New Reactors Division

Step 4 Assessment of Mechanical Engineering for the UK Advanced Boiling Water Reactor

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EXECUTIVE SUMMARY

Office for Nuclear Regulation (ONR) and Environment Agency (EA) developed Generic Design Assessment (GDA) as a four-step process applied where a requesting party asks for assessment of a reactor design in advance of an application for a nuclear site licence. Each GDA step increases in detail and ONR publishes reports at the end of each step, which provide an update on the assessment and highlight any concerns or technical issues. GDA separates generic design issues from site-specific issues where the intention is to construct a generic design on a number of different sites. ONR/EA will not issue permits for new nuclear power stations unless the design and its potential operators meet the high safety, security, environmental and waste management standards that we require.

Hitachi-GE Nuclear Energy Ltd (known as Hitachi-GE) is the designer and GDA Requesting Party for the United Kingdom Advanced Boiling Water Reactor (UK ABWR). Hitachi-GE commenced GDA step 1 in 2013 and completed the final step (step 4) in 2017. The purpose of GDA step 4 is for ONR, EA and Natural Resources Wales (NRW) to undertake a detailed assessment of the safety, security and environmental aspects of the UK ABWR generic design.

This report presents ONR's GDA step 4 mechanical engineering safety assessment of Hitachi-GE's UK ABWR generic design. The assessment examines Hitachi's claims, arguments and supporting evidence building on ONR's assessment for GDA Step 3. The aim is to make a judgement on the adequacy of the generic design from a mechanical engineering perspective based on information contained within the Pre-construction Safety Report and supporting documentation. ONR will use this judgment to inform its decision whether the UK ABWR reactor is suitable for construction in the UK.

The report covers the mechanical engineering assessment of safety functions such as reactivity control, heat transfer and removal, lifting and handling operations, and containment of radioactive substances. This assessment considers mechanical structures systems and components that deliver these safety functions to identify whether there are any weaknesses in the proposed UK ABWR generic design.

The mechanical structures systems and components selected for assessment provide a broad sample of equipment types including lifting and handling systems, ventilation systems, pumps, valves, heat exchangers, heat transport systems and reactivity control mechanisms. The assessment seeks evidence that the equipment has an appropriate level of engineering qualification, supported by suitable operational experience. For example, the assessment considers how the UK ABWR generic design has evolved from the Japanese reference design, adopting well-qualified mechanical engineering systems supported by operational experience.

Assessors vary the depth of their assessment to suit the evidence presented. Hence, a more detailed assessment has been undertaken where Hitachi-GE has identified equipment or processes less well supported by operational experience, employ new or novel techniques or appear not to satisfy relevant good practice in the UK. An example where detailed assessment took place is the spent fuel export system that transfers and lowers packaged fuel down a hoist well. Hitachi-GE's proposals did not initially provide sufficient evidence that the design met UK relevant good practice in terms of reducing risks As Low As Reasonably Practicable (ALARP). Following a detailed ONR assessment, ONR challenged the presented evidence, which resulted in Hitachi revising its design and presenting further evidence. ONR judged that these design improvements supported by the additional evidence were then sufficient to satisfy ONR that the design was satisfactory and met UK relevant good practice.

This assessment facilitated an in depth examination of Hitachi-GE's classification of safety functions and categorisation of structures, systems and components to determine if they have been appropriately assigned. The assessment considers whether Hitachi has appropriately applied a graded approach to safety, to ensure that it has proportionately focused attention to design, procurement, operation, and maintenance activities on structures, systems and components, particularly those with higher safety importance.

Assessment findings are part of the GDA process and provide a mechanism for ONR assessors to identify areas where the requesting party has not presented sufficient evidence at this stage. ONR's expectation is that a licensee should address these findings in its future safety case submissions as part of normal business. For example, ONR found that in some instances, Hitachi-GE had not provided sufficient evidence at this stage, to demonstrate adequate mechanical equipment qualification, appropriate equipment diversity and established relevant good practice. Annex 5 of this report provides a full list of ONR mechanical engineering assessment findings.

In the absence of detailed design information during GDA, there are examples where ONR makes certain assumptions. In a similar manner as assessment findings, it is ONR's expectation that a future licensee should address these assumptions in its safety case submissions as part of normal business. For example, ONR acknowledges that during GDA details of plant limits and conditions, Examination, Inspection, Maintenance and Testing (EIMT) requirements and engineering qualification requirements are not available. Annex 7 of this report provides a full list of ONR mechanical engineering assumptions.

To conclude, I am satisfied with the claims, arguments and evidence laid down within the Pre-Construction Safety Report (PCSR) and supporting documentation. I consider that from a mechanical engineering perspective, the Hitachi-GE UK ABWR generic design is suitable for construction in the UK subject to the specific site licensees securing the necessary ONR permissions and EA or NRW permits.

LIST OF ABBREVIATIONS

ALARP	As Low As Reasonably Practicable
BSL	Basic Safety Level
BSO	Basic Safety Objective
BWR	Boiling Water Reactor
C&I	Control & Instrumentation
CCF	Common Cause Failure
CR	Control Rod
DAC	Design Acceptance Confirmation
DAG	Diverse Additional Generator
DBA	Design Basis Analysis
EA	Environment Agency
EIMT	Examination, Inspection, Maintenance and Testing
ECCS	Emergency Core Cooling System
EDG	Emergency Diesel Generator
FHM	Fuel Handling Machine
FLSR	Flooding System of Reactor Building
FLSS	Flooding System of Specific Safety Facility
FMCRD	Fine-Motion Control Rod Drive
FMEA	Failure Mode and Effect Analysis
FPC	Fuel Pool Cooling and Clean-up System
FPM	Fuel Preparation Machines
GDA	Generic Design Assessment
HCU	Hydraulic Control Unit
HVAC	Heating Ventilation and Air Conditioning System
IAEA	The International Atomic Energy Agency
J ABWR	Japan Advance Boiling Water Reactor
KIT	Keep In Touch
LCO	Limiting Condition for Operation
LOLER	Lifting Operations and Lifting Equipment Regulations
LOCA	Loss of Coolant Accident
LOOP	Loss of Off-site Power
MS	Main Steam
MSIV	Main Steam Isolation Valve
NFIS	New Fuel Inspection Stand
NRW	National Resources Wales
NSEDP	Nuclear Safety and Environmental Design Principles

ONR	Office for Nuclear Regulation
OPEX	Operational Experience
P&ID	Piping and Instrumentation Diagram
PCSR	Pre-construction Safety Report
PCV	Primary Containment Vessel
PLC	Programmable Logic Controller
PSA	Probabilistic Safety Assessment
PWR	Pressurised Water Reactor
R/B	Reactor Building
RBC	Reactor Building Overhead Crane
RCIC	Reactor Core Isolation Cooling
RCIS	Rod Control Information System
RCW	Reactor Building Cooling Water System
RGP	Relevant Good Practice
RHR	Residual Heat Removal System
RI	Regulatory Issue
RIP	Reactor Internal Pump
RO	Regulatory Observation
RPV	Reactor Pressure Vessel
RQ	Regulatory Query
RSW	Reactor Building Service Water
SAPs	Safety Assessment Principles
SFAIRP	So Far As Is Reasonably Practicable
SFC	Safety Functional Claim
SFP	Spent Fuel Pool
SGTS	Standby Gas Treatment System
SLCS	Standby Liquid Control System
SoDA	Statement of Design Acceptability
SPC	Safety Property Claim
SRV	Safety Relief Valve
SSC	System, Structure, (and) Component
TAG	Technical Assessment Guide
TSC	Technical Support Contractor
UK	United Kingdom
UK ABWR	United Kingdom Advanced Boiling Water Reactor
WENRA	Western European Nuclear Regulators’ Association

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1 INTRODUCTION

1.1 Background

1. Generic Design Assessment (GDA) is the process applied where a requesting party asks ONR to assess a reactor design in advance of an application for a nuclear site licence. GDA separates design issues from specific site related issues, which is likely to be beneficial where the licensee intends to construct the generic design on a number of different UK sites.

2. The outcome from a GDA process sought by requesting parties such as Hitachi-GE is a Design Acceptance Confirmation (DAC) from ONR and a Statement of Design Acceptability (SoDA) from the Environment Agency (EA) or Natural Resources Wales (NRW). Further information on the GDA process in general is also available on our website [1]

3. Hitachi-GE as a requesting party, commenced step 1 of GDA in 2013 by proposing a United Kingdom version of their Advanced Boiling Water Reactor (UK ABWR). ONR has published its step 3 summary report for this GDA on our website [2]

4. Hitachi-GE's evidence to underpin its safety claims and arguments was incomplete at the end of step 3 GDA in the area of mechanical engineering. This is common during GDA and ONR outlined recommendations to Hitachi-GE on how to address this shortfall via the mechanical engineering GDA step 3 assessment report [3]. These recommendations informed the scope of my step 4 assessment which is detailed in the assessment plan for mechanical engineering step 4 [4].

5. Step 4 of GDA is the final step and Hitachi-GE completed this for its UK ABWR in 2017. Step 4 is an in-depth assessment of the safety, security and environmental evidence. Through the review of information provided to ONR, the Step 4 process should confirm that the requesting party, in this case Hitachi-GE:

   - Has properly justified the higher-level claims and arguments
   - Has progressed the resolution of issues identified during Step 3
   - Has provided sufficient detailed analysis to allow ONR to come to a judgment of whether a DAC can be issued

6. This report presents my findings, conclusions and recommendations from my Step 4, mechanical engineering assessment of Hitachi-GE's UK ABWR design.

7. ONR publishes any Regulatory Observations (ROs), issued to Hitachi-GE on its website, together with the corresponding Hitachi-GE resolution plans.

1.2 Scope

8. During step 4, I have undertaken a detailed mechanical engineering assessment, on a sampling basis of Hitachi-GE's safety case evidence for UK ABWR. My assessment is limited to nuclear safety aspects. It does not cover security and environmental aspects. ONR's GDA Guidance to Requesting Parties [5] covers the full range of items that might form part of the assessment. These include:

   - Consideration of issues identified in Step 3
   - Judging the design against the ONR Safety Assessment Principles (SAPs) and whether the proposed design reduces risks As Low As Reasonably Practicable (ALARP)

- Reviewing details of Hitachi-GE's design controls, procurement and quality control arrangements to secure compliance with the design intent
- Establishing whether the detailed engineering design substantiates the system performance, safety classification, and reliability requirements
- Assessing arrangements for ensuring and assuring that safety claims and assumptions are realised in the final as-built design.
- Resolution of identified nuclear safety and security issues, or identifying paths for resolution.

9. I have based my step 4 mechanical engineering assessment, of UK ABWR, on the relevant chapters of Hitachi-GE's Pre Construction Safety Report (PCSR) and its supporting references. My objective was to conduct an in-depth assessment of the safety case evidence presented by Hitachi-GE, to underpin the safety claims and arguments examined during Step 2 and 3 of GDA and to review any previously identified shortfalls.

10. I have reviewed and developed my scope as appropriate through liaison with other assessment disciplines. I have also followed additional lines of enquiry as issues have emerged through progression of the initially identified assessment scope.

11. The type of information assessed during Step 4 included:

    - concept optioneering study reports;
    - concept design review minutes (single and multidiscipline);
    - research and development reports;
    - concept/product qualification test data;
    - technical specifications, drawings, Piping and Instrumentation Diagrams (P&ID), calculations etc.
    - Examination, Inspection, Maintenance and Testing (EIMT) regimes;
    - design justification reports;
    - illustrative life time quality records;
    - illustrative factory acceptance test data; and
    - illustrative commissioning test data.

12. In terms of documentation, the overall basis for the Step 4 assessment of the evidence related to mechanical engineering for the UK ABWR is:

    - Selected chapters of the PCSR and supporting documentation.
    - The UK ABWR GDA Master Document Submission List relevant for commencement of Step 4 [6] and any subsequent updates agreed between Hitachi-GE and ONR.
    - The UK ABWR GDA Design Reference Point established at the end of Step 3 and any subsequent updates agreed between Hitachi-GE and the regulators ONR.

- Responses to the Regulatory Queries (RQ) issued from my assessment of the Step 2 and 3 mechanical engineering submissions. A full list of these RQ's is included in [7]

- Deliverables in response to the outstanding Regulatory Observations (RO) issued from my assessment of the Step 2 and 3 mechanical engineering submissions.

- Deliverables in response to crosscutting ROs issued during Step 2 and 3 that are of interest to mechanical engineering assessment.

1.3 Interpretation

13. For this mechanical engineering assessment, "safety claim" is interpreted as being the ability of a Structure, System or Component (SSC) to deliver its safety function during normal operations (including reactor shutdown), fault sequences and accident conditions, with adequate consideration of the following characteristics:

- Inherent safety – hazard avoidance, in preference to hazard control (*SAP. EKP. 1*)

- Fault tolerance – sensitivity to potential faults to be minimised (*SAP EKP. 2*)

- Defence in depth – provision of adequate levels of protection (*SAP. EKP. 3*)

- Safety function – structured fault analysis undertaken for both normal operation (including shutdown), and fault sequences (*SAP. EKP. 1*) and

- Safety measures – should be identified to deliver the required safety function (*SAP. EKP. 5).*

14. For mechanical engineering, "safety argument" is interpreted as being the robust auditable, rational basis:

- in specifying an SSC safety function (*SAP series: ECS and ERL*)

- that an SSC is able to secure its safety function (*SAP series: EDR, EAD, ELO, EMT, EQU, and ECM*) and

- to demonstrate that an SSC has been sufficiently optioneered to show hazards are adequately controlled and risks have been reduced So Far As Is Reasonably Practicable (SFAIRP)) (*SAP series: EDR, EAD, ELO, EMT, EQU, and ECM*).

15. For my mechanical engineering assessment, "safety evidence" is interpreted as being the robust auditable design substantiation that underpins a SSCs design basis.

1.4 Method

16. My assessment complies with internal guidance on the mechanics of assessment within ONR:

- Guidance on demonstration of ALARP [8]

- Guidance on production of reports [9]

2 ASSESSMENT STRATEGY

2.1 Pre-Construction Safety Report (PCSR)

17. ONR's GDA Guidance to Requesting Parties (http://www.onr.org.uk/new-reactors/ngn03.pdf) states that the information required for GDA may be in the form of a Pre-Construction Safety Report (PCSR).

18. Technical Assessment Guide (TAG) 051 sets out regulatory expectations for a PCSR (http://www.onr.org.uk/operational/tech_asst_guides/ns-tast-gd-051.pdf).

2.2 Standards and criteria

2.2.1 Safety Assessment Principles

19. The key Safety Assessment Principles (SAPs) applied within this assessment are included within annex 1.

2.2.2 Technical Assessment Guides

20. The key ONR Technical Assessment Guides (TAGs) that I have used as part of this assessment are set out in annex 2

2.2.3 National and international standards and guidance

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

2.3 Use of Technical Support Contractors (TSCs)

22. During GDA ONR may engage with Technical Support Contractors (TSC's). For example to provide additional capacity, to enable access to independent advice and experience, to analyse techniques and models, and to access additional resources so that ONR inspectors can focus on regulatory decision making. For this mechanical engineering step 4 assessment, TSC support provided additional capacity. The scope of work at step 4 was to follow up a sample of those areas previously reviewed by the TSC at step 3, targeting the adequacy of evidence presented at step 4. This was achieved by sampling the Hitachi-GE submissions for the following systems, structures and components (SSCs):

- Safety Relief Valves
- RHR Heat Exchanger
- Main Steam isolation Valves

23. The TSC findings where reviewed by ONR mechanical engineering inspectors and the outcome of this review has been reported in this step 4 report as appropriate.

2.4 Integration with other assessment topics

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

- Severe Accidents.
- Categorisation and Classification.
- Heating, Ventilation and Air Conditioning (HVAC)
- Fuel Export Lifting and Handling
- Testing and Maintenance of Safety Systems

- Additional electrical power source
- radiolysis gases generated under normal operations
- Design Basis Analysis of essential services and support systems
- Operational Experience

25. During my mechanical engineering assessment, I have interfaced with other ONR inspectors who are experts in discipline areas including fault studies, probabilistic safety assessment, internal hazards and other engineering specialisms. Coordination with other disciplines has occurred as part of the normal assessment process achieved through regular interactions across the ONR GDA team, local discussion, regular Keep in Touch (KIT) meetings and cross discipline Level 4 meetings with Hitachi-GE. Given the sampling nature of assessment, this process has proved to be effective and efficient in determining the adequacy of safety cases, and identifying areas of weakness for further resolution.

26. The ability for a mechanical engineering SSC to deliver its safety function and the requirements placed on it by the safety case is the subject of this step 4 mechanical engineering assessment report. However, the completeness with which Hitachi-GE have identified appropriate safety functions and the level of engineering protection required to deliver those functions has primarily been led by ONR fault studies specialists. Hence, mechanical engineering assessors have worked closely with fault studies assessors to ensure there has been adequate consideration of SSC's and their claimed safety function.

2.5 Sampling strategy

27. A common way to assess a safety case is by identifying the claims on Structures, Systems, Components (SSC) and people. This offers a 'top down' approach from safety claim to detailed assessment. The nature of mechanical engineering, and its associated assessment, favours an alternative 'bottom up' type approach. In this assessment method, assessors identify mechanical items important to safety then assess them based on their safety function and safety classification.

