay (12371), the Cask Loading Pit (12463) and the Cask Washdown Pit (12462).

660. The bounding dropped load from the Cask Handling Crane is 136 tonnes over 14m high on Room 12463 (Cask Loading Pit). This is the bounding scenario for all dropped loads outside Containment.

661. The RP identified a number of dropped zones ('A' to 'F') and listed all relevant rooms under the hook coverage of the Cask Handling Crane that could be impacted. The RP conservatively assumed that a dropped Spent Fuel Cask could penetrate the floors of the levels below.

662. Such a dropped load could affect the FHS, SFS and RNS systems which could lead to complete loss of CLP water inventory, complete loss of Cask Washdown Pit water inventory, loss of SFP water to EL 109'- 6", and RNS LOCA.

663. The RP identified the following safety measures in the Topic Report and hazard schedule (Ref. 80):

- Class 1 passive end-stops (claim IH-7.1.3).
- A dropped load in zones 'C', 'D' and 'E' could affect Class 1 SSCs associated with the RNS system; however the PXS is the primary means of providing the safety function which will remain unaffected.
- Containment isolation valves, to those located in Room 12365, are available inside Containment.
- A dropped load in zones 'A' or 'B' could affect two out of three legs of the low SFP level decision logic. However, the PMS would initiate refuelling cavity isolation.
- Connections to SFP are made at an elevation that precludes the possibility of inadvertently draining the water in the pool to an unacceptable level.

664. The RP therefore demonstrated that suitable and sufficient SSCs are available to deliver the Category A safety function following a single dropped load (IH 7.2.1).

665. The lifting devices which operate above the Fuel Handling Area are the Fuel Handling Machine, the Cask Handling Crane and the New Fuel Elevator.

666. The passive end-stops on the Cask Handling Crane rails are integrated into the rail's concrete support structure which restricts movement of the crane in the westerly direction such that the Spent Fuel Cask cannot be handled over the SFP (12563) or its walls.

667. The design of the Spent Fuel Cask will be completed post GDA. I queried the location of the end-stops to obtain some confidence on the claim IH-7.1.3 made. The RP, based on an indicative diameter of a Spent Fuel Cask of 2.44 m,

explained that the location of the end-stops is such that the centreline of the Cask Handling Crane hoist is 4.57 m from the SFP. Therefore, a Spent Fuel Cask drop will not impact the SFP, in line with claim IH-7.1.3. This will require confirmation post GDA and during detailed design and procurement of the Spent Fuel Cask. I have confidence that the licensee shall be able to confirm this matter.

668. With regard to claim IH-7.1.2, the RP undertook a bounding analysis (a dropped fuel assembly with a control rod assembly and a handling tool attached from the Fuel Handling Machine) and concluded that the SFP liner will not be penetrated (UK **AP1000** Fuel assembly drop accident report, Ref. 181). This analysis provides the requisite evidence to support claim IH-7.1.2. This is discussed further below.

669. The RP considered the consequential effects of vibration caused by dropped loads. Class 1 SSCs are seismically qualified to Seismic Category C-1 and will maintain both functionality and integrity under seismic loading within the design basis.

670. The seismic design criteria include the effects of vibration, which bounds those associated with a dropped load. I raised regulatory query RQ-AP1000-1675 (Ref. 53) to gain clarity on the vibration effects on Class 1 SSCs.

671. The RP indicated that the vibration effects will be lessened and dissipated due to the local damage of the concrete floor and yielding of the reinforcement. The walls will also limit the vibration effects. Therefore, the impact on SSCs in adjacent rooms, due to vibration, would be limited. This argument satisfies claim IH 7.2.2.

672. I acknowledge that the control and operation of lifting equipment on the **AP1000** plant will be carried out in accordance with the licensee's written process and procedures and statutory legislation. This will be addressed post GDA.

673. The RP identified a number of operating restrictions which limit the use of lifting devices within Containment in modes 1 to 4, including:

- There is no personnel access to Containment in modes 1 to 4 except for infrequent inspections.
- Power supplies to the Polar Crane are isolated and locked off during modes 1 to 4 to prevent spurious activation.
- The Refuelling Machine Control console is removed from Containment during plant operation to protect it from the effects of radiation.
- Jib cranes are parked when not in use and are locked to prevent movement
- Equipment and Maintenance Hatch hoists are operated from a wall-mounted pushbutton station inside Containment. Power to the hoists is isolated by a switch which is locked off when not in use. The hoists are permanently attached to the hatches which are bolted closed during modes 1 to 4.

674. Overall, I am satisfied that the identified claims and arguments are suitable and sufficient. The Topic Report and hazard schedule identify all lifting devices, SSCs and safety functions at risk and lists all the claims made. This is in line with my expectations and ONR's SAPs EHA.1, EKP.5 and ECS.2.

9.3.3 COMBINED CONSEQUENTIAL HAZARDS

675. Combined events and their associated combined consequential loads have the potential to compromise the safety measures in place against dropped loads.

676. The RP undertook a study to identify all credible consequential, correlated or independent hazards relevant to dropped loads and captured them in a hazard schedule of combinations of hazards (Combined hazards topic report, Ref. 71).

677. The RP assumed that a dropped load could consequentially initiate an internal flood and/or pressure part failure.

678. With regard to the pressure part failure, the consequences are confined to the room and bounded by the dropped load, which assumes complete loss of all SSCs in the affected room.

679. With regard to consequential flooding, three dropped loads hazards have been considered: a dropped load inside Containment; a dropped load in the non-RCA side of the Auxiliary Building and a dropped load in the RCA side of the Auxiliary Building.

680. Within Containment, the combined consequential effects are those associated with a drop from the Polar Crane, as described by FL1 to FL4 in the hazard schedule. Within Containment, a dropped load cannot result in the loss of all divisions of Class 1 SSCs as all Class 1 SSCs required for safe shutdown are qualified for the effects of flooding.

681. In the RCA side of the Auxiliary Building, the combined consequential effects are those associated with a drop from the Cask Handling Crane into the Cask Washdown Pit (12462) and flooding as described by FL21 in the combined hazard schedule (Combined hazards topic report, Ref. 71). The operators in the MCR will respond to the flood-up alarms, which are not affected by dropped loads evaluated in the RCA side of the Auxiliary Building.

682. In the non-RCA side of the Auxiliary Building a drop from the MSIV monorail hoist in 12404 ‘B’ would impact the FPS piping in 12306 before impacting division ‘B’ of the IDS batteries in 12104. During modes 5 and 6, the FPS is aligned with the fire water storage tank and the maximum flood height on level 1 of the non-RCA would exceed that for rupture of the FPS system during modes 1 to 4.

683. The RP proactively raised an outstanding issue to review the configuration of the FPS on the non-RCA side during modes 5 and 6 such that the available inventory from the FPS does not exceed that from the PCCWST during modes 1 to 4. This is captured in the Design Reference Point document (Ref. 64) via ‘UK AP1000 Internal Hazards Design Change Proposal’ APP-GW-GEE-5401 ‘Limitation of FPS Supply to Non-RCA Auxiliary Building’.

684. The effects of flooding are further considered in Section 5 of this report.

685. The RP also considered consequential effects from seismic induced dropped loads.

686. The RP concluded that the Category A safety functions will continue to be delivered following various design basis combined hazards. However, in Section 4 of this report I raised an assessment finding CP-AF-AP1000-IH-02.

9.4 ASSESSMENT OF GI-AP1000-IH-06 – SUBSTANTIATION OF CLAIM

9.4.1 DROPPED LOAD ANALYSIS METHODOLOGY

687. The RP undertook a deterministic dropped load consequences analysis which included the following (Dropped loads topic report, Ref. 80):

- identification of all cranes and lifting devices;
- identification of plant areas that could be affected;
- identification of modes of operation;
- identification of safe load paths;
- identification of bounding dropped loads;
- consequences analysis.