28. It is seldom possible, or necessary, to assess a safety case in its entirety. Therefore, ONR adopts a sampling approach to limit the areas scrutinised and improve the overall efficiency of the assessment process. ONR seeks to identify any weaknesses in the safety case by adopting a focused and targeted sampling strategy. This method of assessment is in accordance with ONR internal guidance on the "Purpose and Scope of Permissioning" [10].

29. My mechanical engineering assessment strategy encompasses SSCs that generally contain dynamic elements and interfaces. This is different from the discipline of structural integrity, which is concerned with SSCs that are static in nature. These static SSCs primarily provide a containment safety function pressure boundary. Notwithstanding this definition, a number of static components will also be of interest to the mechanical engineering discipline, and subject to appropriate assessment.

30. Examples of mechanical engineering dynamic SSCs are:

- Control Rod Drive Mechanisms
- Pumps
- Valves (check valves, motor operated valves, safety relief valves, squib valves, and isolation valves)

- Cranes
- Mechanical Handling Systems
- Nuclear Ventilation systems used to augment nuclear containment barriers.
- Heating Ventilation and Air Conditioning (HVAC)
- Emergency diesel generators

31. Examples of mechanical engineering static SSCs are:

- Heat exchangers
- Gloveboxes and cabinets
- Penetrations (e.g. through civil structures)

2.6 Out of scope items

32. Hitachi-GEs document on design status of mechanical SCCs [11] along with PCSR chapter 1 [12] sets out the items agreed as being outside the scope of GDA.

33. Hitachi-GE’s overall quality plan for its design processes.

3 REQUESTING PARTY'S SAFETY CASE

34. Hitachi-GE has identified the generic PCSR as the key submission within GDA that outlines the reasons supporting its top level claim that the "UK ABWR constructed on a generic site within the United Kingdom, can be operated safely under all operating and fault conditions.

35. The PCSR has 32 chapters. The following chapters are most relevant to ONR's Mechanical Engineering assessment:

- Chapter 1: Introduction [12]
- Chapter 5: General Design Aspects [27]
- Chapter 17.3: Turbine Main Steam, Turbine Auxiliary Steam and Turbine Bypass System. [68]
- Chapter 18: Radioactive Waste Management [187]
- Chapter 19: Fuel Storage and Handling [41]

36. While the PCSR is clearly a vital and fundamental part of the UK ABWR safety case, the claims and arguments made within are relatively high level, as expected. Sitting beneath the PCSR (and referenced from it) are a large number of Topic Reports and Basis of Safety Case Reports. It is these documents (and supporting references from these reports) which have been main areas for mechanical engineering assessment during GDA Step 4 and provide the technical basis for most of the regulatory judgements included in this report.

37. The nature of GDA is such that not all information is available to ONR until the future licensee commences detailed design together with site-specific considerations. Such information is not necessary to complete a successful Generic Design Assessment. For example, during this step 4 assessment, Hitachi-GE's design processes were assessed to seek assurance that the site specific, detailed mechanical engineering design, will meet UK expectations during future detailed design.

38. Hitachi-GE's PCSR does not collate all information relevant to mechanical engineering as a separate topic. Therefore, I have identified the equipment and processes selected for assessment in Section 4 of this report.

39. Throughout my mechanical engineering assessment, there have been regular level 4 technical interactions with Hitachi-GE. During these meetings, mechanical engineering assessors have challenged the depth and nature of the mechanical engineering design by suitable questioning and examination of appropriate design information including basis of safety cases (BoSC) and topic reports for specific systems.

40. Hitachi-GE has continued to develop its safety case throughout step 4. Where possible I have based my assessment on the latest document revision available at that time. However, towards the end of step 4, Hitachi-GE undertook an extensive review of its safety case documentation to ensure it has provided an accurate account of the design prior to handing it over to the licensee. Hitachi-GE prepared a list of changes [13] related to mechanical systems grouped according to the nature of the change. Hitachi-GE identified 15 groups of change ranging from minor document formatting changes through to changes in design or claims, arguments and evidence. I considered these changes and I am satisfied that Hitachi-GE's recent document revisions do not affect my assessment findings reported below.

41. Hitachi-GE's safety case for Mechanical engineering is documented in Table 1 below:

Table 1 Requesting party Safety Case

Submission Title	Content	Link to Assessment Scope	Summary Arguments
UK ABWR Pre-Construction Safety Report	For Mechanical Engineering Hitachi-GE's Generic Pre-Construction Safety Report (PCSR) is a summary document. It aims to set out all the systems that underpin the safety requirements of the safety case. It adopts a hierarchical approach; setting out how the top level and the general plant level claims are cascaded to the specific Mechanical Engineering SSCs.	For mechanical engineering safety claims are interpreted as being the ability of an SSC to deliver its safety function during normal operations (including for shutdown), fault sequences, and accident conditions.	Not applicable

Submission Title	Content	Link to Assessment Scope	Summary Arguments
Level "2" Safety case Basis of Safety Case documents	For mechanical engineering Level "2" documents are system based that are of a principal interest to mechanical engineering. They aim principally to set out and substantiate the required level of engineering. Proportionate to the assessment phase each BoSC document aim is to extract the safety analysis (deterministic and probabilistic) claims from the Level "1" safety analysis to define the SSCs' safety functional requirements, reliability, performance demands and the level of engineering required to deliver the assigned safety classification.	As above.	Proportionate to the assessment phase each BoSC document has specific sections that aims to set out: ■ A system overview; safety role, safety function, basis of configuration and modes of operation ■ Each SSC plant's safety basis claims ■ Each SSC design rationale arguments and the design substantiation evidence that underpins the safety design basis claims; design engineering safety functions, reliability and performance requirements; safety categorisation and classification; assigned codes and standards; qualification and Examination; Inspection; Maintenance and Testing (EIMT) requirements.
Level "2" Mechanical Engineering SSCs Topic Report	A strategy document that aims to set out Hitachi-GE's design approach and principles to secure an SSC design basis.	As above.	Considered a high-level strategy document that aims to apply a consistent approach and principles to secure each SSC design basis.

4 ONR STEP 4 ASSESSMENT UNDERTAKEN

4.1 Assessment of Hitachi-GE's Design Process

4.1.1 Introduction

42. I undertook a sampled assessment of Hitachi-GE's design process, to ensure it has robust design practices in place. I consider this an important aspect that underpins the safety justification of the UK ABWR design. In undertaking my assessment, I have used the internal technical assessment guide, Design Safety Assurance [14], to guide my assessment strategy and conclusions.

43. The following Safety Assessment Principle (SAP) is also relevant to this aspect:

- EQU.1 states 'Qualification procedures should be in place to confirm that structures, systems and components that are important to safety will perform their required safety function(s) throughout their operational lives.'

4.1.2 Overview of Hitachi-GE's Design Process Safety Case

44. ONR SAPs and Technical Assessment Guides (TAGs) provide inspectors with a framework for making consistent regulatory judgements on the safety of activities. However, they are not sufficient on their own for designers to use as design or operational standards. Therefore, Hitachi-GE has developed its Nuclear Safety and Environmental Design Principles (NSEDPs) [15]. These provide Hitachi-GE with a framework of acceptance criteria by which the adequacy of the design will be judged to ensure that the risks arising from all aspects of UK ABWR lifecycle are reduced As Low As Reasonably Practicable (ALARP).

45. It is beyond the scope of my assessment to assess Hitachi-GE's NSDEPs in detail. However, based on my sample assessment of the design of various systems, I am satisfied that Hitachi-GE's NSEDPs should result in engineering design principles that are equivalent to the guidance set out in ONR's SAPs.

4.1.3 Strategy for Assessment of Hitachi-GE's Design Process

46. As part of my assessment, I targeted a number of key areas seeking evidence to confirm Hitachi-GE is a responsible designer capable of managing a design process for safety critical equipment. This subject has been progressed as part of the lines of enquiry and assessment for the various mechanical engineering equipment types and I have reported these in the sections of this report. In particular, I consider that Hitachi-GE should have:

- Adequate arrangements in place to ensure adequate compilation and transfer of safety functional requirements to inform detailed design and manufacturing.

- Adequate arrangements to enable the identification and transfer of the plant operating limits and conditions to the licensee. This should be sufficient to allow it to undertake its regulatory duties and to generate adequate plant operating rules and procedures.

- Adequate arrangements for the identification of items important to safety through an appropriate equipment safety classification process. This should be sufficient to allow adequate design and procurement, and generation of a Plant Maintenance Schedule to support the Examination, Inspection, Maintenance and Testing (EIMT) requirements.

- Adequate arrangements to manage interdisciplinary requirements and interfaces with the necessary degree of Quality Assurance.

4.1.4 Assessment of Hitachi-GE's arrangements for Design

47. During step 2 GDA ONR raised RO–ABWR-016 – Mechanical Engineering Design Process Arrangement. This is a cross cutting observation involving conventional safety, structural integrity, civil engineering, Control and Instrumentation (C & I), electrical and Management for Safety and Quality Assurance (MSQA). ONR issued RO–ABWR-016 because it considered that Hitachi-GE had not provided adequate assurance of a robust arrangement, for its conceptual design process, in line with UK relevant good practice (RGP). Hitachi-GE's response is summarised in an RO summary report [16].

48. In response to RO-ABWR-016, all Hitachi-GE's mechanical design sections produced a list of SSC's applicable to the UK ABWR safety case. Each section provided internal conceptual design procedures, rules and manuals related to the design of the SSC's. Hitachi-GE then undertook a gap analysis between the design of J-ABWR and applicable UK legislation and RGP. Hitachi-GE also considered Operational Experience (OPEX), as observed in RO-ABWR-045, as part of the general design process arrangements. Hitachi-GE concluded that there was a gap with its ALARP philosophy and ONR indicated that Hitachi-GE's general design process was not in a format auditable by regulators. Hitachi-GE subsequently revised its general design process [17]. ONR sampled a number of SSC's described below to ensure that Hitachi-GE was adequately applying its process. Once ONR was satisfied that Hitachi-GE had provided sufficient evidence, it closed RO-ABWR-016.

49. ONR also sought evidence that Hitachi-GE had adequately demonstrated that its detailed design arrangements were capable of reducing the risk so far as is reasonably practicable. ONR raised RO-ABWR-0052 requiring Hitachi-GE to demonstrate how it intends to reduce the risk so far as is reasonably practicable and demonstrate that the detailed design of each SSC is ALARP. Hitachi-GE revised its concept design process which underpins the detailed design Hitachi-GE shared its revised design process with all its senior mechanical design engineers and training on its application was provided to Hitachi-GE's designers. Hitachi-GE reviewed all its respective designs in accordance with this revised design process and amended design manuals for specific SSCs as necessary. Hitachi-GE also shared this design approach across all of ONR's engineering disciplines.

50. I assessed the revised 'design process approach document' [18] . The document indicates that UK ABWR is broadly similar to the J-ABWR reference design modified to meet UK and international requirements. The arrangements for applying a gap analysis against UK and international requirements are set out and a process to identify and understand deviations is included. Based on Hitachi-GE's revised detailed design process ONR subsequently closed RO-ABWR-0052.

51. Hitachi-GE has divided the operating stages of the plant life cycle into discrete stages including planning, siting, design, construction, commissioning, operation and decommissioning. I am satisfied that Hitachi-GE has considered each stage and where appropriate has subdivided the stages. For example, it has considered various operating and shutdown conditions that might affect system requirements differently.

52. In accordance with the ONR SAP EKP.1, an 'inherently safe' design is one that avoids radiological hazards rather than controlling them. It prevents a specific harm occurring by using an approach, design or arrangement that ensures that the harm cannot happen. Where inherently safe design is not achievable, the design should be fault tolerant such that faults and other disturbances to normal operating should not lead to unacceptable consequences. Hitachi-GE's design philosophy for mechanical plant is

to offer several layers of protection from unacceptable consequences in normal operating conditions and following faults. Hence, from the systems sampled I am satisfied that Hitachi-GE has applied this philosophy.

4.1.5 Assessment Sample for Design Process

53. Early in the step 4 assessment, ONR indicated to Hitachi-GE that it wished to select four sample systems where Hitachi-GE had applied its revised design philosophy. This enabled ONR to close out RO-ABWR-016 as described above. The four representative samples selected and the subsequent assessment findings are described, as follows:

- Design of cranes
- Design of wet lifting beams
- Design of flexible hoses
- Design of piping (with respect to ensuring pipe runs have sufficient gradient)

54. **Design of Cranes and design of wet lifting beams-** Hitachi-GE did not initially produce a lifting schedule as part of its design documentation which was identified as a design shortfall by ONR compared with UK RGP. Hitachi-GE's revised design approach resulted in design manuals dedicated to mechanical lifting devices prepared with assistance from a UK consultancy. I sampled the design manual for safety related mechanical lifting devices [19] and the design manual for wet lifting beams [20]. I also sampled the basis for safety case on fuel handling systems and overhead crane systems [21] and a topic report on lifting attachments [22]. Detailed comments on these documents are provided later in this report but I am satisfied that they have been prepared in accordance with the design approach and include consideration of UK legislation and RGP.

55. **Design of flexible hoses -** I assessed Revision 2 of Hitachi-GE's piping design specification [23]. I considered that Hitachi-GE provided sufficient detail in its piping design specification [23] to demonstrate that the design of flexible hoses and piping gradient is adequately covered.

56. The requirement to use flexible hoses is determined through a risk assessment against the J-ABWR reference design, in accordance with the piping design specification. Hitachi-GE has produced flexible hose schedules, which list all the required flexible hoses with their design parameters. These schedules enable management of hoses and provide parameters for detailed design and procurement. I assessed an example of a flexible hose schedule [24] and was satisfied with its format.

57. **Design of piping -** During the sixth technical exchange meeting [25], Hitachi-GE described its ALARP assessment process for determining pipe gradient [26]. The process described sets out fluid properties, pipe layout and fluid phases necessary to determine the risk. Hitachi-GE then compares the risk with UK RGP and following an ALARP review, a decision is made to determine if a pipe gradient is necessary. I am satisfied that this process offers a systematic approach to determining where a piping gradient is necessary.

58. I am satisfied that the four design areas considered above provide me with sufficient evidence that Hitachi-GE has a general design process that is in an auditable format by regulators. The detail of this process may need further scrutiny by ONR during detailed design. However, I am satisfied that Hitachi-GE provided an adequate response to RO-ABWR-0016 and RO-ABWR-052 both of which were subsequently closed in their entirety.

4.1.6 Assessment of Hitachi-GE's Equipment Qualification Process

59. ONR TAG NS-TAST-GD-057 [14] states that in addition to categorising / classifying designs in accordance with the ONR SAPs, physical parameters should be established to define design duties, such as temperature, load combinations, damage and ageing mechanisms, environmental considerations, design life considerations etc. This links to the overall consideration of equipment qualification.

60. SAP EQU.1 says that, Qualification procedures should be applied to confirm that structures, systems and components would perform their allocated safety function(s) in all normal operational, fault and accident conditions identified in the safety case and for the duration of their operational lives.

61. Chapter 5 of the PCSR, Generic Design Aspects, [27] indicates that for UK ABWR mechanical equipment with a high safety importance (Safety Class 1 or 2) is qualified according to the equipment qualification (EQ) process. The methods of qualification are summarised as:

- Performance of a type test on representative equipment to be supplied
- Performance of an actual test on the supplied equipment
- Application of pertinent past experience in similar applications
- Analysis based on reasonable engineering extrapolation of test data or operating experience under pertinent conditions
- An appropriate combination of these four methods

62. I consider that this proposed qualification process is a suitable means of verifying that mechanical equipment with a safety function can deliver its design intent in normal and fault conditions.

63. Hitachi-GE proposes to evaluate the validity and effectiveness of equipment by means of qualification tests, analysis or comparative evaluation of past J-ABWR qualification data. Hitachi-GE has indicated that the intention is to apply test conditions such as environment simulation of severe accidents, postulated accidents, and transient conditions. Hitachi-GE will then specify dynamic loads, static loads and functional requirements for equipment in each design specification. For mechanical equipment and systems, such detailed information is not available during GDA although I am satisfied that such detail can be generated during detail design. Hitachi-GE has indicated that equipment not related to dynamic function qualification, such as pressure components and support structures manufactured in accordance with ASME codes are not subject to qualification tests by analysis since the design report takes into account load combinations. Hitachi-GE has provided lists of applicable codes for such equipment and I am satisfied with this philosophy.

64. Chapter 5 of Hitachi-GE's PCSR [27] indicates that Hitachi-GE will compile a report of qualification tests, analysis or evaluation of past qualification data once it has clarified its qualification method (qualification test/analysis). I consider that this qualification documentation should ensure the traceability of product or installation delivered to the nuclear power plant.

65. The J-ABWR reference design has a 40-year design life whereas Hitachi-GE has increased the UK ABWR design life to 60 years. During step 2 GDA ONR issued RO–ABWR-015 Mechanical Engineering SSC's Qualification and Layout Provision. This was issued because ONR considered Hitachi-GE had provided limited assurance that;

- It had considered the need to undertake additional qualification of SSCs to support the 60 year design life claim, and,

- Building layout design was adequate to replace SSCs that are to retain a 40-year design life or less.