688. The RP in support of the above undertook the following studies:

- They have provided a ‘load path assessment’ report (Ref. 179). This identifies all cranes and lifting devices within the NI. It also identifies the expected load travel paths for lifts that avoid impacting SSC delivering Category A safety functions. The scope of the load path assessment report was to define operational areas within which the loads will be handled during normal operation. This document has been subsequently superseded by the latest revision of the dropped loads topic Report (Ref. 80).

  During my assessment, I acknowledged that at Step 4 of the GDA an Assessment Finding AF-AP1000-ME-23 has been raised which states that ‘*lifting plans will be established outlining the safe load path for each hoist. As such, the potential for a load to impact (either directly or following a dropped load) Class 1 SSC will be minimised. Furthermore, the lift plans will seek to minimise the height at which the load is carried and in turn the consequences following a drop*’ (Step 4 Mechanical engineering assessment report, Ref 177).

Therefore, I expect the load path assessment (Ref. 179) to be revised as part of resolving this assessment finding.

- A ‘handling equipment dropped load exclusions assessment’ (Ref. 182). This document summarises the methodology used and reasoning behind excluding handling equipment for further evaluation. It also presents the evaluation of the remaining lifting devices. This evaluation is based on the general arrangements drawings, safe load paths and the maximum rated loads dropped over the maximum hook height. This document has been subsequently superseded by the latest revision of the dropped loads topic report (Ref. 80).

689. The RP used the following criteria to identify lifting devices which do not need an in-depth evaluation:

- the hook coverage of the lifting device does not overlap any Class 1 SSC required to provide the principal means of delivering a Category A Safety Function;
- the lifting device has passive devices controlling the load path in a safe manner;
- the lifting device is unavailable during plant modes 1 to 4 and cannot lift a load over Class 1 SSC required to provide the principal means of delivering a Category A Safety Function during refuelling.

690. The RP also used the following conservative assumptions:

- a dropped load is assumed to penetrate all floors directly below;
- a dropped load may occur anywhere within the lifting device hook coverage;
- within Containment the maximum area of impact is equivalent to the footprint of the IHP;
- maximum hook height is assumed as the drop height, allowing for clearance over all SSCs under the hook coverage of the various lifting devices;

- outside Containment, all Class 1 SSCs contained in an affected room or zone lose functionality;
- swinging loads are bounded by the drop scenarios captured in the hazard schedule.

691. Overall, the above methodology, criteria and assumptions are in line with my expectations and relevant good practice established in the UK.

Inside Containment

692. Within the Containment, the following cranes and lifting devices have been considered by the RP:

- Polar Crane (Class 1) – used only in modes 5 and 6;
- Refuelling Machine (Class 2) – used only in mode 6;
- Equipment and Maintenance Hatch Hoists (Class 1) – used only in modes 5 and 6;
- Steam Generator Jib Crane – used only in modes 5 and 6.

693. Within Containment, the bounding dropped load was identified as a drop of the IHP from the Polar Crane. This scenario bounds a dropped load on the RPV from the Refuelling Machine. The SSCs under this load path that could be affected by a dropped load, during plant modes 5 and 6, are the IRWST and RPV. The consequences on IRWST are discussed above.

694. The dropping of the IHP onto the RPV would impose a large impact load onto the RPV which could damage the vessel, connecting pipework and supports. This could affect the ability to maintain sufficient cooling to the fuel in the reactor vessel.

695. The RP undertook a quantitative consequences analysis utilising Finite Element analysis (Reactor vessel head drop evaluation, Ref. 180). The responses of the RPV, RPV support assemblies, main loop piping, and direct injection piping were evaluated. The stresses and strains caused by the impact were evaluated to demonstrate acceptability, based on maintaining the structural integrity of the critical components for core cooling.

696. The RP concluded that the RPV, piping, RPV support assemblies and concrete support structure are capable of meeting the set criteria for a 4.88 m drop of a 226,796 kg of IHP onto the RPV flange. The RPV and associated supports and connections (RCS and DVI) would withstand the impact from the dropped load such that cooling of the reactor core can be maintained (Reactor vessel head drop evaluation, Ref. 180).

697. I subjected the Reactor vessel head drop evaluation to assessment and raised regulatory query RQ-AP1000-1675 (Ref. 53) to gain clarity on the sensitivity analysis and cliffe-edge effects on components that have reduced margins of safety.

698. I liaised with ONR's civil engineering inspector (Ref. 147) in the assessment of this document. Overall, the analysis and assessment undertaken, including the assumption used in the modelling, are conservative and the results are reasonable.

699. The RP therefore provided sufficient and suitable information to substantiate claim IH-7.1.1.

700. I am therefore satisfied with the selection of the bounding dropped load event and the detailed quantitative consequence analysis undertaken.

Outside Containment

701. Outside the Containment, the following cranes and lifting devices have been considered:

- In the Auxiliary Building the following:
  - Fuel Handling Machine (Class 2);
  - New Fuel Elevator (Class 2);
  - Cask Handling Crane (Class 1);
  - Rail Car / Truck Bay Crane;
  - Auxiliary Building Staging Area Monorail Hoist;
  - Filter Cask Portable Handling Device;
  - Closed Van Loading Scissor Lift (Portable).
  - MSIV Monorail Hoists;
  - Elevators and Lift Platforms.
- In the Shield Building Annulus the following:
  - Shield Building Annulus Personnel Basket and Baffle Plate Hoist.

702. The RP's overall approach, outside Containment, is that no more than one division of Class 1 SSC could be impacted by a single dropped load in any given area. The provision of redundant SSCs capable to deliver the Category A safety function, in segregated locations ensures that the ability to deliver Category A safety functions is unaffected.

703. The RP identified the drop of the Spent Fuel Cask from the Cask Handling Crane as a bounding dropped load inside the RCA side of the Auxiliary Building. The dropped load consequences of this scenario on the structures were not quantitatively assessed. However, delivery of Category A safety function is discussed above.

704. The RP assumed that all floors under the hook coverage would fail. I raised regulatory query RQ-AP1000-1675 (Ref. 53) to gain clarity on the defence-in-depth measures in place and on the impact on civil structures. The RP in response, articulated the following provisions:

- The likelihood of a drop from the Cask Handling Crane due to mechanical failure has been minimised through the application of a conservative design with appropriate safety factors.
- Abnormal operation, with the potential to result in an accident, is prevented by administrative and engineered controls, which provide indication and alarms to alert operators to the potential fault condition.
- Protection systems are provided, which will terminate all design basis fault progressions prior to escalation to an accident condition.
- Redundant divisions of Class 1 SSC are available to deliver Category A safety functions.

705. A number of these provisions however are relevant to the design of the Cask Handling Crane and, therefore, outside the scope of the internal hazards assessment.

706. As previously stated, the detailed design of the Spent Fuel Cask is out of scope of the GDA. The RP identified that the drop of a Spent Fuel Cask has the potential to result in the design basis for the Spent Fuel Cask being exceeded. The RP conservatively assumed that the floors will fail. Survivability of the floors could be critical in maintaining the design basis of the Spent Fuel Cask.

707. In response to RQ-AP1000-1675, the RP indicated that post GDA and during detailed design, a number of options could be considered including de-rating the lifting capacity of the Cask Handling Crane such that the impact energy can be tolerated by the floors, and/ or introducing impact limiters to maintain the integrity of the floors.

708. Given the significance of this dropped load on the civil structures and potentially on the Spent Fuel Cask, I raised the following assessment finding.

CP-AF-AP1000-IH-11 – The licensee shall carry out a quantitative dropped load assessment of the selected Spent Fuel Cask.

709. The RP also undertook a drop load consequences analysis of a fuel assembly (Ref. 181) (1,814 kg maximum load) handled by the Fuel Handling Machine.

710. The Fuel Handling Machine north hoist handles spent fuel in the SFP (12563), Cask Loading Pit (12463) and the Fuel Transfer Canal (12564). The south hoist handles new fuel between the Truck Bay (12371), New Fuel Storage Pit (12462) and the SFP (12563). The hoists share a common bridge beam.

711. The RP identified, based on the operating envelope, that either hoist can operate above Room 12362, 12363, 12371, 12462, 12463, 12471, 12472, 12563 and 12564. A drop from the Fuel Handling Machine has the potential to impact the floors below including the SFP.

712. The RP evaluated the impact on the floors and the SFP using a Finite Element model. Five scenarios have been considered involving impacts with the SFP liner in the tool storage area, fuel rack module and on the Rail Car Bay floor. These drops have the potential to impact the SFP and potentially penetrate the floor leading to an uncontrolled drain down and damage to the racks. The analysis showed that for the bounding drop, the reinforced concrete floors of the RCA side of the Auxiliary Building would withstand a dropped load at the capacity weight of the Fuel Handling Machine from the maximum hook height.

713. The RP also indicated that such a drop would not affect the delivery of the Category A safety function as the Class 1 SSCs associated with the RNS are not required for safe shutdown. The PXS is the primary means of providing the safety function and is unaffected by this dropped scenario. Also redundant containment isolation valve for the RNS, FHS and SFS are available inside the Containment.

714. I subjected References 181 to assessment and raised regulatory query RQ-AP1000-1608 (Ref. 50) to gain clarity on the assumptions made and on the sensitivity analysis.

715. I concluded that the assumptions used in the modelling are conservative and the results are overall reasonable. This conclusion was also supported by the civil engineering discipline (ONR Step 4 civil engineering assessment report, Ref. 14). The RP therefore provided sufficient and suitable information to substantiate claim IH-7.1.2.

716. I also sampled Chapter 16 of the PCSR (Ref. 70) in terms of identified internal hazards requirements, as given in the Topic Reports, and how the design criteria against each relevant internal hazard have been captured in the civil engineering design.

717. I concluded that Chapter 16 of the PCSR (Ref. 70) was not fully aligned with Chapter 11 of the PCSR (Ref. 69). Dropped load was not included in Chapter 16 of the PCSR. In addition, the specific loads that the civil engineering design is required to withstand were no given. Furthermore, Chapter 16 of the PCSR referenced an outdated Barrier Matrix document (Ref. 103).

718. I raised regulatory query RQ-AP10001786 (Ref. 57) to ensure alignment between Chapter 11 and Chapter 16 of the PCSR, and to obtain clarity on how the design internal hazards criteria have been captured in the civil engineering design. This is discussed in Section 6.7.1 above.

719. An early submission of the Barrier Matrix Document (Ref. 103) did not include any barriers relevant to dropped load. In response to regulatory query RQ AP1000-1786, the RP updated the Barrier Matrix (Ref. 104) document to include dropped load. The document identified a number of rooms including SFP (12563), Cask Loading Pit (12463), Cask Washdown Pit (12462), Truck Bay / Filter Storage Area (12371) and Fuel Handling Area (12562) relevant to Fuel Handling Machine dropped loads discussed above.

720. I also queried the availability of design calculation for the end-stops of the Cask Handling Crain (Regulatory query RQ-AP1000-1792, Ref. 63). The RP indicated that the current design is based on the Cask Handling Crane and Spent Fuel Cask supplied to US customers. The final design calculations for the end-stops cannot be completed until during the licensing stage as it will depend on the size and weight of the licensee's Spent Fuel Cask. However, the cask shall not extend further than 1.22m from the centreline to the furthest extremity and as such will prevent from lifting above the SFP.

721. Overall, I am satisfied with the selection of the bounding dropped load events and the detailed quantitative consequence analysis undertaken to substantiate the claims made.

9.5 ASSESSMENT FINDINGS

722. During my assessment one item was identified for a licensee to take forward in their site-specific safety submissions. Details of this are contained in Annex 2.

723. This matter does not undermine the generic safety submission and is 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.

9.6 MINOR SHORTFALLS

724. There are no minor shortfalls identified from my assessment of dropped loads.

9.7 GDA ISSUE GI-AP1000-IH-06 CONCLUSION

725. The submission provides the requisite information relating to the identification of dropped loads, consequences analysis and the identification of safety measures. Suitable and sufficient claims have been made and these were supported by the requisite arguments and evidence. I am, therefore, satisfied that during the GDA, dropped loads have been subjected to detailed review and substantiation.

726. One assessment finding identified.

727. I am therefore satisfied that GDA Issue GI-AP1000-IH-06 can be closed.

10. COMPARISONS WITH STANDARDS

10.1 STANDARDS, GUIDANCE AND RELEVANT GOOD PRACTICE

728. This assessment has been carried out in accordance with HOW2 guide NS-PER-GD-014, ‘Purpose and Scope of Permissioning’ (Ref. 1).

729. Comparison with standards is discussed in Section 2 and throughout my assessment report.

10.2 OVERSEAS REGULATORY INTERFACE

730. No overseas regulatory interactions have taken place.

11. CONCLUSIONS

731. This report presents the findings of the assessment of the internal hazards GDA Issues GI-AP1000-IH-01 to IH-06 relating to the **AP1000** reactor GDA closure phase.

732. The GDA issues actions arose in Step 4 were due to lack of sufficient information, which limited the extent of ONR's Step 4 assessment. A number of areas were identified where the safety case presented failed to adequately address the requisite claims, arguments and evidence.

733. Early in the closure phase of the six GDA issues, the RP recognised the need to review and re-issue all claims, arguments and evidence for each GDA issue.

734. The RP submitted Topic Reports for each GDA issue and a revised PCSR. The Topic Reports and PCSR were supported by a number of lower tier documentation.

735. My assessment focused on the suitability and sufficiency of the claims, arguments and evidence presented in the Topic Reports and in the revised PCSR for closing out the six GDA issues related to internal hazards. I gave particular focus on the identification of initiating events, analysis methodologies and criteria, consequences analysis, suitability of the engineering safety measures, and on the adequacy of the redundancy, segregation and separation principles adopted in the **AP1000** plant design.

736. In the area of internal fire the RP undertook significant fire analysis using different modelling techniques to justify the bounding fire case. This was subsequently used in the substantiation of concrete fire barriers and the steel-concrete-steel fire barriers.

737. The RP also undertook a gap analysis on the design of fire dampers between the UK and US codes and standards. It identified a number of shortfalls which will be implemented during the site licensing phase of the UK **AP1000** reactor project.

738. In the area of internal flooding, the RP presented a revised safety case on internal flooding. The RP has undertaken a systematic identification of flooding scenarios and consequences analysis based on gross failure, which culminated in the derivation of specific claims. The claims were supported by the requisite arguments and evidence.

739. In the area of pressure part failure, the RP responded positively to my challenge and revised its design criteria to meet ONR expectations. This now requires that they undertake gross failure analysis for a large number of high energy lines. The RP recognising that they could not complete the amount of analysis for all these lines within the timescales available for GDA closeout, agreed to provide examples of their analysis of some representative case studies to build ONR confidence in their approach and to demonstrate that the overall design will be fundamentally unaffected. Full implementation of the revised design criteria can only be completed post GDA and during detailed analysis. Based on qualitative arguments and the limited case studies available, I was able to gain sufficient confidence that the full implementation of the revised design criteria should not result in major design modifications. Overall, the submissions provided information relating to the process and methodology used in the identification of pressure part failure events, characterisations of the consequences and identification of safety measures.

740. In the area of internal explosion, the RP significantly revised its safety case to address the GDA issue. This included revised claims, arguments, and evidence for both the battery rooms as well as the routing of hydrogen pipework.

741. In the area of internal missile, the RP undertook a systematic identification of all potential internal missiles and characterisation of the consequences analysis. Revised claims arguments and evidence were presented.

742. In the area of dropped load, the RP identified all lifting devices and undertook a systematic drop load identification study. Bounding dropped load consequences analysis presented which aided the development of suitable claims, arguments and evidence.