66. Hitachi-GE's response is summarised in its RO summary report [28]. In response to RO-ABWR-0015, Hitachi-GE developed [29], its strategy to determine the operational design life and replacement frequency of UK-ABWR SSCs. Hitachi-GE's mechanical engineering design sections developed their strategy that sets out the documented process for evaluating the design life of an SSC. The process determines one of the following outcomes;

- The current J-ABWR design is qualified for the expected operational period for the UK ABWR;
- Re-qualification is necessary to meet the expected operational period for the UK ABWR;
- The SSC cannot be qualified for the operational period of the UK ABWR and periodic replacement is required.

67. If option 1 is the outcome Hitachi-GE will justify in its safety case that existing qualification evidence bounds the 60-year life. If Option 2, Hitachi-GE will develop equipment qualification plans for that SSC. If Option 3, the layout designers in conjunction with the SSC designers will ensure the following conditions can be met;

- route for SSC replacement exists with adequate ingress and egress;

- SSC replacement provision exists e.g. lifting equipment or modular disassembly;

- limitations on maximum SSC dimensions are determined.

68. Hitachi-GE recognises that the need for replacement of equipment procured from external suppliers may be unknown in GDA due to a manufacturer not yet being selected or the concept design not sufficiently developed. For these cases, Hitachi-GE assumes that replacement is necessary and the suitability of the layout to enable replacement is considered.

69. I consider the above arrangements to be adequate in determining the design life of an SSC that in turn drives the requirements for equipment qualification. This meets guidance in ONR SAP ELO.1 that states, "The layout should make provision for construction, assembly, installation, erection, decommissioning, maintenance and demolition.

70. ONR recognises that further detailed considerations for SSC qualification and layout provision will occur throughout the detailed design process. However, I consider the above high-level arrangements, for determining SSC design life, are adequate to close RO-ABWR-0015 [30].

71. Hitachi-GE proposes that the licensee, in accordance with the Qualification Plan, carries out qualification tests. For qualification of equipment with a significant safety function, qualification by a third party inspection agency might be included, depending on the importance of the items requiring qualification.

72. Hitachi-GE indicated that it has designed mechanical equipment in accordance with ISO, BS and European standards. However, nuclear specific equipment such as the fine motion control rod drives, hydraulic control units and reactor internal pumps are designed to the manufacturer's standards. The detail design of these items needs to

be justified prior to construction. I have identified an assumption that Hitachi-GE shall provide sufficient information to the licensee (AS-ABWR-ME02).

4.1.7 Assessment of Hitachi-GE's Design Change Control Process

73. The ONR TAG on design safety assurance [14] indicates that changes to frozen design information should be formally justified and the implications assessed before thorough integration of the changes into the modified design.

74. Hitachi-GE has implemented a six-step process that applies to design changes taking place after the reference point in GDA. Hitachi-GE describes this process in its generic design development control document [31] as being in accordance with its overall quality plan. ONR's Quality Assurance specialists have undertaken assessment of Hitachi-GE's overall quality plan, including the design change process, throughout GDA.

4.1.8 Assessment of Hitachi-GE's Safety Classification Process

75. Categorisation of the required safety function and classification of any associated equipment to deliver this function are crosscutting issues, covering several disciplines. In terms of mechanical engineering, they are an important input to the definition of design requirements, procurement processes (specifically assurance activities), installation and commissioning, and Examination, Inspection, Maintenance and Testing (EIMT) activities.

76. The following SAPs are relevant to this aspect:

- SAP ECS.1 (Ref. 4) states 'The safety functions to be delivered within the facility, both during normal operation and in the event of a fault or accident, should be categorised based on their significance with regard to safety.'
- SAP ECS.2 (Ref. 4) states 'Structures, systems and components that have to deliver safety functions should be identified and classified on the basis of those functions and their significance with regards to safety.'

77. Hitachi-GE has described its process for a systematic and comprehensive identification of faults and their categorisation according to their potential unmitigated consequences and frequency. This process is in line with fundamental safety functions identified by International Atomic Energy Association (IAEA). Hitachi-GE makes design provisions for each safety function and the resultant safety measures are classified according to their importance in delivering the associated safety function(s).

78. Hitachi-GE use the classification to ensure that SSCs are designed and operated using codes, standards and procedures commensurate with their importance for safety as expressed in their safety classification and the categorisation of the safety function(s) they deliver.

79. Hitachi-GE has presented the categorisation and classification scheme used in the UK ABWR safety documentation based on guidance given in their NSEDPs.

80. ONR, as part of its system sample, has considered Hitachi-GE's application of categorisation and classification. I have reported these findings later in this report.

4.1.9 Assessment of Hitachi-GE's Safety Function and Safety Property Claims

81. Chapter 5 of Hitachi-GE's PCSR [27] indicates that Safety Property Claims (SPC)s are principles applied by Hitachi-GE when designing SSCs in accordance with their Nuclear Safety and Environmental Design Principles (NSEDP)s [15]. Safety

Functional Claims (SFCs) are claims made on SSCs based on wide ranging and comprehensive fault analysis. Many of the mechanical systems I assessed fulfil multiple SFCs whereas the SPCs will be unique to the SSC and might for example include properties such as redundancy, diversity and environmental qualification. I am satisfied that individual safety cases for the systems sampled have clearly set out the SFCs and SPCs. Hence, I conclude that Hitachi-GE have demonstrated a satisfactory application of their design process for each system sampled.

4.1.10 Comparison of UK ABWR with J-ABWR Reference Design

82. I assessed the 'design process approach document' [64]. The document indicates that UK ABWR is broadly similar to the J-ABWR reference design modified to meet UK and international requirements. The arrangements for applying a gap analysis against UK and international requirements are set out and a process to identify and understand deviations is included. The approach outlines the phases of design and I am satisfied that it provides the framework to cover design from concept through detailed design and manufacture to commissioning and eventual decommissioning.

4.1.11 Assessment of Hitachi-GE's Decommissioning Arrangements

83. ONR recognises that during GDA it is not possible to define detailed design requirements for each system. However, Hitachi-GE has identified certain systems as important during decommissioning. Decommissioning requirements may include considering how the licensee can design Structures, Systems and Components (SSC) for decommissioning. Furthermore, an SSC may provide an important function to facilitate decommissioning. For example, the Basis of Safety Case (BoSC) indicates that building cranes might be used to support decommissioning. Hitachi-GE has not provided all requirements for decommissioning in the basis for safety cases or topic reports. However, I am satisfied that the licensees can establish this detail during detail design. I have identified an assumption that should be considered by a future licensee (AS-ABWR-ME04).

4.1.12 Conclusions

84. I am satisfied that my assessment strategy provides me with an adequate sample to identify Hitachi-GE's application of its design process, to ensure it has robust design practices in place.

85. I consider that Hitachi-GE has adopted a suitable design process that should enable it to satisfy the requirements of legislation and RGP.

86. I have identified two assumptions to ensure that detailed design adequately considers equipment qualification and decommissioning requirements.

4.1.13 Regulatory Assumptions

87. **AS-ABWR-ME02 -** ONR acknowledges that during GDA, Hitachi-GE had not sufficiently developed its design to enable the licensee to identify the appropriate design qualifications details. ONR assumes that the licensee shall establish detailed design substantiation, factory acceptance test information and site acceptance test information for individual mechanical items and their associated systems, which are important to safety. ONR also assumes that the licensee shall generate appropriate evidence that equipment qualification is adequately specified for all mechanical items important to safety in accordance with UK expectations.

88. **AS-ABWR-ME04 -** During GDA, Hitachi-GE identified certain equipment that is required to perform activities associated with decommissioning of the plant at the end of its 60-year life. ONRs expectation is that the licensee will provide suitable plant from

the outset to avoid unnecessary modifications to plant in future prior to decommissioning. ONR assumes that the licensee shall identify and confirm the use of equipment for decommissioning shall consider design features to facilitate decommissioning and reduce future dose uptake by workers and where reasonably practicable include any necessary design features in the final design.

4.2 Assessment of Limits and Conditions including Examination, Inspection, Maintenance and Testing requirements

4.2.1 Introduction

89. A key feature of a safety case is the identification of the limits and conditions that define the safe operating envelope for plant operation. In the UK, these limits and conditions are termed ‘operating rules’ (Licence Condition 23) and are the responsibility of the licensee. However, it is important that designers generate sufficient technical information and technical support through the safety case to enable licensees to comply with Licence Condition 23.

90. To enable continuous power generation between reactor outages, Examination, Inspection, Maintenance and Testing (EIMT) may be required, e.g. scheduled EIMT or equipment failures. Limits and conditions specify minimum acceptable levels of plant and equipment availability required (or can be taken out of service) to support the safety case. Limits and conditions therefore have a key role in determining EIMT requirements.

4.2.2 Assessment of Hitachi-GE’s Limits and Conditions

91. ONR’s expectation is that Hitachi-GE will consider all mechanical engineering items with a safety classification, to ensure that limits and conditions are adequately recognised and documented. Hitachi-GE has used a number of Basis of Safety Cases (BoSCs) and topic reports to present this evidence to ONR. I have assessed a sample of these documents in the various sections of this report and I am satisfied sufficient information is presented to enable a licensee to identify appropriate limits and conditions during detailed design.

4.2.3 Assessment of Hitachi-GE’s EIMT Principles

92. ONR SAP SC.6 states ‘*The safety case for a facility should identify the important aspects of operation and management required for maintaining safety*’. Furthermore, ONR SAP EMT.1 states ‘*Safety requirements for in-service testing, inspection and other maintenance procedures and frequencies should be identified in the safety case*.’

93. These aspects of the safety case are important to inform the licensee who is required (under Licence Condition 28, EIMT) to generate a plant maintenance schedule to define the safety important maintenance activities, with appropriate periodicities and instructions.

94. There is an expectation in the UK that a PCSR should identify EIMT requirements for SSCs important to safety. Hitachi-GE should therefore generate a sufficient depth of information during GDA to facilitate this process, and should clearly identify the links to the safety case assumptions in this respect. However, during my assessment sample of draft Rev C of the PCSR I did not find sufficient evidence that Hitachi-GE had identified these requirements. ONR challenged Hitachi-GE on this issue and Hitachi-GE has revised the text in the current revision of the PCSR. I am now satisfied that the latest revision of the PCSR addresses my concerns in this respect.

95. I recognise that further detailed considerations for EIMT will occur during detailed design. My expectation is that responsibility for making and implementing adequate EIMT arrangements in respect of licence conditions will rest with the licensee to ensure EIMT is informed by the limits and conditions identified in the safety case. I have identified an assumption that should be considered by a future licensee (AS-ABWR-ME05).

4.2.4 Assessment of Hitachi-GE's Isolations and Configurations During EIMT

96. During step 3, ONR challenged Hitachi-GE on its strategy for relying on single isolation of plant during EIMT. ONR considered this a shortfall against regulatory expectations as it does not reduce the risks SFAIRP and does not reflect RGP. ONR raised RO–ABWR-018 seeking evidence that Hitachi-GE could satisfy RGP for plant isolations.

97. In response to RO-ABWR-0018, Hitachi-GE undertook a multidisciplinary review of its policy on EIMT isolations and configurations based on guidance for safe isolation of plant, prepared by the Health and Safety Executive (HSE) [32]. The review resulted in Hitachi-GE preparing a revised strategy with a process to manage documents, share information and consider applicable legislation.

98. Hitachi-GE shared its revised isolations strategy [33] at a technical workshop with ONR [34]. ONR made a number of observations to indicate that:

- it was content with progress
- application of the strategy needs to be across the whole plant
- the licensee should engage with Hitachi-GE to develop this strategy in future.

99. Hitachi-GE is continuing to review its policy on EIMT isolations and configurations of SSC's to meet UK RGP. The actions from this review will be summarised by Hitachi-GE in a final report submitted to ONR after conclusion of the GDA process. During step 4 Hitachi-GE conducted detailed ALARP assessments of all SSC's that rely on valve isolation for the off gas system, residual heat removal system and the standby liquid control system. ONR considered that this sample assessment provided evidence that Hitachi-GE's review was adequate.

100. The results of Hitachi-GE's GDA findings should enable the licensee to implement the safety case into the plant life cycle. Therefore, ONR was satisfied that Hitachi-GE's response was adequate and ONR subsequently closed ABWR- RO- 018.

101. Hitachi–GE's ALARP assessments of SSC's that rely on 'single valve isolation' is summarised in the RO-018 closure summary report [35]. ONR was content with Hitachi-GE's findings for these particular SSC's. For this group of SSC's, Hitachi-GE concluded that:

- the hazard levels of the substances was lower,
- the additional burden of installing two valves (in terms of additional maintenance work and waste increase) was higher,
- there was a potential risk of loss of the safety function by having additional isolation,
- process fluids are removed,
- EIMT frequency is low

102. Isolation and configuration during EIMT on fine motion control rod drives remains a residual risk because Hitachi-GE could not fully implement its isolating and configuration strategy. ONR raised a separate RQ on this aspect that Hitachi-GE dealt with this separately from RO-ABWR-0018. I have assessed this RQ later in this my report.

103. Hitachi–GE made a separate ALARP argument for removal of reactor internal pumps. Hitachi-GE's solution [36] is to modify the J-ABWR design such that the upper plug includes an additional O-ring seal or lip seal. Hitachi-GE has confirmed that in principle these options can provide an effective sealing arrangement although further refinement will be required in the detailed design phase. I note that, if it is not reasonably practicable for the licensee to implement this modification, the J-ABWR design may represent an adequate option even though it may not fully meet RGP. I am content with this approach.

4.2.5 Conclusions

104. I am satisfied that Hitachi-GE has considered limits and conditions of operation and EIMT in its safety case submission.

105. I have identified an assumption that should be considered by a future licensee to ensure that detailed design adequately considers EIMT requirements.

4.2.6 Regulatory Assumptions

106. **AS-ABWR-ME05 –** ONR acknowledges that during GDA, Hitachi-GE has not sufficiently developed its design to enable the licensee to identify the appropriate EIMT requirements in line with UK expectations. ONR assumes that the licensee shall ensure that it considers EIMT requirements for all mechanical engineering items that attract a safety classification. ONR also assumes that the licensee shall establish whether these requirements are directly driven by the safety case, are based on manufacturer recommendations or are based on plant operating experience (or appropriate combinations).

4.3 Assessment of Safety Function Categorisation and Equipment Classification

4.3.1 Introduction

107. Fundamental safety functions are high-level objectives that must be delivered during normal operation and fault conditions. These safety functions should be categorised based on analysis of the design. Hitachi-GE has adopted a scheme similar to that suggested in ONR SAPs, adopting the following categories:

- Category A – any function that plays a principal role in ensuring nuclear safety.
- Category B – any function that makes a significant contribution to nuclear safety.
- Category C – any other safety function contributing to nuclear safety.

108. The structures, systems and components (SSCs) that deliver the safety functions should be classified, according to their importance. Hitachi-GE has adopted a classification scheme similar to that suggested in ONR SAPS, adopting the following:

- Class 1 – any SSC that forms a principal means of fulfilling a Category A safety function.

- Class 2 – any SSC that makes a significant contribution to fulfilling a Category A safety function, or forms a principal means of ensuring a Category B safety function.

- Class 3 – any other SSC contributing to a categorised safety function.

109. Categorisation and classification influence the process adopted during design, procurement, installation, commissioning as well as EIMT requirements. Hence, categorisation of safety functions and classification of SSCs to deliver them are important considerations from a mechanical engineering perspective.

110. From a mechanical engineering perspective, I consider the following SAPs to be relevant to this topic for UK ABWR:

    - SAP ECS.1 states that the safety functions to be delivered within the facility, both during normal operation and in the event of a fault or accident, should be categorised based on their significance with regard to safety.
    - SAP ECS.2 states that the structures, systems and components that have to deliver safety functions should be identified and classified on the basis of those functions and their significance with regards to safety.
    - ONR SAPs para 540 considers that there are three key fundamental safety functions for a nuclear reactor:
        a) Control of reactivity (including preventing re-criticality following an event);
        b) Removal of heat from the core;
        c) Radioactive material confinement.

4.3.2 Overview of Hitachi-GE's application of categorisation and classification

111. Hitachi-GE has generated its own structure with three classes of component as described in the ONR SAPs. Hitachi-GE's structure links failure frequency (ff) for continuously operating systems and probability of failure on demand (pfd) for demand-based systems. ONR fault studies and probabilistic safety specialists assessed this structure, in terms of the links to failure frequencies and probability of failure on demand, and reported their findings accordingly.

112. ONR expects that the licensee will continue to confirm, update and extend safety functions as required in accordance with fault studies analysis during detailed design.

4.3.3 Strategy for Assessment of Safety Function Categorisation and Equipment Classification

113. The ONR TAG on categorisation and classification [37] recognises that in general, there are no UK nuclear specific codes and standards to define the requirements for the categorisation and classification of mechanical engineering aspects. Instead, Hitachi-GE must ensure that it has robust quality management arrangements to satisfy the required SSC reliability. Hence, my assessment strategy for step 4 has been to seek evidence that Hitachi-GE quality management arrangements are robust from a mechanical engineering perspective.

114. This particular topic is crosscutting affecting a range of ONR assessment disciplines. Throughout the GDA process, ONR has discussed the subject of safety function categorisation and equipment classification with Hitachi-GE via Level 4 mechanical engineering technical meetings. The evidence obtained from these discussions has influenced the findings reported below from the mechanical engineering assessment.