743. I concluded that the Topic Reports and revised PCSR provide the requisite information relating to the identification of potential initiating events, consequences analysis and the identification of safety measures. Suitable and sufficient claims have been made and these were supported by the underpinning arguments and evidence.

744. My conclusion is based upon the following factors:

- During the GDA closeout phase, I held regular interventions and workshops with the RP.
- I challenged and influenced the RP to revise its design criteria on flooding, pressure part failure, internal missiles and dropped load in line with the relevant good practice established in the UK.
- The challenges ensured consistency in the assessment criteria of pressure part failure between internal hazards, structural integrity and fault studies.
- We achieved convergence on UK regulatory expectations on pressure part failure and flooding.
- I challenged the RP on the qualitative and quantitative consequences analysis undertaken for all areas including the computational modelling analysis.
- I carried out a detailed assessment of the Topic Reports and PCSR Chapter 11 to gain confidence that the claims are suitable and sufficient, and that are supported by robust arguments and evidence.
- I challenged the RP via targeted regulatory queries, which influenced and improved the safety case submissions for all areas.
- the RP adopted a positive and reactive approach to addressing the GDA issues, which led to them producing additional documentation in response to my concerns.
- I liaised with ONR specialist disciplines of civil engineering, fault studies, structural integrity and probabilistic safety assessment to maintain consistency and clarify interfaces between our assessments.

745. I raised 11 Assessment Findings and one minor shortfall for a licensee to consider and take forward in their site-specific safety submissions. These relate to further consideration of postulated events, consequence analysis and adequacy of safety measures using site specific information to ensure risks are reduced to So Far As Is Reasonably Practicable (SFAIRP). These matters do not undermine the generic safety submission, but are at the next stage of the project and may require licensee input/decision.

746. Overall, I am satisfied with the safety case submission. I reached this decision as there are no significant shortfalls against relevant good practice, established standards or significant failure in the technical quality of the final GDA submissions.

747. To conclude, I recommend that the internal hazards GDA issues GI-AP10000-IH-01 to IH-06 be closed.

12. REFERENCES

1. ONR HOW2 Guide NS-PER-GD-014 Revision 5 – Purpose and Scope of Permissioning. August 2015. www.onr.org.uk/operational/assessment/index.htm

2. Safety Assessment Principles for Nuclear Facilities. 2014 Edition Revision 0. November 2014. www.onr.org.uk/saps/saps2014.pdf.

3. ONR Nuclear Safety Technical Assessment Guide. Internal Hazards. NS-TAST-GD-014, Revision 4. September 2016. www.onr.org.uk/operational/tech_asst_guides/index.htm

4. ONR Nuclear Safety Technical Assessment Guide. Guidance on the Demonstration of ALARP ((As Low As Reasonably Practicable), NS-TAST-GD-005, Revision 7. December 2015. www.onr.org.uk/operational/tech_asst_guides/index.htm

5. Western European Nuclear Regulators' Association (WENRA). Safety Reference Levels for Existing Reactors. September 2014 – Issue S: Protection against Internal Fires. www.wenra.org.

6. International Atomic Energy Authority (IAEA). Safety Guide No. NS-G-1.7. Protection against Internal Fires and Explosions in the Design of Nuclear Power Plants. 2004.

7. International Atomic Energy Authority (IAEA). Safety Guide No. NS-G-1.11. Protection against Internal Hazards other than Fire and Explosions in the Design of Nuclear Power Plants. 2004.

8. International Atomic Energy Authority (IAEA). Safety Guide No. NS-G-2.1. Fire Safety in the Operation of Nuclear Power Plants. 2000.

9. ASME Section III (2015) – Boiler & Pressure Vessel Code for Construction of Nuclear Power Plant Components.

10. ASME B31.1 (2012) – Standards of Pressure Piping – Power Piping.

11. BS6133 (1995) – Code of Practice for Safe Operation of Lead-Acid Stationary Batteries.

12. BS EN 50272-1 (2001) – Safety Requirements for Secondary Batteries and Battery Installations.

13. ONR-GDA-AR-11-001 – Step 4 Internal Hazards Assessment of the Westinghouse AP1000® Reactor (2011). ONR TRIM Ref. 2010/579786.

14. ONR-GDA-AR-11-002 – Step 4 Civil Engineering and External Hazards Assessment of the Westinghouse AP1000® Reactor (2011). ONR TRIM Ref.: 2010/581528.

15. ONR-GDA-AP-14-011 – AP1000 GDA Internal Hazards Assessment Plan, Revision 0. ONR TRIM Ref. 2015/149251.

16. UKP-GW-GL-144 – AP1000 UK Safety Categorisation and Classification of Systems, Structures and Components, Revision 2. ONR TRIM Ref. 2016/417844.

17. GRS – Provision of Support on Internal Hazards Safety Assessment of the AP1000 GDA Issues, Draft ONR TRIM Ref. 2017/59044.

18. Westinghouse UK AP1000 Generic Design Assessment Resolution Plan for GI-AP1000-IH-01 – Internal Fire Safety Case Substantiation, Revision 8. ONR TRIM Ref. 2016/154409.

19. Westinghouse UK AP1000 Generic Design Assessment Resolution Plan for GI-AP1000-IH-02 – Internal Flooding Safety Case, Revision 9. ONR TRIM Ref. 2016/92225.

20. Westinghouse UK AP1000 Generic Design Assessment Resolution Plan for GI-AP1000-IH-03 – Pressure Part Failure, Revision 10, ONR TRIM Ref. 2016/154421.

21. Westinghouse UK AP1000 Generic Design Assessment Resolution Plan for GI-AP1000-IH-04 – Internal Explosion Safety Case Substantiation, Revision 9. ONR TRIM Ref. 2016/154435.

22. Westinghouse UK AP1000 Generic Design Assessment Resolution Plan for GI-AP1000-IH-05 – Internal Missile Safety Case, Revision 9. ONR TRIM Ref. 2016/92242.

23. Westinghouse UK AP1000 Generic Design Assessment Resolution Plan for GI-AP1000-IH-06 – Substantiation and Analysis of the Consequences of Dropped Loads and Impact from Lifting Equipment Included Within the AP1000 Design, Revision 6 ONR TRIM Ref. 2016/92245.

24. RQ-AP1000-1301 – Request to clarify the submissions relevant to GI-AP1000-IH-01. ONR TRIM Ref. 2015/96917.

25. RQ-AP1000-1302 – Request to clarify the submissions relevant to GI-AP1000-IH-02. ONR TRIM Ref.2015/96921.

26. RQ-AP1000-1303 – Request to clarify the submissions relevant to GI-AP1000-IH-03. ONR TRIM Ref. 2015/96925.

27. RQ-AP1000-1304 – Request to clarify the submissions relevant to GI-AP1000-IH-04. ONR TRIM Ref. 2015/96932.

28. RQ-AP1000-1305 – Request to clarify the submissions relevant to GI-AP1000-IH-05. ONR TRIM Ref. 2015/96936.

29. RQ-AP1000-1306 – Request to clarify the submissions relevant to GI-AP1000-IH-06. ONR TRIM Ref. 2015/96940.

30. RQ-AP1000-1336 – Internal Flooding Roadmap. ONR TRIM Ref. 2015/226020.

31. RQ-AP1000-1340 – Internal Fire Roadmap. ONR TRIM Ref. 2015/224193.

32. RQ-AP1000-1378 – Pressure Part Failure – Design Criteria Clarification. ONR TRIM Ref. 2015/349436.

33. RQ-AP1000-1429 – Plant Fire Resistance. ONR TRIM Ref. 2016/123363.

34. RQ-AP1000-1430 – Fire / Smoke Dampers. ONR TRIM Ref. 2016/104392.

35. RQ-AP1000-1431 – Internal hazards impacts on the Diverse Actuation System. ONR TRIM Ref. 2016/91069.

36. RQ-AP1000-1439 – Internal explosion – Hydrogen Assessment. ONR TRIM Ref. 2016/261899.

37. RQ-AP1000-1440 – Internal missiles. ONR TRIM Ref. 2016/195147.

38. RQ-AP1000-1443 – Documentation on Internal Hazard Impacts on Wall 11. ONR TRIM Ref. 2016/466853.

39. RQ-AP1000-1444 – Additional Internal Missiles Reference. ONR TRIM Ref. 2015/448923.

40. RQ-AP1000-1506 – Internal explosion – Hydrogen Assessment. ONR TRIM Ref. 2016/121815.

41. RQ-AP1000-1507 – Internal missiles. ONR TRIM Ref.2016/116129.

42. RQ-AP1000-1515 – Internal fire – Fire Protection Analysis. ONR TRIM Ref.2016/181092.

43. RQ-AP1000-1516 – Internal flooding – Water spray, steam generation and wave formation. ONR TRIM Ref. 2016/130169.

44. RQ-AP1000-1529 – ONR Comments on Resolution Plans GI-AP1000-IH-01 to IH-06. ONR TRIM Ref. 2016/154384.

45. RQ-AP1000-1549 – Pressure Part Failure Methodology, Criteria, Examples. ONR TRIM Ref. 2016/209852.

46. RQ-AP1000-1585 – Pressure Part Failure Proposed Bounding Examples for GDA, and Proposal to Address Remaining Lines Post GDA. ONR TRIM Ref. 2016/341228.

47. RQ-AP1000-1590 – Comments on Fire Protection Dampers – UK Compliance Report. ONR TRIM Ref. 2016/323040.

48. RQ-AP1000-1591 – Internal Fire. ONR TRIM Ref. 2016/276554.

49. RQ-AP1000-1607 – Internal Missiles. ONR TRIM Ref. 2016/294798.

50. RQ-AP1000-1608 – Dropped Loads Modelling. ONR TRIM Ref. 2016/300110.

51. RQ-AP1000-1641 – Internal Fire – Comments from the Internal Hazards July 2016 Workshop. ONR TRIM Ref. 2016/347047.

52. RQ-AP1000-1642 – Internal Explosions – Comments from the Internal Hazards July 2016 Workshop. ONR TRIM Ref. 2016/401416.

53. RQ-AP1000-1675 – Comments on the UK AP1000 Internal Hazards Topic Report - Dropped Loads – Revision 0. ONR TRIM Ref.2016/378676.

54. RQ-AP1000-1676 – Comments on the UK AP1000 Internal Hazards – Flooding Topic Report – Revision 0. ONR TRIM Ref. 2016/378550.

55. RQ-AP1000-1677 – Comments on Pressure Part Failure Topic Report UKP-GW-GLR-114 Rev A. ONR TRIM Ref.2016/378647.

56. RQ-AP1000-1702 – Comments on Pressure Part Failure Topic Report UKP-GW-GLR-114 Rev 0; on Supplemental Gross Failure Analysis Cases for Generic Design Assessment (Rev. 1) WEC_REG_001218; and on Response to ONR's letter NPP_JNE_001218. ONR TRIM Ref. 2016/411211.

57. RQ-AP1000-1786 – PCSR Chapter 16 – Internal Hazards Input Requirements. ONR TRIM Ref. 2017/27352.

58. RQ-AP1000-1797 – Internal Fire Assessment Queries. ONR TRIM Ref.2017/42648.

59. RQ-AP1000-1796 – Internal Flooding Assessment Queries. ONR TRIM Ref.2017/42649.

60. RQ-AP1000-1795 – Pressure Part Failure Assessment Queries. ONR TRIM Ref.2017/42650.

61. RQ-AP1000-1794 – Internal Explosions Assessment Queries. ONR TRIM Ref. 2017/42653.

62. RQ-AP1000-1793 – Internal Missiles Assessment Queries. ONR TRIM Ref. 2017/42657.

63. RQ-AP1000-1792 – Dropped Load Assessment Queries. ONR TRIM Ref. 2017/42658.

64. UKP-GW-GL-060 – AP1000 Design Reference Point for UK GDA, Revision 10, ONR TRIM Ref. 2017/18158.

65. UKP-GW-GL-793 – Pre Construction Safety Report, Chapter 5 – Engineering Principles, Revision 0B. ONR TRIM Ref. 2016/382157.

66. UKP-GW-GL-793 – Pre-construction Safety Report, Chapter 6 – Plant Description and Operation, Revision 0C. ONR TRIM Ref. 2017/18995.

67. UKP-GW-GL-793 – Pre-construction Safety Report, Chapter 8 – Fault and Accident Analysis, Revision 0E. ONR TRIM Ref. 2017/33053.

68. UKP-GW-GL-793 – Pre-construction Safety Report, Chapter 11 – Internal Hazards Revision 0C. ONR TRIM Ref. 2016/421144.

69. UKP-GW-GL-793 – Pre-construction Safety Report, Chapter 11 – Internal Hazards Revision 0D. ONR TRIM Ref. 2017/29228.

70. UKP-GW-GL-793 – Pre-construction Safety Report, Chapter 16 – Civil Engineering, Revision 0B. ONR TRIM Ref. 2016/421181.

71. UKP-GW-GLR-036 – UK AP1000 Internal Hazards Topic Report – Combined Hazards, Revision 0. ONR TRIM Ref. 2016/345456.

72. UKP-GW-GLR-107 – UK AP1000 Internal Hazards – Flooding Topic Report, Revision 0. ONR TRIM Ref. 2016/211042.

73. UKP-GW-GLR-107 – UK AP1000 Internal Hazards – Flooding Topic Report, Revision 1. ONR TRIM Ref. 2017/24707.

74. UKP-GW-GLR-108 – UK AP1000 Internal Hazards Topic Report - Internal Missiles, Revision 0. ONR TRIM Ref. 2016/196089.

75. UKP-GW-GLR-108 – UK AP1000 Internal Hazards Topic Report - Internal Missiles, Revision 1. ONR TRIM Ref. 2017/29147.

76. UKP-GW-GLR-109 – UK AP1000 Internal Hazards Topic Report – Explosion, Revision 0. ONR TRIM Ref. 2016/261980.

77. UKP-GW-GLR-109 – UK AP1000 Internal Hazards Topic Report – Explosion, Revision 1. ONR TRIM Ref. 2017/29162.

78. UKP-GW-GLR-110 – UK AP1000 Internal Hazards Topic Report – Dropped Loads, Revision A, ONR TRIM Ref. 2016/96210.

79. UKP-GW-GLR-110 – UK AP1000 Internal Hazards Topic Report – Dropped Loads, Revision 0, ONR TRIM Ref.: 2016/280536.

80. UKP-GW-GLR-110 – UK AP1000 Internal Hazards Topic Report – Dropped Loads, Revision 1, ONR TRIM Ref. 2017/29171.

81. UKP-GW-GLR-111 – UK AP1000 Internal Hazards Topic Report - Fire Protection, Revision 0, ONR TRIM Ref. 2016/202909.

82. UKP-GW-GLR-111 – UK AP1000 Internal Hazards Topic Report - Fire Protection, Revision 1, ONR TRIM Ref. 2017/24690.

83. UKP-GW-GLR-114 – UK AP1000 Plant Internal Hazards Topic Report – Pressure Part Failure, Revision A. ONR TRIM Ref. 2016/170306.

84. UKP-GW-GLR-114 – UK AP1000 Plant Internal Hazards Topic Report – Pressure Part Failure, Revision 0. ONR TRIM Ref. 2016/332094.

85. UKP-GW-GLR-114 – UK AP1000 Plant Internal Hazards Topic Report – Pressure Part Failure, Revision 0A. ONR TRIM Ref. 2016/431926.

86. UKP-GW-GLR-114 – UK AP1000 Plant Internal Hazards Topic Report – Pressure Part Failure, Revision 1. ONR TRIM Ref. 2017/24720.

87. APP-AB01-Z0-001 – Blockouts and Barriers (Penetrations, Seals and Fire Stops) Westinghouse Safety Case C, Seismic Category 1, Revision 4. ONR TRIM Ref. 2015/224211.