115. Throughout GDA, Hitachi-GE's SSC classification has influenced my choice of which SSC's to select for my assessment sample. Generally, I have assessed SSC's with higher safety classification, although I have also included some lower class systems to offer a balanced assessment sample and assure myself that Hitachi-GE has appropriately classified SSCs.

4.3.4 Assessment of Hitachi-GE's application of safety function

116. Chapter 5 of the PCSR [27] systematically develops the safety functions for the UK ABWR based on two major safety category groups. These groups help differentiate between SSCs required for normal operation (e.g. the reactor pressure vessel) and those that prevent an escalation or provide immediate control in abnormal conditions.

117. Chapter 5 of Hitachi-GE's PCSR [27] identifies five high level fundamental safety functions which I consider align with the three high level safety functions stated in ONR SAPs para 540 described above. I sampled systems during my mechanical engineering assessment that address each of the five fundamental safety functions as shown in Table 1.

118. Table 1 – Hitachi-GE's fundamental safety functions

Hitachi-GE's High Level Fundamental Safety Function	Examples of systems sampled by mechanical engineering assessment
Control of reactivity	Control rod drives
Fuel cooling	Reactor coolant system and associated plant. Residual Heat Removal (RHR) system
Long term heat removal	Residual Heat Removal (RHR) system, Auxiliary Cooling Water (ACW) systems, Emergency Core Cooling System (ECCS).
Confinement of radioactive materials	Gloveboxes Shielding Mechanical handling systems
Others (largely for support functions required to enable one or more of the above safety functions)	Fuel handling systems

4.3.5 Assessment of Hitachi-GE's application of categorisation and classification

119. Chapter 5 of the PCSR [27] describes the purpose and methodology for categorisation and classification that Hitachi-GE has used in the UK ABWR safety case. Hitachi-GE analyses results to categorise SSCs according to their importance in the overall safety of the plant. The classification reflects the importance of each SSC to the safety of the plant and links engineering, such as codes and standards for design, manufacture, inspection, maintenance, and testing directly to the safety case. Finally, deterministic and probabilistic safety assessments demonstrate that the resulting design meets all risk targets and reduces risk SFAIRP. I am satisfied that this aligns with the methodology set out in the SAP's. I am satisfied with Hitachi-GE's approach which, if properly applied should ensure that SSC's meet the desired safety function.

4.3.6 Assessment of Hitachi-GE's Codes and Standards

120. Hitachi-GE has provided lists of recommended codes and standards in the Codes and Standards section of chapter 5 of the PCSR [27]. The list indicates that Hitachi-GE

generally adopts the American Society of Mechanical Engineers (ASME) nuclear specific codes and standards for SSCs in Classes 1 and 2 where appropriate. Hitachi-GE has also provided examples where it has adopted specific manufacturers' standards for specialised nuclear specific equipment, for example control rods. For situations where ASME is not appropriate, Hitachi-GE intends to design in accordance with International (ISO), British European (BS EN) standards or other equivalent codes and standards suitably enhanced for nuclear application. For, SSCs in Class 3, appropriate non-nuclear-specific codes and standards may be applied.

121. Codes and standards are not mandatory in the UK but they are part of a range of options that provide a means of satisfying RGP. Licensees may use various sources of RGP to help them demonstrate compliance with UK legislation. Hence, I consider that it is reasonable for Hitachi-GE to propose a range of internationally recognised codes and standards that may be suitably applied and enhanced as appropriate to achieve an adequately safe design solution for a particular SSC.

4.3.7 Assessment of Hitachi-GE's links between categorisation and classifications and Examination, Inspection, Maintenance and Testing (EIMT) Requirements

122. Hitachi-GE has considered EIMT in terms of ensuring that performance of SSCs satisfy the safety requirements intended in the design. Hitachi-GE prescribes EIMT according to the safety class of each SSC. Hitachi-GE bases EIMT activities for class 1 SSC's, performing a category A safety function, or class 2 performing a category A function, on appropriate codes and standards. However, Hitachi-GE recognises that EIMT may be required to go beyond code compliance (particularly for very high integrity applications). Hitachi-GE will therefore identify details of EIMT in the design specification, quality plan and/or inspection and test plan for each component. I am satisfied with Hitachi-GE's application of EIMT with respect to its categorisation and classification process.

4.3.8 Assessment of Hitachi-GE's Equipment Qualification Plan

123. The ONR TAG on categorisation and classification [37] indicates the importance of demonstrating how the safety classification of an SSC influences the whole project life cycle, including for example:

- design approach;
- concept qualification;
- level of auditable design substantiation;
- applied codes and standards;
- material selection;
- procurement phase, detailed design, fabrication, inspections and factory acceptance tests;
- site construction and commissioning phase;
- operational phase in-service EIMT; and
- decommissioning.

124. In terms of mechanical engineering, consideration of the above life cycle phases is essential to demonstrate that a particular SSC can continue to meet its desired safety classification. Throughout my step 4 mechanical engineering assessment, I have assessed Hitachi-GE's BoSC's and topic reports seeking evidence that the above life cycle criteria have been considered by Hitachi-GE. In summary, I am satisfied that Hitachi-GE's process should adequately address these life cycle criteria.

125. An important aspect of ensuring that the safety classification of SSC's is met is Hitachi-GE's equipment qualification plan. The purpose of the qualification plan is to inform the licensee so that it is able to generate evidence that equipment will operate on demand and meet system performance requirements under specified service and fault conditions. Hitachi-GE's process requires satisfactory completion of examination, inspection and testing before receiving components and placing them into service.

126. The first step in establishing the equipment qualification plan is to identify Limits and Conditions of Operation (LCO). During my assessment, I considered LCOs for each system I sampled. I have reported these findings for each system assessed, later in this report. In summary, I am satisfied that the various sections of the Hitachi-GE PCSR offer suitable links to the detail necessary to establish LCO's, for example the BoSC's and other documentation making up the safety case.

127. ONR has sampled Hitachi-GE qualification arrangements throughout the GDA process. I am satisfied that these offer a reasonable approach that should enable the licensee to follow a logical process that will demonstrate the reliability of mechanical components. I am satisfied that the process aligns with ONRs guidance in the category and classification TAG [37].

128. Hitachi-GE's equipment qualification is planned and carried out considering the 60-year design life of the UK ABWR. If the design life or the expiration date is set, requirements for maintenance, surveillance, and periodic test can also specified in order to maintain the integrity of equipment. I am satisfied that this strategy should ensure that the design will be qualified for through life reliability.

129. Hitachi-GE provided a list of systems that might be required during the decommissioning phase. However, ONR considered that these systems were not clearly identifiable in early drafts of the PCSR. ONR was concerned that important design requirements, to enable these systems to be available for decommissioning, might be overlooked. Hitachi-GE has subsequently revised the PCSR and I am satisfied this should ensure that decommissioning requirements are appropriately considered during detailed design.

4.3.9 Conclusions

130. Based on my assessment during step 4, I consider Hitachi-GE has now progressed this topic, and responded positively to the ONR guidance provided. It has now generated an approach to safety function categorisation and equipment classification in line with ONR SAPs. I am now satisfied with Hitachi-GE's safety function categorisation and equipment classification methodology, and its application, for the UK ABWR, from a mechanical engineering GDA perspective against SAPs ECS.1, ECS.2 and ERC.1.

4.3.10 Regulatory Findings or Shortfalls

131. None

4.4 Legislation, Codes, and Standards

4.4.1 Introduction

132. Compliance with legislation, codes and standards are a means of ensuring that the UK ABWR design meets relevant good practice (RGP) for safety. ONR's particular areas of interest are:

- Application of UK relevant good practice
- Identification of applicable regulations, codes and standards

- Gap analysis between Japanese and UK regulations codes and standards

4.4.2 Overview of Hitachi-GE Safety Case

133. Hitachi-GE's codes and standards report [38] provides a high-level comparison between Japanese and UK regulations for nuclear specific applications.

134. The topic report on Acts, Regulations, codes and standards [39] identifies relevant UK legislation and standards for the design, manufacturing, construction, inspection, installation, commissioning, quality assurance and maintenance of Structures, Systems and components (SSCs) of the UK ABWR. Hitachi-GE identifies the codes and standards for use with class 1,2 & 3 SSCs, clearly indicating those that are nuclear specific.

135. The topic report on safety requirements for mechanical SSCs [40] identifies the codes and standards that will be used.

136. I am satisfied that the approach demonstrates that Hitachi-GE has adequately considered the use of relevant UK legislation, codes and standards which compares to the assessment guidance in ONR SAPs ECS.3 and ECS.4.

4.4.3 Strategy for Assessment of Legislation, Codes, and Standards

137. Three Hitachi-GE documents that outline the use of legislation, codes and standards [38] [39] [40] have been considered for the purposes of this assessment. Other references used to inform my judgement are:

- NS-TAST-GD-057 Design Safety assurance [14] that states establishing appropriate technical standards to underpin the design process is an essential early activity, which clearly has significant safety implications.

- ONR SAP ECS.3, Codes and standards states SSCs that are important to safety should take in to account the appropriate standards. I consider that relevant codes and standards have been identified with identification of the appropriate codes and standards embedded within basis of safety cases documents.

- ONR SAP ECS.4 Absence of established codes and standards states: where there are no appropriate established codes and standards, an approach derived from existing codes for similar equipment, in applications of similar safety significance should be applied. I consider appropriate action has been taken to address the absence of established codes and standards in section 4.3.9 of the Topic Report on Safety Requirements for mechanical SSCs.

4.4.4 Assessment of Hitachi-GE's Legislation, Codes, and Standards Process

138. Hitachi-GE has provided sufficient evidence to demonstrate an adequate codes and standards compliance gap analysis has been undertaken. Hitachi-GE has compared the current standards used on the J ABWR with those applicable to the UK ABWR. This has provided me with confidence that Hitachi-GE is aware of the applicable UK codes and standards.

139. Hitachi-GE has continued to develop its understanding of UK legislation, codes and standards with the submission of a topic report [39]. The information provided in this submission further underpins Hitachi-GE's understanding of UK requirements and aligns to ONR SAPs ECS.3 & ECS.4. Hitachi-GE also indicated that it has applied approved codes of practice supporting UK regulations, which provides me with further confidence that UK Relevant Good Practice (RGP) is being followed. Hitachi-GE has assigned nuclear specific codes and standards to class 1 and 2 components, class 3 components are to adhere to established industry standards should a nuclear

equivalent not exist. I consider this meets the guidance set out in ONR TAG on classification and categorisation [37].

140. The topic report on safety requirements for mechanical SSCs [40] substantiates the use of UK legislation, codes and standards further. Hitachi-GE identifies, for different combinations of classifications and categories the level of regulations codes and standards to be applied. I consider the processes outlined to meet the guidance set out in the ONR TAG on classification and categorisation [37].

4.4.5 Conclusions

141. I am satisfied that my assessment strategy provides me with an adequate sample to identify Hitachi-GE's means of satisfying relevant UK legislation and how it intends to apply relevant codes and standards.

142. I consider that Hitachi-GE has adopted a suitable design process that should enable it to satisfy the requirements of relevant legislation and RGP.

4.4.6 Regulatory Findings or Shortfalls

143. None

4.5 Lifting and Handling

4.5.1 Introduction

144. Lifting nuclear materials or lifting loads over nuclear safety significant equipment are necessary activities within any nuclear power plant. To consider the potential risks associated with these lifts for UK ABWR, I assessed the Reactor Building Overhead Crane (RBC) and Fuel Handling Machine (FHM), which between them perform the majority of such lifts. The majority of lifts occur in the area referred to as the 'Operating Deck' which is the normal operational access level within the reactor building surrounding the Spent Fuel Storage Pool, reactor well and the dryer/separator storage pool.

145. The particular lifting activities sampled were as follows:

- Dryer and separator removal from the reactor
- Lifting associated with opening and closing the reactor during outage

146. The above activities present potentially greater risk than those associated with activities associated with new and irradiated fuel. Hence, I have considered fuel-handling activities later in this step 4 report.

4.5.2 Overview of Hitachi-GE Safety Case

147. Hitachi-GE's Basis of Safety Case (BoSC) details design requirements for what it terms 'Nuclear Special Cranes'. In terms of mechanical engineering, the key hazards identified are:

- Dropped loads onto nuclear safety significant Structures and Systems Components (SSCs)
- Radiological hazards caused by over raising irradiated loads from the reactor or Spent Fuel Storage Pool resulting in inadequate water cover to reduce radiation to a safe level.
- Fuel damage caused by dropping a fuel assembly, or by heavy structure/crane collapse onto a fuel assembly.

- Collision between two nuclear special cranes, a crane and civil structure or a crane and a SSC.
- Conventional safety considerations

148. Chapter 19 of Hitachi-GE's PCSR [41] summarises the case presented in the BoSC. Topic reports present the results of detailed analysis including Failure Modes and Effects Analysis (FMEA) studies, design assessments and identified modifications to J-ABWR.

149. Hitachi-GE produced hazard schedules to collate data from its hazard studies and present the likelihood and consequence associated with each fault. These schedules identify protection system requirements (i.e. with their appropriate category and classification), engineering systems provided on the crane and the operator actions required. In support of these, Hitachi-GE has generated a number of safety measures diagrams that aim to demonstrate a viable means of delivering adequate safety and protection.

150. I am satisfied that Hitachi-GE's approach demonstrates that it has adequately considered the likely failure mechanisms, consequences of failure and design mitigation in a manner that aligns with ONR SAP EKP.4 which states that the safety function(s) to be delivered within the facility should be identified by a structured analysis. I am also satisfied that Hitachi-GE has considered conventional safety hazards assessed against conventional health and safety legislation.

4.5.3 Strategy for Assessment of Lifting and Handling

151. Two key documents I sampled were Hitachi-GE's Basis of Safety Case (BoSC) on fuel handling systems and overhead crane systems [21] and a topic report on mechanical handling equipment [42]. To inform my judgment of the overall lifting systems, I also assessed a topic report on lifting attachments [22], a topic report on design philosophy for lifting beams [43] , and a list of applicable equipment qualification [44].

152. Earlier versions of the BoSC where assessed by ONR's Technical Support Contractor (TSC) against the ONR SAP's and RGP. The TSC's findings identified shortfalls in the evidence presented by Hitachi-GE in the earlier versions of the BoSC. ONR presented these shortfalls in a report issued via RQ-ABWR-0620 [45]. This RQ required Hitachi-GE to:

    i. familiarise itself with the report findings and observations,
    ii. confirm the report factual accuracy;
    iii. prepare to discuss the findings, observations and expectations in detail as part of the planned mechanical engineering technical workshops; and
    iv. in advance of the planned technical workshops, develop and advise its strategy to address the findings, observations and expectations.

153. Throughout step 4, ONR has engaged with Hitachi-GE through various level 4 engagements. During these engagements, I have challenged Hitachi-GE on the shortfalls identified by the TSC and this has resulted in revised submissions of the BoSC. I am now satisfied that Hitachi-GE's revised BoSC provides sufficient evidence to address the TSC's initial observations and findings.

154. ONR has undertaken numerous interactions with Hitachi-GE relating to nuclear special cranes at all stages of GDA via Level 4 contacts, Regulatory Queries (RQs) and Regulatory Observations (RO). The significant interactions are summarised in my assessment below.

155. Toward the end of engagements with Hitachi-GE on this topic, I collated a list of outstanding issues related to nuclear special cranes [46]. Hitachi-GE provided

satisfactory evidence to close these issues and I have summarised its responses in the text below.

4.5.4 Equipment Description

156. **Reactor Building Overhead Crane (RBC) –** This is an electric overhead travelling crane with a main hoist, an additional hoist and an auxiliary hoist. The RBC operating envelope covers a significant proportion of the operating deck and it operates in proximity to several systems, structures and components (SSCs). The crane presents a risk to these SSCs in the event of a dropped load or collision fault.

157. The RBC's primary function is for outage preparation (opening and closing the reactor), outage operations, handling of new fuel as it enters the reactor, spent fuel cask handling operations and removal of solid wastes.

158. Hitachi-GE has designated the RBC as a high integrity nuclear crane (classification 1) with a dual load path. Operator control is from the crane cab although Hitachi-GE has indicated that it will consider alternative control positions during detail design.

159. **Fuel Handling Machine (FHM) –** This is a gantry type crane operating exclusively within the storage pools system. The FHM has a dual load path, main hoist consisting of a telescopic mast suspended from the FHM trolley. The mast is fitted with an air-powered grapple for attaching to fuel assemblies and certain in core components. The FHM is also equipped with a reactor internal pump inspection hoist that can accept various tools and attachments.

160. The FHM travels on floor rails and is operated either from a control room or on the fuel handling machine platform. Hitachi-GE has indicated that it will consider the fuel handling machine control options during detailed design.

161. The FHM's primary function is to transfer new and irradiated fuel assemblies using its main hoist. The fuel handling machine fuel handling activities are:

- New and spent fuel transfers to and from the reactor core
- Spent fuel transfers to the Spent Fuel Storage Pool racks
- New fuel transfers from the fuel preparation machine to the spent fuel storage rack.
- Irradiated fuel transfers from the spent fuel storage rack to the fuel preparation machine for inspection then return to the spent fuel storage rack.
- Spent fuel transfers from the spent fuel storage pool racks to a cask located in the cask pit.