88. APP-AY20-Z0-003 – Design Specification for MSIV Compartment Lower Relief Panels, Revision 1. ONR TRIM Ref. 2016/479110

89. APP-GW-G1-002 – AP1000 Equipment Qualification Methodology, Revision 4. ONR TRIM Ref. 2015/350769.

90. APP-GW-N1-001 – Pipe Rupture Protection Design Criteria for the AP1000 Plant, Revision 5. ONR TRIM Ref. 2015/226079.

91. APP-GW-N1-007 – AP1000 Design Criteria for Protection from Flooding, Revision 4. ONR TRIM Ref. 2015/226049.

92. APP-GW-VP-030 – AP1000 Environment Conditions (For Equipment Qualification), Revision 5. ONR TRIM Ref. 2016/354096.

93. APP-PXS-M3C-034 – Containment Flood-Up Level, Revision 3. ONR TRIM Ref. 2015/316661.

94. APP-SSAR-GSC-743 – AP1000 PRHA Evaluation of Internal Flooding in Containment, Revision 3. ONR TRIM Ref. 2014/467147.

95. APP-SSAR-GSC-761 – AP1000 PRHA Safety-Related Equipment Flooding Target Identification, Revision 4. ONR TRIM Ref. 2015/299910

96. APP-SSAR-GSC-780 – AP1000 PRHA Auxiliary Building Analysis of Internal Flooding as a Result of Postulated Moderate-Energy Line Breaks Affecting Level 2 Areas and Higher, Revision 2. ONR TRIM Ref. 2014/467138.

97. APP-1000-CCC-017 – Structural Acceptance of PRHA Internal Flood in Auxiliary and Shield Buildings, Revision 0. ONR TRIM Ref. 2016/459962.

98. APP-1000-E6C-002 – AP1000 Nuclear Island Hydrogen Piping Equipment and System Assessment, Revision 0. ONR TRIM Ref. 2016/417861.

99. NPP_JNE_000150 – AP1000 Internal Flooding Safety Case Roadmap, Revision 1. ONR TRIM Ref. 2015/226033.

100. UKP-GW-GL-114 – UKP AP1000 Auxiliary Building Battery Rooms – Hydrogen Assessment, Revision 0. ONR TRIM Ref.2016/262012.

101. UKP-GW-GLC-003 – UK AP1000 Flooding Variations, Revision 0. ONR TRIM Ref. 2016/354093.

102. UKP-GW-M3C-002 – UKP-GW-M3C-002 - Revision 0 - UKP AP1000 Chemical Explosion Evaluation, Revision 0. ONR TRIM Ref. 2016/328356.

103. UKP-1000-GEC-004 – UK AP1000 Barrier Matrix, Revision 0. ONR TRIM Ref. 2016/108543.

104. UKP-1000-GEC-004 – UK AP1000 Barrier Matrix, Revision 1. ONR TRIM Ref. 2017/29176.

105. UKP-1000-N4C-002 – UKP AP1000 Assessment of the Potential for Hydrogen Combustion due to Leakage from the Hydrogen Injection Line in the Auxiliary Building, Revision 0. ONR TRIM Ref. 2016/261988.

106. UKP-1000-N4C-003 – UKP AP1000 Assessment of the Potential for Hydrogen Combustion due to Leakage from the Hydrogen Injection Line in Containment, Revision 0. ONR TRIM Ref. 2016/262054.

107. UKP-1000-N4C-004 – UK AP1000 WLS and WGS Hydrogen Assessment, Revision 0. ONR TRIM Ref. 2016/401416.

108. UKP-1000-N4C-005 – AP1000 Hydrogen Gas Explosion Evaluation using TNO MEM, Revision 0. ONR TRIM Ref. 2016/262005.

109. UKP-1000-N4C-007 – FAI/16-1170, AP1000 Hydrogen Migration Analysis for CVS Hydrogen Line Break in Selected Rooms of the Auxiliary Building and Containment Building, Revision 1. ONR TRIM Ref. 2017/29196.

110. UKP-1000-N4C-008 – FAI/16-1481, Unmitigated Explosion Hazard Analysis for AP1000 Division B Battery Room 1 (Room 12104), Revision 0. ONR TRIM Ref. 2017/29203.

111. UKP-GW-GLR-118 – UK AP1000: Review of Claims for Internal Fires within Containment, Revision 0, ONR TRIM Ref. 2016/431089.

112. APP-1200-N4C-001 – Heat transfer Analysis of postulated Fires on Exposed Auxiliary Structural Steel Floors, Revision 0. ONR TRIM Ref. 2016/314813.

113. ONR-NR-AR-16-010, Rev 0, GDA close out for the AP1000 reactor GDA Issue GI-AP1000-SI-01: Avoidance of fracture. ONR TRIM Ref. 2017/74011.

114. Organisation for Economic Co-Operation and Development (OECD) Nuclear Energy Agency (NEA), Committee on the Safety of Nuclear Installations (CSNI) OECD Fire Project –Topical Report No.1, Analysis of High Energy Arching Fault (HEAF) Fire Events, NE/CSNI/R(1013)6, France June 2013.

115. Organisation for Economic Co-Operation and Development (OECD) Nuclear Energy Agency (NEA), Committee on the Safety of Nuclear Installations (CSNI), Working Group IAGE (WGIAGE): OECD NEA CSNI WGIAGE Task on high Energy Arching Fault Events (HEAF), NE/CSNI/R(1015)10, France June 2015.

116. Organisation for Economic Co-Operation and Development (OECD) Nuclear Energy Agency (NEA), Committee on the Safety of Nuclear Installations (CSNI) OECD Fire Project –Topical Report No.1, Analysis of High Energy Arching Fault (HEAF) Fire Events, NE/CSNI/R(1013)6, France 2013.

117. Organisation for Economic Co-Operation and Development (OECD) Nuclear Energy Agency (NEA), Committee on the Safety of Nuclear Installations (CSNI), Combinations of Fires and Other Events – The Fire incidents Records Exchange Project Topical Report No. 3, NE/CSNI/R(1016)7, France July 2016.

118. APP-GW-N4R-003 – Fire protection Analysis Report, Revision H. ONR TRIM Ref. 2015/224349.

119. EN 1991-1-2 – Actions on Structures Exposed to Fires, November 2002.

120. APP-PRA-GSC-379 – NPP JNE 001127 Transmittal of Supporting Calculations EN1991-1-2 to APP-PRA-GSC-379 AP1000 Fire Modelling Assessment of Wall exposure Temperature profiles for Selected Rooms ONR TRIM Ref. 2016/290893.

121. APP-PRA-GSC-379 – AP1000 Fire modelling Assessment of Wall Exposure Temperature Profiles for Selected Rooms, Revision C. ONR TRIM Ref. 2016/276605.

122. APP-PRA-GSC-379 – AP1000 Fire Modelling Assessment of Wall Exposure Temperature Profile for Selected Room, Revision D. ONR TRIM Ref. 2016/487351.

123. UKP-GW-AF-101 – AP1000 Plant Fire Resistance Rating for Nuclear Island Reinforced Concrete Structures, Revision 0. ONR TRIM Ref.: 2016/132504.

124. Civil Engineering Resolution Plan GI-AP1000-CE-01.ONR TRIM Ref. 2014/368133.

125. ONR-NR-AR-16-018 – Probabilistic Safety Analysis for the Westinghouse AP1000 Reactor, GDA Issue 02.

126. ONR-NR-AR-16-040 – Revision 0, GDA Close out for the AP1000 Reactor GDA Issues GI-AP1000-CE-01 and GI-AP1000-CE-02 Revision 1, Revision 0. ONR TRIM Ref. 2016/274970.