4.5.5 Assessment of Hitachi-GE's Design Process for lifting and handling equipment

162. During step 2 GDA ONR raised cross cutting regulatory observation RO-ABWR-0016 – Mechanical Engineering Design Process Arrangement. ONR considered that Hitachi-GE had not provided adequate assurance of a robust design process arrangement in line with UK relevant good practice and its process was not in a format auditable by regulators. In responding to this observation, Hitachi-GE identified a gap with its ALARP philosophy and therefore revised its design process. I sampled Hitachi-GE's revised design manual for safety related mechanical lifting devices [19] and its design manual for wet lifting beams [47]. This provided sufficient evidence that Hitachi-GE had properly implemented its revised process and resulted in me closing RO–ABWR-016 [16]. Furthermore, I consider that Hitachi-GE's design philosophy satisfies ONR

SAP EDR.1 which is aimed at ensuring that SSCs are designed to be inherently safe using a formal analysis.

163. Hitachi-GE did not initially produce a lifting schedule as part of its design documentation which ONR identified as a design shortfall against UK RGP. The revised design manuals now include adequate lifting schedules that have benefited from input by a UK consultant.

164. Hitachi-GE now uses its lifting schedule as an input to consider whether each operation is necessary or whether it can reasonably practicably eliminate it. I consider that this is a key step in Hitachi-GE's risk assessment methodology that is compliant with UK RGP. For example, Hitachi-GE has eliminated a significant lifting risk by substituting a complex lifting arrangement with a pneumatic puller for removing the internals from main steam isolation valves (assessed later in this report).

165. In accordance with ONR SAP EQU.1, design qualification procedures should be applied to confirm that SSCs perform their allocated safety function(s) in all normal operational, fault and accident conditions. I consider that Hitachi-GE's design approach ensures that it has designed nuclear special cranes to withstand the bounding service conditions, operational conditions and environmental conditions. Hitachi-GE has provided a list of applicable equipment for qualification [44].

166. ONR SAPs EAD.1 to EAD.5 describes the expectations for ageing and degradation to be evaluated and defined at the design stage as well as reviewing obsolescence of SSCs. The BoSC does not specifically state the life expectancy of the nuclear special cranes although the EIMT section implies a 60-year life inspection regime. Hitachi-GE has clarified that it has designed nuclear special cranes with a 60-year life justified through its PCSR.

4.5.6 Assessment of Hitachi-GE's Safety Function and Safety Property Claims for lifting and handling equipment

167. In accordance with ONR SAP EKP.4 I am satisfied that Hitachi-GE has identified the safety function(s) to be delivered using a structured analysis. The key safety claims made by Hitachi-GE address the following hazards:

- Fuel damage due to a dropped fuel assembly, heavy structure or crane collapse
- Radiation exposure due to failure to ensure that irradiated loads are suitably submerged under water during handling
- Collision between nuclear special cranes or other structures

168. Hitachi-GE has assigned nuclear special cranes with safety categorisations that satisfy ONR SAP ECS.2, which requires SSCs to be classified on the basis of their safety function. Hitachi-GE has also ensured that failure of a lower class SSC will not propagate to an SSC of a higher safety class. Hitachi-GE has indicated that a probability of failure on demand of 10^{-4} for nuclear special cranes can be achieved. I consider this is reasonable and in line with the ONR TAG on categorisation and classification [37] which indicates that for Class 1 SSC's a probability of failure on demand between 10^{-3} and 10^{-5} is achievable.

169. Hitachi-GE's BoSC indicates that it has supported its claims by comparison with similar equipment/systems within the nuclear industry. Detail design of the RBC after GDA will be necessary to substantiate these claims but I consider that this approach should enable the licensee to undertake detailed design, manufacture and procurement to an appropriate classification. Hitachi-GE has provided a list of components subject to diversity and/or independence (independence achieved through segregation or

redundancy) requirements in reference [21] which further assists the licensee to ensure safety.

170. I sought evidence that Hitachi-GE's proposals for the RBC satisfy UK legislation including Lifting Operations and Lifting Equipment Regulations (LOLER). Hitachi-GE has reviewed how the requirements of UK legislation will affect the UK ABWR RBC. I am satisfied that this comparison was adequate and should enable Hitachi-GE to specify a crane that will comply with the requirements of LOLER.

171. I challenged Hitachi-GE on its analysis of additional crane safety measures such as mechanical, hoist motor followers. Hitachi-GE presented evidence that it had adequately assessed these options and dismissed them as not reasonably practicable due to the high safe working load of the crane capacity and the restrictions imposed by a dual load path system. I am satisfied that Hitachi-GE has adequately considered such devices and I accept Hitachi-GE's decision that such devices may not be reasonably practicable in this application.

172. Hitachi-GE applies safety property claims, which are principles, applied when designing SSCs in accordance with its Nuclear Safety and Environmental Design Principles (NSEDP)s. Hitachi-GE has provided evidence that its NSEDPs specify requirements that are similar to ONR SAP's. I am therefore satisfied that Hitachi-GE has a process that should address UK expectation with respect to lifting and handling.

4.5.7 Comparison of UK ABWR with J-ABWR reference design

173. Hitachi-GE provided sufficient evidence that it had undertaken an adequate gap analysis against the reference J-ABWR design considering UK legislation, UK RGP and OPEX (as observed in RO-ABWR-0045). A UK consultant was utilised for part of this work and I consider that this adds further confidence that Hitachi-GE has adequately considered UK RGP. The BoSC presents a number of modifications to mechanical handling operations for UK ABWR because of this gap analysis. The key modifications can be summarised as:

- RBC control modes and zones
- Mechanical load path protection systems
- Use of lifting slings
- Revised method for dryer and separator handling

4.5.8 Assessment of Hitachi-GE's Dryer Separator handling

174. It is reasonably foreseeable that the dryer or separator could drop onto the open reactor whilst the crane is lifting them. ONR raised RO-ABWR-0049 covering dropped load consequences. In response, Hitachi-GE satisfied ONR that it had adequately considered dropped load consequence and that it had a suitable audit trail, via its lifting schedule. ONR internal hazards specialists raised RQ-ABWR-994 to clarify affected SSCs if the separator drops. Hitachi-GE provided a satisfactory response [47] indicating dropping the separator would not result in a significant safety concern and no further queries where raised.

175. The available depth of reactor water provides shielding above the steam dryer and separator when lifting them from the reactor. During GDA step 2 [48], ONR indicated that relying on operator judgement to control the RBC height relative to reactor water level did not represent UK RGP or an ALARP design. ONR raised cross cutting RO-ABWR-0050 to capture this major concern and Hitachi-GE provided a summary response [49] which proposed a change to automate water level detection and control the lift height of the Reactor Building Overhead Crane hook. Hitachi-GE's response provided evidence of a robust optioneering process that satisfies ONR SAP safety case guidance SC.5 and engineering key principles set out in SAPs EKP.1 to EKP.5.

However, Hitachi-GE indicated [48] that its solution involved suspending the separator over the open reactor for several hours during a staged lift whilst the water level was slowly increased. ONR considered that this method did not meet RGP as this increased the time at risk for dropping a suspended load over the open RPV. During a technical L4 engagement [50] ONR indicated that it considered this to be a significant concern requiring closure during the GDA process. Hitachi-GE has now committed to adopt a design where the hook is wetted thereby negating the need to suspend the load whilst the pond level is slowly increased. Hitachi-GE has included the revised design in the BoSC for fuel handling systems and overhead crane systems [21].

176. ONR sought further evidence that wetting the hook reduces risks ALARP and raised RQ-ABWR-0259 (Design philosophy – pool lifting beam) and RQ-ABWR-047 (Mechanical Engineering-wet lifting beams – materials of construction), seeking further evidence. To satisfy RO-ABWR-0047, Hitachi-GE submitted its design manual for wet lifting beams [51]. I noted that the minimum material properties state that stainless steel shall be used for class 1, wet lifting attachments and that nylon slings shall not be used for class 1 lifting attachments. I am satisfied that these procedures now reflect RGP and the materials used for wetted lifting equipment are suitable. I am also satisfied that the specification of stainless steel reduces the potential for corrosion and degradation of wet lifting attachments. Hitachi-GE assessed the risk reduction achieved by handling the dryer submerged and judged it to outweigh any disadvantages of submerging the RBC hook. Hitachi-GE supported its claim with OPEX from the United States where wetting of the hook is now standard practice supported by the US Boiling Water Reactor (BWR) owners groups.

177. Supported by ONR radiological protection specialists, I queried how Hitachi-GE intends to decontaminate wetted components. Hitachi-GE's response was to provide a report [52] that identifies and describes locations for handling and decontaminating wet lifting equipment. From a mechanical engineering perspective, I did not identify any issues related to lifting and handling items into decontamination areas.

4.5.9 Assessment of Hitachi-GE's Operating Zones and Modes of Control

178. ONR's nuclear lifting operations TAG [53] indicates that UK legislation aims to control risks from human error by setting out a hierarchy for control measures. Within this hierarchy, engineered safeguards are at the top and employee instructions are at the bottom. I am satisfied that Hitachi-GE has not foreclosed options for the provision of engineered safeguards that prevent operators entering potential danger zones during lifting operations. Hitachi-GE has indicated [52] that the licensee will be responsible for providing appropriate arrangements to ensure the safety of all lifting operations. Hitachi-GE indicated that its concept design also considered the number and location of personnel required to support lifting operations in the development of the nuclear lifting equipment that formulates the BoSC. In particular, I examined the method for removal/installation of the RPV head in response to RQ-ABWR-1312. ONR human and organisational factors specialists also assessed this document. I considered the methods, tools and equipment, from a mechanical engineering perspective, to gain evidence that the methods minimised duration that personnel needed to be in that area. Reducing the time in the area can minimise the time at risk from lifting operations as well as reducing radiological doses.

179. ONR queried how Hitachi-GE would prevent operators from working under suspended loads during lifting operations. Hitachi-GE presented a diagram indicating worker locations [54] that was provided in response to RQ-ABWR-0611 that seeks information on person around the charge hall. I am satisfied that this diagram demonstrates a feasible solution where workers do not work under suspended loads. The detail of this solution is for a licensee to implement through its operating procedures to ensure safe lifting operations.

180. A collision between the Fuel Handling Machine (FHM) and the Reactor Building Overhead Crane (RBC) with any other civil structure or mechanical SSC could foreseeably result in a dropped load, or collapse of the crane above the Spent Fuel Storage Pond. At the sixth mechanical engineering workshop, Hitachi-GE explained that it has designed the FHM to be automated so that it follows a designated route that avoids obstructions to place loads automatically in a dedicated laydown area. I consider that this enhances safety by reducing the likelihood of human error, assuming that the associated control system is suitably reliable. In addition, the licensee operating procedures should prohibit the use of more than one nuclear special crane at the same time. There is also an emergency stop system available, enabling the workers to safely stop the lifting operations if required.

181. Hitachi-GE has designed zoning systems for the nuclear special cranes that use hard-wired interlocks as ultimate limits to provide protection. A class 3 control system controls movement within the protection zones. Captive keys initiate hard-wired zone controls that enable the use of particular lifting attachments and prevent unsafe use.

182. I consider that Hitachi-GE's approach is adequate to ensure that control and protection systems minimise the risks of collision. Furthermore, I consider that Hitachi-GE has provided adequate detail to enable the licensee to develop detailed procedures and lifting plans to ensure that UK RGP and legislative requirements are satisfied.

4.5.10 Assessment of Hitachi-GE's Recovery

183. Hitachi-GE's design gives priority to setting down loads safely following a trip, which I consider appropriate. Furthermore, the design includes recovery provision such as towing points on the crane enabling the use of a winch to recover a nuclear special crane. Brakes and lifting attachments have manual release mechanisms compatible with long reach tools where necessary.

4.5.11 Assessment of Hitachi-GE's Decommissioning for lifting and handling equipment

184. ONR SAP SC.3 requires that; for each lifecycle stage, control of the hazard should be demonstrated by a valid safety case that takes into account the implications from previous stages and for future stages. Hitachi-GE has indicated in its BoSC on fuel handling systems and overhead crane systems [21] that the FHM and RBC are required to support decommissioning operations. For example, retrieving spent fuel and cutting up the reactor pressure vessel and reactor internals. Hitachi-GE's generic design considers that most of these activities are the same as routine operations. However, Hitachi-GE has indicated that the RBC will be required to perform some lifting operations during decommissioning that are different to routine operations.

185. The lifting and handling BoSC [21] indicates that Hitachi-GE will consider 60 years operational life, decommissioning design life and appropriate utilisations during detail design. Fatigue cycle monitoring will be included and maintenance activities will continue to the end of decommissioning. Furthermore, prior to the commencement of decommissioning activities, the nuclear special cranes will undergo inspection and refurbishment, if required, to ensure they remain fit for purpose. Hitachi-GE predicts that the cranes will not have become irradiated or activated at the end of their life. I am therefore satisfied that the nuclear special cranes can be decommissioned by conventional means even after using them for decommissioning activities.

186. I consider that Hitachi-GE's inclusion of decommissioning in the BoSC [21] is adequate to satisfy lifecycle demonstrations in the safety case as described in ONR SAPs SC.3 and ensure that the licensee considers decommissioning during detailed design. Chapter 19 of the PCSR [41] provides a link to the PCSR chapter on decommissioning (chapter 31) which identifies the requirement for the ongoing use of nuclear special

cranes during decommissioning. I have assumed that in future, detailed design of the crane will take into account the requirements for decommissioning. I have identified an assumption that should be considered by a future licensee (AS-ABWR-ME04).

4.5.12 Assessment of Hitachi-GE's Examination, Inspection, Maintenance and Testing (EIMT) for lifting and handling equipment

187. Hitachi-GE's design of nuclear special cranes includes in-use monitoring with data logging. This system is provided to ensure that nuclear special cranes do not exceed fatigue design limits. Hitachi-GE has not specified surveillance tests at this stage, however, the nuclear special crane operating procedures (to be developed by the licensee during detail design) should define all the system checks and safety related procedures required. I recognise that further detailed considerations for EIMT will occur during detailed design.

188. I consider that Hitachi-GE's BoSC satisfies ONR SAP EMT.1 that states that safety requirements for in-service testing, inspection and other maintenance procedures and frequencies should be identified in the safety case.

189. I am satisfied that a licensee can develop full EIMT provisions during detail design. I consider that Hitachi-GE's approach satisfies ONR SAP EMT.2, which states that SSCs should receive regular and systematic EIMT as defined in the safety case. Hitachi-GE has indicated its intention to specify EIMT in the crane Operation and Maintenance Manuals (OMM) and integrated with the plant maintenance instructions. I consider this is an appropriate approach for GDA. Furthermore, Hitachi-GE has demonstrated that it has considered UK based EIMT experience and it intends to extend this to cover OPEX available from the J-ABWR and US BWR fleets, as well as taking account of UK RGP.

190. Hitachi-GE has indicated the parking positions for the fuel handling machine and Reactor Building Overhead Crane in the lifting and handling topic report [42]. Hitachi-GE proposes that EIMT takes place in these positions. I consider this is a reasonable proposal although Hitachi-GE needs to establish details of access, ancillary lifting equipment requirements and radiological considerations during detailed design.

191. My expectation is that responsibility for making and implementing adequate EIMT arrangements in respect of licence conditions will rest with the licensee. I have identified an assumption that should be considered by a future licensee (AS-ABWR-ME05).

4.5.13 Conclusions

192. I am satisfied that my assessment strategy provides me with an adequate sample to identify the SSCs that are important for safety.

193. I consider that Hitachi-GE has adopted a suitable design process that should enable it to satisfy the requirements of UK legislation and RGP.

194. Hitachi-GE has provided adequate evidence that its case has been prepared in accordance with its safety case development manual and GDA ALARP Methodology. I am satisfied that Hitachi-GE has a process which robustly enables it to consider normal operating and potential fault conditions including internal and external hazards and the conventional safety and human factor influences that could affect safety.

195. The integrity of the SSCs has been demonstrated through to the point at which risks have been reduced ALARP, taking due account of potential ageing (i.e. fatigue) and degradation mechanisms.

196. I consider the likelihood of mechanical failure due to inherent defects within the lifting systems to be low providing; rigorous design and manufacturing quality assurance regimes are applied and EIMT is appropriate throughout plant life. Hitachi-GE's approach should enable the licensee to identify SSCs important to safety. I have captured this requirement with a regulatory assumption.

197. I am satisfied that Hitachi-GE has considered EIMT in its safety case submission. I have identified a regulatory assumption to capture the requirement for detailed design to consider EIMT requirements.

198. I am satisfied that Hitachi-GE has considered decommissioning in their safety case submission. I have identified an assumption that detailed design adequately considered decommissioning requirements.

4.5.14 Regulatory assumptions

199. **AS-ABWR-ME04 -** During GDA, Hitachi-GE identified certain equipment that is required to perform activities associated with decommissioning of the plant at the end of its 60-year life. ONRs expectation is that the licensee will provide suitable plant from the outset to avoid unnecessary modifications to plant in future prior to decommissioning. ONR assumes that the licensee shall identify and confirm the use of equipment for decommissioning shall consider design features to facilitate decommissioning and reduce future dose uptake by workers and where reasonably practicable include any necessary design features in the final design.