127. UKP-GW-GL-123 – Heat Transfer Analysis of SC Walls and Floors in a Three-Hour Standard Fire, Revision 0. ONR TRIM Ref. 2016/141925.
128. APP-1200-N4C-001 – Heat Transfer Analysis of Postulated Fires on Exposed Auxiliary Building Structural Steel Floor, Revision 0. ONR TRIM Ref. 2016/314813.
129. APP-1100-S3C-018, Structural Stability Analysis of the Auxiliary Building under Postulated Fire Load, Revision 0. ONR Ref. 2016/314824.
130. UKP-GW-GL-131 – Global Analysis for postulated Fire Loadings, Revision 0. ONR Ref. 2016/495703.
131. UKP-GW-AF-001 – AP1000 Fire Protection Dampers – UK Compliance Report, Revision A. ONR TRIM Ref. 2015/278169.
132. UKP-GW-AF-001 – AP1000 Fire Protection Dampers – UK Compliance Report, Revision 0. ONR TRIM Ref. 2016/124266.
133. UKP-GW-AF-001 – AP1000 Fire Protection Dampers – UK Compliance Report, Revision 1. ONR TRIM Ref. 2016/323040.
134. APP-GW-G1-002 – AP1000 Equipment Qualification Methodology, Revision 4, ONR TRIM Ref. 2015/350769.
135. APP-GW-N1-001 – Pipe Rupture Criteria for AP1000 Plant, Revision 5. ONR Ref. 2015/226079.
136. ONR-WEC-0023R – Letter Pressure Part Failure. ONR TRIM Ref. 2015/452454.
137. NPP_JNE_000776 – Summary of AP1000 Pressure Part Failure Methodology for Mechanistically and Deterministically Designed Piping. ONR TRIM Ref. 2016/116526.
138. NPP_JNE_000723 – AP1000 Pressure Part Failure Assumptions. ONR TRIM Ref. 2016/90769.
139. NPP_JNE_000719 – Identification of Assessment Examples – Pressure Part Failure. ONR TRIM Ref. 2016/90880.
140. NPP_JNE_000829 – AP1000 Internal Hazards Pressure Part Failure Supplemental Gross Failure Analysis Cases for Generic Design Assessment. ONR TRIM Ref. 2016/133278.
141. NPP_JNE_001026 – Supplementary Gross Failure Analysis Cases for Generic Design Assessment. ONR TRIM Ref. 2016/239023.
142. WEC-REG-1026R – Letter Pressure Part Failure. ONR TRIM Ref. 2016/303589.
143. APP-GW-GLR-075 – AP1000 Pipe Rupture Hazard Analysis (PRHA) Summary Report for the Auxiliary Building – All Levels, Revision 2. ONR TRIM Ref. 2016/378433.
144. APP-GW-GLR-076 – AP1000 Pipe Rupture Hazard Analysis (PRHA) Summary Report for the Containment Building – All Levels, Revision 2. ONR TRIM Ref. 2016/378445.
145. WEC_REG_001218 – AP1000 Internal Hazards Pressure Part Failure Supplemental Gross Failure Analysis Cases for Generic Design Assessment, Revision 1. ONR TRIM Ref. 2016/341228
146. ONR Internal Communication. ONR TRIM Ref. 2017/44263.
147. ONR Internal Communication. ONR TRIM ref. 2017/44298.
148. NPP-JNE-001515 – Nuclear Site Licensing Task Summary for AP1000 Plant Pressure Part Failure Assessment Process. ONR TRIM Ref. 2017/8305.
149. NPP_JNE_001515 – Examples of AP1000 Plant Pressure Part Failure Load Combination for DB1/DB2 Events. ONR TRIM Ref. 2017/8302.
150. APP-GW-SH-001 – AP1000 Pipe Whip Restraints Design Criteria, Revision 1. ONR TRIM Ref. 2016/467253.
151. UKP-GW-GL-045 – AP1000 Equivalence/ Maturity Study of US codes and Standards, Revision 2. ONR TRIM Ref. 2015/473946.
152. APP-SH50-SHC-001 – Break SGS-022 Design for jet Shield SGS-SH-P022, Revision 0. ONR TRIM Ref.: 2016/484379.

153. APP-AY20-20-001 – MSIV Compartment Roof Steam Relief Panels, Revision 0. ONR TRIM Ref 2016/378405.

154. APP-GW-N1-004 – Design Criteria for the Protection from Internally Generated Missiles, Revision 1, ONR TRIM Ref.: 2015/155148.

155. APP-GW-M3C-013 – AP1000 Valve Missile Protection, Revision 0. ONR TRIM Ref. 2014/467298.

156. APP-MV11-Z0-001 – Design Specification for AP1000 Control Rod Drive Mechanism (CRDM) Design Specification (CDRM) Design Specification for System Reactor System (RXS), Rev 6.

157. NPP_JNE_001605 – Draft Internal Missile Methodology. ONR TRIM Ref. 2015/417969.

158. APP-GW-GEE-5405 – UK AP1000 Internal Hazards Design Change Proposal, Revision 0. ONR TRIM Ref. 2016/483812.

159. UKP-1000-N4C-001 – UK AP1000 Nuclear Island Missile Penetration Calculation, Revision 0. ONR TRIM Ref. 2016/294874.

160. UKP-GW-N4C-001 – UK AP1000 Turbine Missile Assessment, Revision 0. ONR TRIM Ref. 2016/294840.

161. ONR-NR-AR-15-001 – Assessment of Missile Analysis from Postulated Failure of AP1000 Reactor Coolant Pump Flywheel, Revision 0. ONR TRIM Ref.2017/24753.

162. ONR-NR-AR-16-005 – AP1000 – Step 4 GDA Assessment Issues – GI-AP1000-SI-03 - Reactor Coolant Pump - Pump Bowl Integrity and Flywheel Disintegration Case, Revision 0. ONR TRIM Ref. 2017/31112.

163. UKP-GW-GL-107 – AP1000 Plant Assessment of Impact from Reactor Coolant Pump Failure on Steam Generator Column Report, Revision 1. ONR TRIM Ref. 2016/498113.

164. ONR-NR-AR-16-013 – GDA issue GI-AP1000-SI-06 – Structural Integrity Categorisation and Classification, Revision 0. ONR TRIM Ref. 2017/67167.

165. APP-GW-GEE-5413 – Steam Generator Vertical Column Support Modification, Revision 0. ONR TRIM Ref.: 2016/415382.

166. ONR Internal Communication. ONR TRIM Ref. 2017/24792.

167. ONR internal Communication. ONR TRIM Ref. 2017/24791.

168. NPP_JNE_000093 – Summary of AP1000 Rotating Equipment Internal Missile Source Identification. ONR TRIM Ref. 2015/155141.

169. APP-SSAR-GSC-802 – Safety-Related Equipment Flooding Target Identification in Containment. ONR TRIM Ref. 2015/299912.

170. GRS Response to Westinghouse Comments in ONR's AR on HEAF Consideration in AP1000 design. TRIM Ref. 2017/113460.

171. NPP_JNE_000263 – AP1000 Dropped Loads Safety Case Roadmap, Revision 0. ONR TRIM Ref. 2015/314010.

172. NPP_JNE_000269 – AP1000 Internal Explosions Safety Case Roadmap, Revision 0. ONR TRIM Ref. 2015/327114.

173. NPP_JNE_000087 – AP1000 Internal Fire Safety Case Roadmap, Revision 1. ONR TRIM Ref. 2015/195364.

174. NPP_JNE_000117 – AP1000 Internal Missiles Safety Case Roadmap, Revision 0. ONR TRIM Ref. 2015/193403.

175. NPP_JNE_000245 – AP1000 Pressure Part Failure Safety Case Roadmap, Revision 0. ONR TRIM Ref. 2015/299825.

176. GDA Close-out for the AP1000 Reactor – GDA issue HF-01 – Completeness of the Human Factors Safety Case, Specifically in the Areas of Human Error Mechanisms, Operator Misdiagnosis Potential and Violation Potential. ONR TRIM Ref. 2016/274996.