200. **AS-ABWR-ME05 -** ONR acknowledges that during GDA, Hitachi-GE has not sufficiently developed its design to enable the licensee to identify the appropriate EIMT requirements in line with UK expectations. ONR assumes that the licensee shall ensure that they consider EIMT requirements for all mechanical engineering items that attract a safety classification. ONR also assumes that the licensee shall establish whether these requirements are directly driven by the safety case, are based on manufacturer recommendations or are based on plant operating experience (or appropriate combinations).

4.6 Fuel Handling

4.6.1 Introduction

201. The UK ABWR fuel route operations can be grouped into four major processes:

- Outage preparation
- Outage operations
- Storage of spent fuel within the spent fuel storage pool
- Export of spent fuel out of the reactor building. (Note that spent fuel handling and storage operations external to the reactor building is not within the scope of the fuel route or this GDA step 4 assessment).

202. The purpose of this fuel handling section is to consider any specific lifting and handling equipment or operations associated with the four processes above. Hence, I have assessed a sample of fuel route handling equipment and operations, as follows:

- The adequacy of the New Fuel Inspection Stand (NFIS)
- The adequacy of the Fuel Preparation Machines (FPM)

- The adequacy of the spent fuel export route, using the Reactor Building Overhead Crane (RBC).

4.6.2 Overview of Hitachi-GE Safety Case

203. Hitachi-GE's fuel route safety case is set out in chapter 19 of the Generic Pre-Construction Safety Report [41]. This describes the major SSCs used from the introduction of new fuel into the Reactor Building to the export of spent fuel. Hitachi-GE identifies three main hazards associated with the handling of fuel in the UK ABWR:

- Direct radiation exposure of workers caused by reduced or loss of shielding
- Criticality accidents
- Radiation exposure of workers or the public caused by loss of containment due to damaged fuel or containers

204. A dropped load or loss of control of lifting operations has the potential to cause significant on-site or off-site dose. Hitachi-GE claims that it has controlled these risks by designing the UK ABWR to UK and international good practice. Hitachi-GE's arguments are set out in the BoSC [21] based on a number of key assumptions as follows:

- A high integrity (classification 1) Reactor Building Overhead Crane should ensures that loads remain under control at all times

- The design of the spent fuel export, hoist well ensures that the risk of a cask snagging or toppling is minimised

- Modifications to enhance the J-ABWR reference design will be included

- A withstand claim is made on the cask used for spent fuel export by virtue of an impact limiter permanently mounted on the hoist well floor.

- Failed canisters will not be exported.

4.6.3 Strategy for Assessment of Fuel Handling

205. I assessed the use of the Reactor Building Overhead Crane to lower spent fuel casks a significant height during spent fuel export activities to establish whether Hitachi-GE's baseline design reduces risks ALARP. During my assessment [55] I considered three potential consequences if the load is not controlled for the entire lift, as follows:

- Release of radioactivity due to failure of the cask

- Conventional safety related incidents due to a falling load

- Reputational damage to both the licensee and the wider industry

206. ONR has undertaken numerous interactions with Hitachi-GE relating to cranes and handling equipment throughout all stages of GDA via Level 4 contacts, Regulatory Queries (RQs) and Regulatory Observations (ROs). Significant interactions are summarised in my assessment below.

4.6.4 Equipment Description

207. **Fuel Preparation Machines (FPM) -** are located on the wall of the Spent Fuel Storage Pool for lifting and lowering irradiated and new fuel assemblies in the pond. There are two identical machines and positioned between these is a jib crane for handling irradiated channel boxes associated with the fuel assemblies.

208. **New Fuel Inspection Stand (NFIS) -** is located on the operating deck of the reactor building adjacent to the Spent Fuel Storage Pool. The New Fuel Inspection Stand provides a facility for handling and a means of access for operators to carry out visual inspection of the fuel assemblies prior to storing them in the Spent Fuel Storage Pool. The new fuel inspection stand incorporates a hoist for raising and lowering the fuel assembly from the new fuel inspection pit to the operating deck during inspections. A jib crane is located on top of the New Fuel Inspection Stand for installing channel boxes to the fuel assemblies.

209. **The UK ABWR spent fuel export route –** is based on an enhanced version of the J-ABWR reference design. As proposed for the J-ABWR, Hitachi-GE is proposing to lower casks, loaded with transfer canisters containing spent fuel, a significant height down a hoist well. This operation will be carried out using the Reactor Building Overhead Crane (RBC). Throughout step 3 and 4 of GDA, ONR challenged Hitachi-GE on the basis that the cask did not have a withstand capability for this height during a dropped load fault scenario. For example, ABWR-RQ-0757 and ABWR-RQ-0862 both relate to spent fuel export. I have detailed my assessment of Hitachi-GE's response to ONR's challenges below.

4.6.5 Assessment of Hitachi-GE's Design Process for fuel handling equipment

210. In the section of this report that assesses lifting and handling, I concluded that Hitachi-GE's crane design process was adequate. The BoSC covers all cranes including the fuel preparation machine (FPM) and the new fuel inspection stand (NFIS). Therefore, I draw the same conclusion, that I am satisfied that Hitachi-GE's design process will be appropriate to ensure that the fuel preparation machine and the new fuel inspection stand satisfy guidance in ONR SAP EDR.1. This SAP says that; *due account should be taken of the need for SSCs to be designed to be inherently safe, or to fail in a safe manner and potential failure modes should be identified using a formal analysis.*

211. Chapter 19 of the PCSR [41] indicates that the new fuel inspection stand is categorised as safety category C (safety function contributing to nuclear safety) and the SSCs that deliver it are designed to meet safety class 3 requirements (component that contributes to a categorised safety function). The FPM bounding function is categorised as safety category A (plays a principal role in ensuring nuclear safety) and the SSCs that deliver it are designed to meet Safety Class 1 requirements (structure that forms a principal means of fulfilling a category A safety function). My assessment of handling and lifting equipment concludes that Hitachi-GE's approach is adequate to ensure that it has designed cranes to withstand the bounding service conditions, operational conditions and environmental conditions. Hence, I am satisfied that Hitachi-GE's procedures should ensure that the FPM and the NFIS would be designed and manufactured to an appropriate class to satisfy bounding operations and fault conditions.

212. ONR sought evidence that Hitachi-GE had considered diverse systems to control the lifting and lowering of casks containing spent fuel during export throughout the lift. Hitachi-GE's optioneering study [56] indicates that Hitachi-GE considered using a load follower to control the cask in the event that the RBC loses control of the load. From a mechanical engineering perspective, I am satisfied that Hitachi-GE's consideration of a diverse safety mechanism was robust and that it presented adequate evidence of its optioneering process. Hitachi-GE's subsequent assessment of this option concluded that the cost of implementing a load follower is grossly disproportionate to the benefit and that it introduces novel technology with associated risks. I am satisfied that Hitachi-GE undertook an appropriate ALARP assessment [57] as part of this design optioneering study to support its conclusions.

4.6.6 Assessment of Hitachi-GE's Safety Function and Safety Property Claims for fuel handling equipment

213. Many of the safety functions and safety property claims made in the BoSC for all nuclear cranes are relevant to the fuel preparation machine and the new fuel inspection stand. These are included in the lifting and handling section of this report under assessment of Hitachi-GE's safety function and safety property claims.

214. Start-up neutron monitors and local power range monitors are relatively long with a highly irradiated section near the top. To reduce radiation to workers, Hitachi-GE has designed special tooling that bends and converts the monitors into 6m long sections. This ensures the irradiated section remains under an adequate depth of shield water during lifting operations. Solutions such as this provide me with confidence that Hitachi-GE has adequately considered reducing risks ALARP. I assessed Hitachi-GE's claims on a sample of lifting attachments. For each case I am satisfied that Hitachi-GE has considered how to minimise radiation to workers.

215. In accordance with ONR SAP EKP.4, I am satisfied that Hitachi-GE has identified the safety function(s) to be delivered using a structured analysis. The high level safety claims made by Hitachi-GE, in terms of mechanical engineering for the Fuel Preparation Machine and New fuel inspection stand, are as follows:

- The Fuel Preparation Machine is designed to withstand normal operational and fault loadings
- The Fuel Preparation Machine includes failsafe protection systems to prevent fault conditions
- Impact limiter in the Fuel Preparation Machine fuel cart prevents damage to irradiated fuel in the event of a dropped load fault

216. Hitachi-GE's ALARP study examined the use of a load follower whilst lowering spent fuel casks 21m during export [57]. Hitachi-GE concluded that an impact limiter in the base of the hoist well offers an equivalent level of protection to the cask but through a much simpler, passive protection system. Relevant good practice, adopted worldwide, is to fasten impact limiters to the exterior of packages for irradiated fuel (e.g. casks). However, for the spent fuel export route, Hitachi-GE proposes to install an impact limiter across the entire drop zone at the base of the hoist well. An ONR structural integrity inspector has assessed the performance of the impact limiter and its ability to protect the cask integrity. This response was the subject of two RQ's that ONR has now accepted (RQ-ABWR-0945, RQ-ABWR-0980).

4.6.7 Comparison of UK ABWR with J-ABWR reference design

217. Hitachi-GE's gap analysis between the J-ABWR and UK ABWR revealed that over raise protection on the fuel preparation machine required operators to set over-raise limits prior to each use. There was no means of detecting the presence of fuel within the cart with a potential risk of radiation dose to the operators. I am satisfied that Hitachi-GE has now modified this design by introducing over raise limit switches and a mechanical stop within the fuel preparation machine. Furthermore, Hitachi-GE proposes introducing a load cell to detect a loaded cart. The design prevents loaded carts from reaching the top of their lifting travel.

218. Hitachi-GE has not finalised the cask design for spent fuel export during GDA. For the purpose of GDA, Hitachi-GE has proposed the biggest cask system that is commercially available at this time. Based on J-ABWR and other recognised cask designs, the cask drop withstand is likely to be less than the height of the hoist well. Hitachi-GE has demonstrated that the canister can be designed with sufficient margin against the foreseeable drop faults as detailed in Hitachi-GE's report on canister integrity [58]. Hitachi-GE also identified further measures and design improvements that it considers will adequately protect the canister containment boundary, and reduce the likelihood of a drop down the hoist well. These include enhancements to the RBC, enhancements to lifting attachments, introduction of the impact limiter and introduction

of a cask stand in the cask pit to reduce risk ALARP. Hitachi-GE has committed to include these modifications into the final design of its spent fuel export route.

219. ONR has accepted that moving the Spent Fuel Storage Pool, re-designing the reactor building civil structure or substantially changing the layout is disproportionate. Hence, I am satisfied that it is acceptable for Hitachi-GE to dismiss options for the spent fuel export route that would involve substantial modification of the J-ABWR civil structure.

220. During the import of new fuel into the reactor building Hitachi-GE proposed to rotate the fuel from horizontal to vertical using the RBC. Hitachi-GE has indicated that dropping new fuel during this process has low nuclear safety risk. However, lifting loads must comply with the requirements of the Lifting Equipment and Lifting Operations Regulations (LOLER) [59]. I consider that rotation of a load introduces handling complexity that might be eliminated if the fuel could be rotated using an alternative method. I have identified this as an assessment finding detailed below for further consideration during detailed design after GDA.

4.6.8 Assessment of Hitachi-GE's Recovery of fuel handling equipment

221. I have based my assessment of recovery on a worst-case scenario that is, recovery of a canister containing spent fuel that has dropped through the full height of the hoist well [55]. Hitachi-GE has demonstrated that its proposed spent fuel export system and reactor building civil structures will all remain intact if a spent fuel cask drops down the hoist well and onto the impact limiter [60]. Hitachi-GE claims that no release of radioactive materials occurs; shielding remains intact and the cask continues to provide passive cooling. This allows operators sufficient time to adequately plan and undertake safe recovery of the cask. I am also satisfied that the ability for the cask to passively cool the fuel would prevent fuel from overheating if the Reactor Building Overhead Crane broke down during a lifting operation resulting in a suspended load.. Hitachi-GE has indicated that the radiological dose estimation, based on the assumed spent fuel export system, during repacking and recovery is double the dose estimate for a standard cask loading campaign. Nevertheless, the dose estimate is well within the numerical targets stated in ONR SAP NT.1.

222. I am satisfied that operating rules will be in place to prevent spent fuel in damaged canisters being transferred via the hoist well. Hitachi-GE has indicated that a method of recovery and repacking will exist should it be necessary to over pack a damaged canister albeit that this is not covered under GDA. The current design does not include such a facility although I am satisfied that Hitachi-GE has considered recovery to an appropriate level for this stage of GDA. I am satisfied that recovery is not time critical and therefore the licensee could safely develop Hitachi-GE's generic recovery plan into a detailed plan following an event.

4.6.9 Conclusions

223. I am satisfied that my assessment strategy provides me with an adequate sample to identify the SSCs that are important for safety.

224. I consider that Hitachi-GE has adopted a suitable design process that should enable it to satisfy the requirements of UK legislation and RGP.

225. Hitachi-GE has provided adequate evidence that its safety case has been prepared in accordance with its safety case development manual and GDA ALARP Methodology. I am satisfied that Hitachi-GE has a process which robustly enables it to consider normal operating and potential fault conditions including internal and external hazards, conventional safety and human factor influences that could affect safety.

226. The integrity of the SSCs has been demonstrated through to the point at which risks have been reduced ALARP. From a mechanical engineering perspective I am satisfied that Hitachi-GE's consideration of the fuel handling equipment is robust and offers adequate evidence to support its decision that a load follower for the spent fuel export route offers low levels of risk reduction for grossly disproportionate costs.

227. I consider the likelihood of mechanical failure due to inherent defects within the spent fuel export route to be low providing rigorous quality assurance regimes are applied during manufacture, and the associated level of EIMT is applied through lifetime of the plant. Hitachi-GE's approach should enable the licensee to identify SSCs important to safety.

228. I have identified an assessment finding for design improvement consideration for the new fuel-rotating task.

4.6.10 Regulatory Findings

229. **AF_ABWR_ME_003 -** Hitachi-GE proposes to use high integrity cranes designed for nuclear use for conventional lifting tasks associated with rotating incoming new fuel from horizontal to vertical. ONR considers that these lifts are complex and if incorrectly implemented could foreseeably lead to crane damage, crane collisions or dropped loads any of which could present a risk to nuclear safety. The licensee shall demonstrate, during detail design, that they have considered alternative methods for rotating fuel that do not involve using nuclear use cranes for complex lifting tasks. The licensee shall implement alternative methods wherever reasonably practicable.

4.7 Turbine Building Cooling Water Systems

4.7.1 Introduction

230. The turbine building service water system is located in the heat exchanger building. The pump and heat exchanger of turbine building cooling water system are also located in the heat exchanger building. The turbine cooling water system supplies water to the turbine auxiliary equipment for cooling. The turbine building service water system supplies sea water coolant to the turbine building cooling water heat exchangers, which in turn remove heat from the turbine building cooling water system.

231. My assessment sample is based on:

- Hitachi-GE's design process
- Claims, arguments and evidence for functional claims
- Examination, Inspection, Maintenance and Testing (EIMT)

4.7.2 Overview of Hitachi-GE's Safety Case

232. Hitachi-GE's Basis of Safety Case (BoSC) details design requirements for the turbine building cooling systems [61]. Hitachi-GE has made the following functional claims as follows:

- The turbine cooling water system supplies cooling water to the turbine auxiliary equipment.
- The turbine service water system supplies water to the turbine cooling water heat exchanger, which removes heat from the turbine cooling water system.

233. Earlier versions of the BoSC were assessed by ONR's Technical Support Contractor (TSC) against ONR SAP's and RGP during GDA step 3. The TSC's findings identified shortfalls in the evidence presented at that time. ONR presented these shortfalls in a report issued via RQ-ABWR-0649. This RQ required Hitachi-GE to:

1. familiarise itself with the report findings and observations,
2. confirm the report factual accuracy;
3. Hitachi-GE to respond to the TSC findings through the formal GDA RQ process for ONR assessment

234. Throughout step 4, ONR has engaged with Hitachi-GE through various Level 4 engagements. During these engagements, I have challenged Hitachi-GE on the shortfalls identified by the TSC and this has resulted in revised submissions of the BoSC. I am now satisfied that Hitachi-GE's revised BoSC provides sufficient evidence to address the TSC's initial observations and findings.

235. Hitachi-GE has provided piping and instrumentation diagrams [62] [63] detailing the design of the turbine building cooling water systems. I am content that the proposed design meets the system intent.

236. Hitachi-GE has provided claims arguments and evidence for categorisation and classification of the turbine building water systems. I am satisfied that Hitachi has satisfied the guidance given in NS-TAST-GD-094 [37].

4.7.3 Strategy for Assessment of Turbine Building Cooling Water Systems

237. The main document sampled for the purposes of this assessment was the BoSC for the turbine buildings cooling systems [61]. Other references used to inform my judgement are:

- Turbine cooling water system piping and instrumentation diagram [62]
- Turbine service water system piping and instrumentation diagram [63]
- Topic report on Codes and Standards [64]
- Topic report on Safety Requirements for Mechanical SCC's [65]

4.7.4 Equipment Description

- Turbine Cooling Water Pumps – The purpose of the turbine cooling water pumps is to supply cooling water to the turbine auxiliary equipment.
- Turbine Cooling Water Heat Exchangers – The purpose of the turbine cooling water heat exchangers is to remove heat from the turbine cooling water system.
- Turbine Cooling Water Surge Tank – The purpose of the turbine cooling water surge tank is to act as a reservoir for small amounts of leakage from the system and for the expansion and contraction of the cooling water when the system temperature changes.
- Turbine Service Water Pump – The purpose of the turbine service water pump is to supply seawater from the intake pool to the turbine cooling water system heat exchanger and draw off water to the discharge pool.