177. ONR-GDA-AR-11-010 – Step 4 Mechanical Engineering of the Westinghouse AP1000 Reactor. ONR TRIM Ref. 2010/581521.

178. UKP-1000-VMR-100 – AP1000 Load Path Assessment, Revision A. ONR TRIM Ref. 2015/308544.

179. UKP-1000-VMR-100 – AP1000 Load Path Assessment, Revision 0. ONR TRIM Ref. 2015/472445.

180. UKP-MV01-Z0C-081 – AP1000 Reactor Vessel Head Drop Evaluation, Revision 0. ONR TRIM Ref. 2016/365173.

181. UKP-FHS-S0C-001 – UK AP1000 Fuel Assembly Drop Accident Report, Revision 0. ONR TRIM Ref. 2016/3000144.

182. UKP-1000-VMR-101 – AP1000 Handling Equipment Dropped Load Exclusions, Revision 0. ONR TRIM Ref. 2016/471661.

183. APP-GW-GEE-4568 – Changing Design Basis Due to Moderate-Energy Line Break Induced Flooding for the Auxiliary Building RCA and non-RCA Elevations 100'-0" (Level 3) and Higher. ONR TRIM Ref. 2015/346229.

184. APP-PY30-Z0R-001 – Single Stage Orifices, ANSI/ASME B31.1, Data Sheet Report, Revision 7. ONR TRIM Ref. 2017/29228.

185. CPP-PY30-Z0D -001 – Single Stage Orifices. 2”, CL300, Stainless Steel, RF/WN, Sch S-80S, Revision 1. ONR TRIM Ref. 2017/29222.

186. UKP-GW-GL-044 – AP1000 UK Safety Categorization and Classification Methodology, Revision 1. ONR TRIM Ref. 2015/314260.

187. APP-VBS-M3C-101 – VBS Instrument Requirements, Revision 7. ONR TRIM Ref. 2016/475206.

188. UKP-GW-GL-793, – AP1000 Pre Construction Safety Report, Revision 1, ONR TRIM Ref. 2017/43700.

189. UKP-GW-GLR-004 – AP1000 UK Structural Integrity Classification, Revision 3. ONR TRIM Ref. 2017/27086.

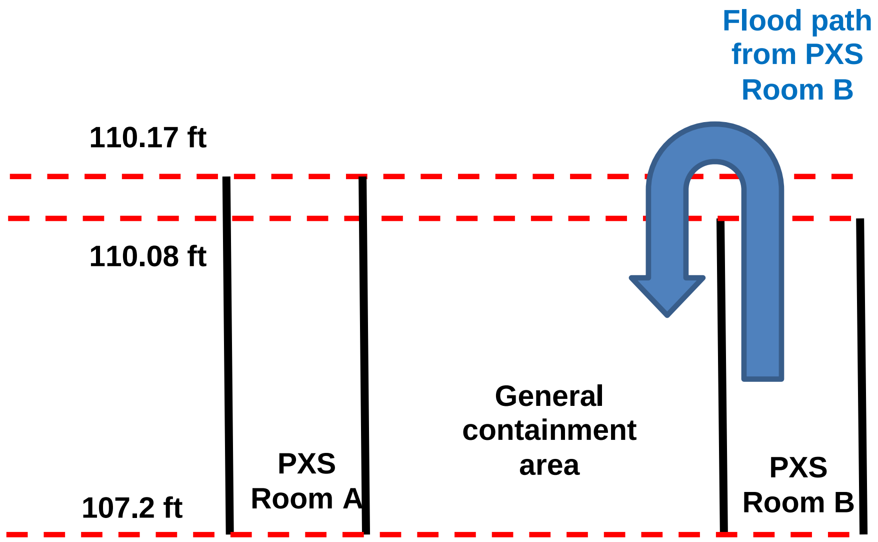
190. ONR-GDA-AR-11-011 – ONR GDA – New Civil Reactor Build, Step 4 Structural Integrity Assessment of the Westinghouse AP1000® Reactor. Revision. 0, ONR TRIM Ref. 2010/581520.

191. UKP-1000-N4C-006 – UKP AP1000 Hydrogen Explosion Evaluation of Battery Rooms within the Annex Building, Revision 0. ONR TRIM Ref. 2016/328349.

192. APP-VLS-GJP-401 – Alarm Response – Containment Hydrogen Control System, Revision 1. ONR TRIM Ref. 2016/470510.

Annex 1 - Figures

Figure 1 - Simplified elevation schematic of the PXS Room A and Room B



Annex 2

Assessment Findings to be addressed during the Forward Programme – Internal Hazards

Assessment Finding Number	Assessment Finding	Report Section Reference
CP-AF-AP1000-IH-01	The licensee shall update the Barriers Matrix document to clearly identify all claimed Class 1 and 2 Barriers (walls, floors and ceilings) for each room, and align it with all internal hazards Topic Reports. The document shall clearly state the internal hazards imposed loads as well as the design loads that the barriers are designed to withstand.	4.3.2
CP-AF-AP1000-IH-02	The licensee shall use site-specific information to: • Complete the screening assessment of hazard combinations and provide justification for those screened out. • Fully analyse all credible external and internal hazards combinations. • Justify the adequacy of the barriers.	4.3.3, 5.6 6.2.5 7.5 8.3.3 9.3.3
CP-AF-AP1000-IH-03	The licensee shall: • Develop controls and procedures to minimise the combustible inventory held in Turbine Building, the Annex Building and in the Diesel Generator Room and provide adequate fire resistance barriers where required. • Consider the impact of turbine selected on the design of the Turbine Building including fire barriers and penetrations	4.3.5
CP-AF-AP1000-IH-04	The licensee shall use site-specific information to undertake a quantitative analysis of localised fire effects on fire barriers.	4.3.5
CP-AF-AP1000-IH-05	The licensee shall carry out flooding sensitivity analysis on the reduced door gap heights, and on the assumed blockage and redundancy of the drains, and assess any impacts on Class 1 SSCs delivering or contributing to Category A functions.	5.4.2
CP-AF-AP1000-IH-06	The licensee shall complete the pressure part failure assessment based on gross failure to quantitatively characterise the total population of Medium Energy and High Energy systems and for all Design Basis Events. This shall include: • Identification and assessment of additional intermediate break locations due to stress and fatigue, erosion and corrosion, and where SSCs are in close vicinity to a High Energy system, assess the potential consequences. • The prediction of pipe behaviour and the consequential dynamic effects and impact	6.4

	on structures and SSCs shall be supported by appropriate modelling. • Evaluation of the consequences of analysing the Normal Residual Heat Removal System (RNS) system as a High Energy system, and therefore evaluating the consequence of gross failure.	
CP-AF-AP1000-IH-07	The licensee shall justify the detailed design of all penetrations on Class 1 barriers against the potential consequences of pressure part failure.	6.6.1
CP-AF-AP1000-IH-08	The licensee shall substantiate the adequacy of restraints, jet shields and relief panels against modern standards.	6.6.3
CP-AF-AP1000-IH-09	The licensee shall justify the mathematical models which: • predict the hydrogen concentration; • predict explosion pressures; to determine that barriers provide adequate protection to SSCs against the potential explosions.	7.4.2
CP-AF-AP1000-IH-10	The licensee shall carry out unmitigated consequence sensitivity analysis in the Battery Rooms for various hydrogen concentrations without crediting venting.	7.4.2
CP-AF-AP1000-IH-11	The licensee shall carry out a quantitative dropped load assessment of the selected Spent Fuel Cask.	9.4.1

Annex 3

Minor Shortfalls – Internal Hazards

Minor Shortfall Number	Minor Shortfall	Report Section Reference
CP-MS-AP1000-IH-01	The licensee shall update the hazard schedule with site-specific information to ensure the internal hazards safety case is fully captured. The hazard schedule shall summarise the Class 1 SSCs, as well the Essential Safety Shutdown SSCs, and capture all claimed safety features and defence-in-depth measures.	7.4.2