- Turbine Service Water Strainer – The purpose of the turbine service water strainer is to protect the turbine cooling water heat exchanger and system pipes against blockage by straining debris and marine organisms from the seawater.

4.7.5 Assessment of Hitachi-GE's Design Process for turbine building cooling water systems

238. Hitachi-GE has conducted the design process for the turbine building cooling systems in accordance with its "General design process approach for mechanical SSCs" [18]. The BoSC for the turbine building cooling system identifies the ALARP process, a risk assessment and compliance to regulations, codes and standards. I consider that the BoSC provides an adequate demonstration that Hitachi-GE has applied RGP throughout the design process. Furthermore, I am content that Hitachi-GE has applied relevant codes and standards, which aligns with guidance in SAP ECS.3 codes and standards and therefore meets ONR expectations for application of RGP.

4.7.6 Assessment of Hitachi-GE's Functional Claims for turbine building cooling water systems

239. I am satisfied that Hitachi-GE has identified the functional claims using a structured analysis as set out in guidance in ONR SAP EKP.4 safety function. Hitachi-GE has made the following functional claims:

- The turbine cooling water system supplies cooling water to turbine auxiliary equipment

- The turbine service water system supplies service water to the turbine cooling water system heat exchanges and removes heat from the heat exchangers

240. Hitachi-GE has provided arguments and evidence to support these claims within the BoSC [61]. One such argument states that instrumentation and control provisions to prevent and detect failure of the piping are in place by monitoring of process parameters and detection of abnormalities. Hitachi-GEs' piping and instrumentation diagrams [62] [63] and BoSC indicates the instrumentation used along with intended alarm levels. The diagrams also show flow paths and auxiliary equipment supply. I consider the evidence provided in the piping and instrumentation diagrams substantiates that the systems provide adequate means of heat transfer and cooling safety functions.

4.7.7 Assessment of Hitachi-GE's Examination, Inspection, Maintenance and Testing (EIMT) for turbine building cooling water systems

241. Hitachi-GE has claimed that the turbine building cooling water systems design considers EIMT. I assessed the system pumps to seek evidence that the EIMT provisions are adequate. I selected these as my sample due to their safety functional importance supporting heat removal.

242. Hitachi-GE identified the pumps as class 3 within the BoSC [61], with EIMT carried out during power operation and/or refuelling outage periods. Hitachi-GE also claims that those components not designed for a 60-year design life will be replaced to ensure delivery of the pump safety functions. I consider that Hitachi-GE's proposed EIMT provisions are aligned with guidance in SAP EMT.1 identification of requirements.

243. I acknowledge that further detailed considerations of EIMT will occur during detailed design. My expectation is that responsibility for making and implementing adequate EIMT arrangements in respect of licence conditions will rest with the licensee.

244. I have identified an assumption that should be considered by a future licensee that detail design adequately considers EIMT requirements (AS-ABWR-ME05).

4.7.8 Conclusions

245. I consider that Hitachi-GE has adopted a suitable design process that should enable it to meet the requirements of relevant UK legislation and RGP.

246. I consider the likelihood of mechanical failure due to inherent defects within the turbine building cooling water systems to be low providing that adequate EIMT is undertaken throughout the lifetime of the plant.

247. I am satisfied that Hitachi-GE has adequately considered EIMT in its safety case submission. I have identified an assumption, captured through an ONR generic finding, that detail design adequately considers EIMT requirements.

4.7.9 Regulatory Assumption

248. **AS-ABWR-ME05** - ONR acknowledges that during GDA, Hitachi-GE has not sufficiently developed its design to enable the licensee to identify the appropriate EIMT requirements in line with UK expectations. ONR assumes that the licensee shall ensure that they consider EIMT requirements for all mechanical engineering items that attract a safety classification. ONR also assumes that the licensee shall establish whether the safety case drives these requirements, whether they are manufacturer recommendations or are based on plant operating experience (or appropriate combinations).

4.8 Turbine Main Steam System

4.8.1 Introduction

249. The role of the Main Steam (MS) system is to supply steam generated from the reactor to the steam turbine to enable power generation. The MS system also supplies steam to the Auxiliary Steam (AS) System and the Turbine Bypass system (TBP).

250. My assessment sampled particular areas of interest as follows:

- Categorisation and classification
- Mechanical Engineering Design Process
- Examination, Inspection, Maintenance and Testing (EIMT)

4.8.2 Overview of Hitachi-GE Safety Case

251. Hitachi-GE's Basis of Safety Case (BoSC) details design requirements for the MS system [66]. Three key hazards identified are:

- Loss of reactor coolant
- Reactor pressure increase
- Loss of power supply

252. During GDA step 3 the BoSC for the turbine main steam system where assessed by ONR's Technical Support Contractor (TSC) against ONR SAP's and RGP. The TSC identified shortfalls in Hitachi-GE's evidence in these earlier versions of the BoSC.

ONR presented these shortfalls in a report that was issued via RQ-ABWR-0643. This RQ required Hitachi-GE to:

i. familiarise itself with the report findings and observations,

ii. confirm the report factual accuracy;

iii. respond to the TSC findings through the formal GDA RQ process for ONR assessment

253. Throughout step 4, I have engaged with Hitachi-GE through various level 4 engagements. During these engagements, I have challenged Hitachi-GE on the shortfalls identified by the TSC and this has resulted in revised submissions of the BoSC. I am now satisfied that Hitachi-GE's revised BoSC provides sufficient evidence to address the TSC's initial observations and findings.

254. The Pre-Construction Safety Report (PCSR) [67] summarises the case presented in the BoSC. Topic reports present detailed analysis results for fault assessment.

255. Hitachi-GE has provided the general arrangement drawings [68] for the turbine building along with system piping and instrumentation diagrams. This describes the mechanical design of the systems along with consideration of EIMT activities.

256. Hitachi-GE has provided claims arguments and evidence for categorisation and classification of the MS system. I am satisfied that this aligns with assessment guidance given in NS-TAST-GD-094 [37].

4.8.3 Strategy for Assessment of Turbine Main Steam System

257. The main document I sampled was the BoSC for the MS system [66]. Additional references used to inform my judgment include:

- PCSR [67]
- Turbine building general arrangement drawings [68]
- Topic report on Codes and Standards [69]
- Topic report on Safety Requirements for Mechanical SCC's [65]

258. ONR has undertaken numerous interactions with Hitachi-GE relating to the MS system at all stages of GDA via Level 4 contacts and Regulatory Queries (RQs). The significant mechanical engineering interactions are summarised in my assessment below.

4.8.4 Equipment Description

- Main Steam (MS) system - the main roles of the MS are to supply steam generated by the reactor to drive the steam turbine, and supply steam to the Auxiliary Steam (AS) System and Turbine By-Pass (TBP) System.
- Auxiliary Steam (AS) system - the main roles of the AS system are to supply driving steam to the reactor feed pump turbine and the steam jet air ejector and to supply heating steam to the moisture separator re heater and the gland steam evaporator.
- Turbine Bypass (TBP) system - the main role of the TBP system is to release steam from the reactor when steam production exceeds the turbine speed

demand. It also releases steam from the reactor to the condenser for reactor internal pressure control during plant start-up and shutdown operations.

4.8.5 Assessment of Hitachi-GE's Design Process for Main Steam system

259. The MS system BoSC gives a general overview of the design process set out within Hitachi-GE's general design process approach to mechanical SSC's [17]. A risk assessment is included highlighting hazards concerning the MS system, along with an audit trail of mitigation. My assessment of a nuclear safety audit trail for the topic report on FMEA [70] indicates that Hitachi-GE has identified appropriate hazards. Hitachi-GE provides a claim table for the mitigation and consequences of identified faults contained within the BoSC. This provided me with sufficient evidence that Hitachi-GE has followed RGP in its design to reduce risk ALARP. Furthermore, I consider that Hitachi-GE's design philosophy aligns with ONR SAP EDR.1 failure to safety.

4.8.6 Assessment of Hitachi-GE's Safety Function and Safety Property Claims for turbine main steam system

260. In consideration of ONR SAP EKP.4 I am satisfied that Hitachi-GE has identified the safety function(s) to be delivered using a structured analysis. The key faults identified by Hitachi-GE are as follows:

- Loss of reactor coolant
- Reactor pressure increase
- Loss of Power supply

261. Hitachi-GE has assigned the MS system with safety categorisations that align with ONR SAP ECS.2 safety classification of structures, systems and components. It has also ensured that failure of a lower class SSC will not propagate to an SSC of higher safety class. Hitachi-GE is proposing a reliability of between10^{-1} and 10^{-2} for the MS system that I consider reasonable based on the ONR TAG on categorisation and classification [37].

262. Hitachi-GE's BoSC indicates that Hitachi-GE has supported its claims by comparison with similar equipment/systems within the J-ABWR. The schematic diagrams presented within the BoSC provide detail of the design layout, matching that of the design intent. Detail design & production of a mature process and instrumentation diagram (P&ID) for the MS system after GDA will be necessary to substantiate the design claims made. I consider that this approach should enable the licensee to undertake detailed design, manufacture and procurement to an appropriate classification.

263. Hitachi-GE provided sufficient evidence that it undertook an adequate gap analysis against the reference J-ABWR design considering relevant UK legislation, UK RGP and OPEX (as observed in RO-ABWR-0045). I consider this approach should enable the licensee to apply UK RGP to the detailed design manufacture and procurement of the MS system.

4.8.7 Assessment of Hitachi-GE's Control Systems for Main Steam system

264. In the BoSC [17], Hitachi-GE states that the design incorporates control systems to prevent and detect failure of the MS system. The BoSC describes the details of the instrumentation to provide control using schematic diagrams albeit that information is limited on how control is performed. The BoSC provides limits and conditions on the systems that would inform the development of the control system.

265. I recognise that further detailed considerations for limits and conditions regarding the control systems will occur during detailed design. My expectation is that responsibility for making and implementing adequate limits and conditions arrangements in respect of licence conditions will rest with the licensee. I have identified an assumption that should be considered by a future licensee (AS-ABWR-ME03)

4.8.8 Assessment of Hitachi-GE's Decommissioning for Main Steam system

266. ONR SAP SC.3 life cycle aspects suggests that for each lifecycle stage, control of the hazard should be demonstrated by a valid safety case that takes into account the implications from previous stages and for future stages. I consider that Hitachi-GE has presented adequate information within the sampled PCSR and BoSC documents to align with the recommendations of this SAP.

4.8.9 Assessment of Hitachi-GE's Examination, Inspection, Maintenance and Testing (EIMT) for Main Steam system

267. To confirm that Hitachi-GE has made adequate provisions for EIMT I sampled relevant sections with in the BoSC and PCSR. I acknowledge that further detailed considerations for EIMT will occur during detailed design. My expectation is that responsibility for making and implementing adequate EIMT arrangements in respect of licence conditions shall rest with the licensee. I have identified an assumption that should be considered by a future licensee (AS-ABWR-ME05)

4.8.10 Conclusions

268. I consider that Hitachi-GE has adopted a suitable design process that should enable the licensee to satisfy the requirements of relevant UK legislation and RGP.

269. I am satisfied that Hitachi-GE has a process which has enabled it to identify and address normal operating and potential fault conditions including internal and external hazards, conventional safety and human factor influences that could affect safety.

270. I consider the likelihood of mechanical failure due to inherent defects within the MS system to be low providing the licensee implements EIMT throughout the lifetime of the plant.

271. I am satisfied that Hitachi-GE has considered EIMT in their safety case submission. I have identified an assumption (AS-ABWR-ME05) that detailed design adequately considers EIMT requirements.

272. I am satisfied that Hitachi-GE has considered limits and conditions in its safety case submission. I have identified an assumption (AS-ABWR-ME03) that detailed design adequately considers limits and conditions.

4.8.11 Regulatory Findings or Shortfalls (if applicable)

273. **AS-ABWR-ME-03** – ONR acknowledges that during GDA, Hitachi-GE had not sufficiently developed its design to enable the licensee to identify the appropriate plant limits and conditions. ONR assumes that the licensee shall identify plant limits and conditions, from the safety case, covering all mechanical engineering equipment important to safety. ONR also assumes that the licensee shall generate sufficient safety case information to satisfy the requirements of LC 23, and specifically they shall establish a suitable interface for transferring this information from the responsible designer.

274. **AS-ABWR-ME-05** – ONR acknowledges that during GDA, Hitachi-GE has not sufficiently developed its design to enable the licensee to identify the appropriate EIMT requirements in line with UK expectations. ONR assumes that the licensee shall

ensure that they consider EIMT requirements for all mechanical engineering items that attract a safety classification. ONR also assumes that the licensee shall establish whether these requirements are directly driven by the safety case, are based on manufacturer recommendations or are based on plant operating experience (or appropriate combinations).

4.9 Severe Accident Mechanical Systems

4.9.1 Introduction

275. Severe accident mechanical systems is a term used by Hitachi-GE to describe the mechanical systems that provide a back-up safety function in the event of a severe accident. Some of these systems must deliver their safety function under design basis fault conditions and in some cases beyond design basis faults.

276. The safety functions delivered by the severe accident mechanical systems and the means by which it is achieved are:

- Reactor core cooling (reactor pressure vessel water injection)
- Drywell cooling (primary containment volume water spray)
- Molten core cooling (lower drywell water injection)
- Primary containment vessel head flange cooling (reactor well water injection)
- Spent fuel pool cooling (spent fuel pool water spray)
- Primary containment volume heat removal (including primary containment volume overpressure protection).

4.9.2 Overview of Hitachi-GE Safety Case

277. Hitachi-GE has produced a Basis of Safety Case (BoSC) document for the severe accident mechanical systems [71]. The Pre-Construction Safety Report (PCSR) summarises the case presented in the BoSC.

278. Some severe accident mechanical systems have a safety function to provide a backup to the primary means of protection for certain design basis faults. I have sampled some of these systems described by Hitachi-GE, for example, the BoSC's for Emergency Core Cooling Systems [32] and the Containment Heat Removal System [72].

279. The topic report for the flooder system of specific safety facility pump is not available at GDA step 4. Therefore, I have based my assessment of this component on the information available in the BoSC.

4.9.3 Strategy for Assessment of Severe Accident Mechanical Systems

280. During my step 4 assessment, I sampled the following aspects of the severe accident mechanical systems:

The flooder system of specific safety facility. I selected this system for assessment based on its safety significance and because it is required to deliver its safety functions in the event of design basis faults, beyond design basis faults and severe accident conditions.

The fusible plug within the Lower Drywell Flooder System. I selected this system because it is the last available means of cooling the corium (melted reactor core) in the event of a severe accident.

4.9.4 Equipment Description of Flooder System of Specific Safety Facility

281. The flooder system of specific safety facility consists of two independent systems (known as trains) each comprising two pumps, which perform coolant injection to a number of destinations to support the required cooling function. Ten dedicated water storage tanks supply the flooder system with water. The system includes the suction lines from the tanks, the injection lines to the required destinations as well as the associated valves, instrumentation and controls.

4.9.5 Assessment of Hitachi-GE's Safety Functions and Equipment Classification for Flooder System of Specific Safety Facility

282. Hitachi-GE identifies the safety functions for the flooder system of specific safety facility in the BoSC [71]. During design basis faults, the flooder systems are required to provide a backup means of cooling the reactor and the spent fuel storage pool. For beyond design basis faults and severe accident events, the flooder system of specific safety facility is also required to provide cooling to prevent primary containment vessel failure. I am satisfied that Hitachi-GE has presented evidence to indicate that it has identified these safety functions by following a structured analysis in line with ONR SAP EKP.4 – Safety function.

283. The safety functions to support reactor core cooling and Spent Fuel Storage Pool cooling in the event of design basis faults are category A safety functions (i.e. they play a principal role in ensuring nuclear safety). The safety functions for beyond design basis faults are category B safety functions (i.e. they make a significant contribution to nuclear safety). The flooder system of specific safety facility components are designed to meet class 2 requirements (i.e. they make a significant contribution to fulfilling a category A function). Hitachi-GE has based this categorisation on their process for categorisation of safety functions and classification of SSCs [73]. I sampled the categorisation of the safety functions and classification of the flooder system of specific safety facility and I am satisfied that it is appropriate and aligns with SAP ECS.1 – Safety categorisation and SAP ECS.2 – Safety classification of structures, systems and components.

4.9.6 Assessment of Hitachi-GE's Limits and Conditions of Operation for Flooder System of Specific Safety Facility

284. Limits and conditions are required to comply with Licence Condition (LC) 23 – Operating rules. The BoSC [71] outlines limits and conditions of operation for the flooder system of specific safety facility and the required action to be taken if these limits and conditions of operation cannot be met. Limits and conditions of operation are considered for power operation, start-up, shutdown, and refuelling. Hitachi-GE has also identified the surveillance requirements to verify that the flooder system of specific safety facility is operable. I am satisfied that Hitachi-GE has adequately presented these aspects at step 4. My expectation is that the licensee will develop detailed limits and conditions of operation for other lifecycle phases, for example during commissioning and de-commissioning, at the appropriate stage.

4.9.7 Assessment of Hitachi-GE's Flow Requirements Flooder System of Specific Safety Facility

285. The BoSC [71] sets out the required flow rate and pressure requirements for the flooder system of specific safety facility pumps to meet their safety functions. ONR queried [74] the claimed injection rate and pressure for the flooder system of specific

safety facility under various modes of operation. Hitachi-GE's response [75] sets out the required flow rate for each fault scenario. This response also sets out the required injection head that the licensee could use to determine the performance requirement for the pumps. From a mechanical engineering perspective, I am satisfied sufficient information has been provided by Hitachi-GE at step 4.

4.9.8 Assessment of Hitachi-GE's Examination, Inspection, Maintenance and Testing (EIMT) for Flooder System of Specific Safety Facility

286. Hitachi-GE has outlined provisions for EIMT in the BoSC [71]. For the flooder system of specific safety facility pump, this includes visual inspection, checks of moving parts and replacement of consumables. Hitachi-GE has not defined the frequency of these activities in step 4, but plans to adopt an 18-month outage cycle. Based on the information presented in the BoSC, I am satisfied that EIMT provisions can be developed further during detailed design to align with ONR SAP EMT.1 – Identification of requirements and SAP EMT.2 - Frequency.

4.9.9 Equipment Description of Fusible Plug

287. There are 10 fusible plugs capable of automatically releasing suppression pool water into the lower drywell if environmental temperatures become too high. Its purpose is to cool the corium (melted core), in an attempt to prevent a molten core-concrete interaction. This would occur following a severe accident that has led to core meltdown, vessel failure and deposition of molten corium on the lower dry well floor.

288. A fusible plug is a passive device comprising a normally closed valve that prevents water flowing from the suppression pool during normal primary containment vessel operating temperatures (design temperature does not exceed 171 °C). A series of levers and linkages hold the valve closed until a fusible alloy plug melts when environmental temperatures reach 260°C. Once the plug melts, it allows a weight to drop, providing sufficient force to move the levers and open the valve to allow the water to flow.

4.9.10 Assessment of Safety Function and Safety Property Claims for the fusible plug

289. ONR raised action 4.21 of RQ-ABWR-0521 requesting that Hitachi-GE provide its qualification process for the fusible plug concept design. In response, Hitachi-GE prepared Reference [76], which outlined its optioneering process and design concept qualification. Hitachi-GE's optioneering considered the key safety functions that the plug must perform i.e. Perform reliably at extremes of its temperature range; achieve leak tightness when closed and provide the necessary water flow rate when open. Hitachi-GE based its evidence on qualification work performed for previous US ABWRs. I am satisfied that this is sufficient for Hitachi-GE to select a conceptual design that meets UK requirements and therefore RQ-ABWR-0521 has been satisfied.

290. Hitachi-GE claims that the fusible plug's ability to flood the lower drywell in the event of a severe accident needs to satisfy class 3 requirements [71]. I consider this is reasonable, as its only purpose is mitigation after several other higher safety classification systems have failed to perform and the core has melted and flowed into the lower drywell. In contrast, the plug's ability to seal the suppression pool water during normal reactor operations must satisfy class 1 requirements. I consider this is an appropriate class for the fusible plug, as it must prevent leakage of suppression pool water during normal operation.

291. Hitachi-GE has designed the lower drywell flooder with 10 fusible plugs providing 110 litres/second (l/s) total capabilities. The BoSC for Severe Accident Mechanical Systems [71] states that; the Lower Drywell Flooder is capable of providing a coolant flowrate of 22l/s and that this value comes from the Severe Accident Analysis. The

Severe Accident Analysis states that 22l/s are achieved with two of the ten Lower Drywell injection lines. None of the other safety case documents refers to two valves delivering the required flow. Hence, ONR Fault Studies specialist inspectors sought clarification of safety case claims against accident analysis assumptions raised in RQ-ABWR-1468. Hitachi-GE responded to RQ-ABWR-1468 [77] clarifying the flow rates and committing to ensure that all relevant documentation is consistent.

4.9.11 Comparison of UK ABWR with J-ABWR reference design for fusible plug

292. The J-ABWR reference design does not use fusible plugs but Hitachi-GE has adopted a US-ABWR design of fusible plug. Hitachi-GE has applied its design process (assessed elsewhere in this step 4 report) to the UK ABWR fusible plug, which has resulted in a change from the US ABWR design. Hitachi-GE has indicated [78] that it may need to carry out further qualification through additional design development and will perform an ALARP evaluation in accordance with its design process. I am satisfied that this process should confirm that the fusible plug satisfies UK requirements and should be adequate to demonstrate that risk is ALARP. To capture my assumption that Hitachi-GE will carry out further qualification. I have identified an assumption that should be considered by a future licensee (AS-ABWR-ME02).Assessment of Examination, Inspection, Maintenance and Testing (EIMT) for fusible plug

293. ONR challenged reliability claims made by Hitachi-GE on its revised plug design. Hitachi-GE proposed a 15-year replacement strategy for the fusible plug with intermediate visual inspection and no intermediate functional testing. ONR considered that it was foreseeable for the valve to start leaking, the actuator mechanism could seize, the valve could seize or stick and the fusible element could become damage or detached.

294. During a Level 4 technical exchange, Hitachi-GE presented reference [79] in response to ONR’s challenge on reliability. This presentation may be summarised as follows:

- Hitachi-GE expects a fusible plug to have a 15 year life expectancy and its case is based on all 10 plugs being replaced every 15 years following a replacement programme to be determined during detailed design. After removing the plugs, destructive functional testing will verify their performance is acceptable.

- The sealing gasket will be non-corrosive graphite and the intention is to demonstrate its performance through engineering qualification. The gasket and fusible element are consumable parts to be replaced every 5 years based on US ABWR OPEX. Engineering qualification tests will demonstrate the validity of this period. Hitachi-GE’s case assumes that visual inspection will be sufficient to confirm the validity of the period.

- Hitachi-GE has designed the valve actuation linkages to be stainless steel, which coupled with the location in the lower drywell, should minimise corrosion potential. Hitachi-GE does not consider it will be necessary to lubricate the stainless steel mechanism to prevent seizure from galling.

- Hitachi-GE is relying on a visual examination, during interim periodic maintenance, to check for leaks and corrosion.

- The intention is to minimise the risk of a ‘failure to open’ fault by checking the spring set force during periodic maintenance.

295. I am satisfied that Hitachi-GE has adequately considered EIMT for the concept design and has indicated that all functional testing will be conducted under the equivalent thermal and radiation conditions experienced during the actual 15 years plant operation.

296. Hitachi-GE has identified the need to define EIMT after the completion of engineering qualification tests. I consider that this qualification and consideration of EIMT is an important requirement, as the J-ABWR reference design cannot provide EIMT experience since it does not use fusible plugs. I have identified an assumption that should be considered by a future licensee (AS-ABWR-ME05), as detailed below.

297. Hitachi-GE has identified EIMT in [71] and I consider that Hitachi-GE's approach aligns with ONR SAP EMT.1, which requires safety requirements for in-service testing, inspection and other maintenance procedures and frequencies to be identified in the safety case.

298. I am satisfied that EIMT provisions, for example set out in ONR SAP EMT.2, which states that SSCs should receive regular and systematic EIMT as defined in the safety case, can be developed in full during detail design.

4.9.12 Conclusions

299. I am satisfied that Hitachi-GE has followed its revised design process for UK ABWR that has resulted in a design change to the fusible plugs used on US-ABWR.

300. I am satisfied that Hitachi-GE has used international OPEX from the US-ABWR to propose a suitable engineering qualification process and EIMT regime for their fusible plug concept design.

301. I consider that a future licensee can develop engineering qualification and EIMT details during detail design.

4.9.13 Regulatory Assumptions

302. **AS-ABWR-ME02 –** ONR acknowledges that during GDA, Hitachi-GE had not sufficiently developed its design to enable the licensee to identify the appropriate design qualifications details. ONR assumes that the licensee shall establish detailed design substantiation, factory acceptance test information, and site acceptance test information for individual mechanical items and their associated systems, which are important to safety. ONR also assumes that the licensee shall generate appropriate evidence that equipment qualification is adequately specified for all mechanical items important to safety in accordance with UK expectations.

303. **AS-ABWR-ME05 –** ONR acknowledges that during GDA, Hitachi-GE has not sufficiently developed its design to enable the licensee to identify the appropriate EIMT requirements in line with UK expectations. ONR assumes that the licensee shall ensure that they consider EIMT requirements for all mechanical engineering items that attract a safety classification. ONR also assumes that the licensee shall establish whether these requirements are directly driven by the safety case, are based on manufacturer recommendations or are based on plant operating experience (or appropriate combinations).

4.10 Safety Relief Valves

4.10.1 Introduction

304. The UK ABWR has 16 safety relief valves designed to provide overpressure protection to the reactor coolant pressure boundary. The safety relief valves also have safety functions to depressurise the reactor pressure vessel under certain fault conditions to support reactor core cooling.

4.10.2 Overview of Hitachi-GE Safety Case for Safety Relief Valves

305. Hitachi-GE has produced a Basis of Safety Case (BoSC) document for reactor coolant boundary overpressure protection systems [80]. The BoSC for the nuclear boiling system [81], emergency core cooling system [32] and severe accident mechanical systems [71] also include claims on the safety relief valves. The Pre-Construction Safety Report (PCSR) summarises the case presented in the BoSCs. The topic report for the safety relief valves [82] provides detailed technical substantiation for these components to support the BoSC documents.

4.10.3 Assessment Strategy for the Safety Relief Valves

306. At step 4, I used a Technical Support Contractor (TSC) to review a sample of the Hitachi-GE submissions for the safety relief valves. The objective was to review the adequacy of the evidence contained within the BoSC [80] submissions and topic report [82] for the safety relief valves, from a mechanical engineering perspective. This review built upon the work already carried out by the TSC at step 3. The step 4 review included determining whether Hitachi-GE had adequately addressed the findings and observations identified at step 3.

307. The key documents considered by the TSC were the BoSC [80] for overpressure protection, the topic report [82] and the response to RQ-ABWR-0635 [83]. Hitachi-GE has generally presented its evidence supporting the safety case in references to these documents. The TSC reviewed a sample of the supporting references to assess the adequacy of this evidence at step 4.

308. The TSC has listed all the documents they sampled in its report [84]. I oversaw the technical support contractor’s work through regular technical and progress meetings and by reviewing its final report [84]. I am therefore content that the review completed by the TSC meets my expectations.

309. For my assessment of diversity of safety relief valves, I considered ONR SAPs EDR.2 to EDR.4 – engineering principles design for reliability. This SAP provides guidance on diversity to prevent common cause failure (CCF). I also considered the ONR Technical Advice Guide on Redundancy, Diversity and Segregation of Mechanical Plant [85].

310. I initially assessed Hitachi-GE’s document, demonstration of adequacy of Safety Relief Valve design [86]. I concluded, from a mechanical engineering perspective, that Hitachi-GE had not provided sufficient evidence to show that it had considered all potential fault scenarios. Hitachi-GE had only introduced diversity measures covering two fault scenarios and therefore Hitachi-GE had not addressed the potential for common cause failure. ONR fault studies and PSA specialists also raised concerns over diversity of safety relief valves with Hitachi-GE. ONR fault studies inspectors are leading this issue.

311. In response to ONR’s challenges, Hitachi-GE prepared a topic report on diversity of safety relief valves [87]. I have now assessed this topic report and discuss my findings below.

4.10.4 Equipment Description for the Safety Relief Valves

312. The UK ABWR has 16 safety relief valves installed in the main steam lines to provide overpressure protection for the reactor. Each safety relief valve is a spring-loaded valve that opens when the steam pressure exceeds the pressure exerted by the spring. The valve controls ensure that the reactor circuit pressure is cannot exceed the head of the high-pressure safety injection systems (reactor core isolation cooling system and high-pressure core flooder system) which form part of the reactor core cooling system.

313. The valves are also capable of actuation using nitrogen gas applied to the actuator cylinder. Hitachi-GE's design provides various other systems to detect faults and provide a signal to operate the nitrogen system. I have therefore not considered the nitrogen system in the scope of my assessment. ONR's fault studies specialists led on assessment of this system and I am satisfied that there were no mechanical engineering issues of concern, associated with this system.

314. The automatic depressurisation system is capable of actuating seven of the 16 safety relief valves. This system enables the safety relief valves to open automatically in the event of a loss of coolant accident that allows water injection into the reactor pressure vessel at low pressure. Figure 1 shows the configuration of the UK ABWR safety relief valve.

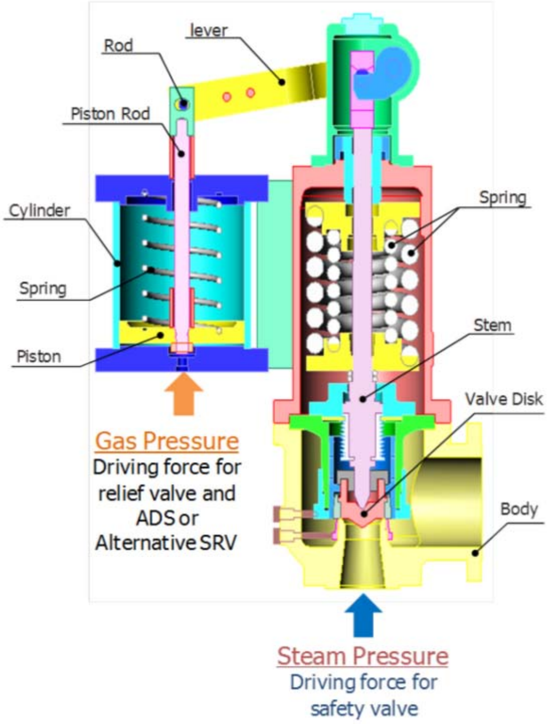

Figure 1 UK ABWR Safety Relief Valve

315. Re-seating of the SRV after it has lifted to depressurise the reactor is necessary to prevent excessive loss of steam into the suppression pool. This is identified as a preventative safety function, the failure of which would lead to a demand on other category A safety systems (i.e. systems that play a principal role in nuclear safety) to provide mitigation.

4.10.5 Safety Functions and Equipment Classification for Safety Relief Valves

316. The topic report [88] identifies the safety functions for the safety relief valves. I am satisfied that Hitachi-GE has identified these safety functions by a structured analysis, which aligns with ONR SAP EKP.4 – Safety function.

317. The principal (category A) safety functions for the safety relief valves are:

- To deliver long-term residual heat removal to reach reactor cold shutdown. This is achieved by depressurization of the reactor pressure vessel in the event of unavailability of the main condenser. (Hitachi-GE safety claim reference NB SFC 3-1.1)

- To release the steam generated during reactor core cooling by high-pressure core cooling systems in the event of faults such as a loss of coolant accident outside the primary containment vessel. (Hitachi-GE safety claim reference NB SFC 2-1.1)
- To deliver overpressure protection of the reactor coolant pressure boundary under abnormal transients and accident conditions that could put excessive pressure on the boundary. (Hitachi-GE safety claim reference NB SFC 4-2.1)
- To depressurise the reactor pressure vessel for reactor core cooling, in a low-pressure state, as part of the emergency core cooling system. This is required so that significant damage to the fuel in prevented, and the reaction between the fuel cladding and the reactor coolant in minimised in the event of a loss of coolant accident. (Hitachi-GE safety claim references NB SFC 2-1.3 and NB SFC 2-1.4)
- To maintain the reactor coolant pressure boundary. This is required to contain reactor coolant during normal operations and form a pressure barrier during fault conditions. (Hitachi-GE safety claim reference RCPB SFC 4-1.2)
- To provide a means of depressurising the reactor pressure vessel in order to provide reactor cooling in the low-pressure state. (Hitachi-GE safety claim references RDCF SFC 2-2.1, RDCF SFC 2-2.2 and RDCF SFC 2-2.3)

318. The topic report [82] lists other safety functions, with a lower safety categorisation for the safety relief valves. The safety relief valves are designed to meet class 1 requirements (i.e. a component that forms a principal means of fulfilling a category A safety function). Hitachi-GE has based this classification on its process for categorisation of safety functions and classification of SSCs [73]. I sampled the categorisation of the safety functions and classification of the safety relief valves as part of my step 4 assessment. I am satisfied that it is appropriate and aligns with SAPs ECS.1 – Safety categorisation and ECS.2 – Safety classification of structures, systems and components.

319. The Technical Support Contractor (TSC) sampled the claims, arguments and evidence at step 4. It concluded that the BoSC provides good arguments and evidence to satisfy the claims and that there is good traceability with referenced evidence. Based on my review of the TSC's report, I am content that the evidence presented by Hitachi-GE is adequate for GDA step 4. I recognise that the preparation of further evidence will develop during detailed design, which ONR can assess as appropriate under normal regulatory business.

4.10.6 Safety Relief Valve Failure Analysis

320. Reference [87] indicates that the key concern is failure to open, related to the spring actuation of the safety relief valve. Hitachi-GE has considered spring failure rates in its analysis. If the spring failed, there is a partially diverse means of relief valve back up actuation via a pressurised nitrogen system. In the case that all means of actuating the safety relief valve fail due to a common cause failure, damage could occur to the reactor coolant pressure boundary, leading to a loss of coolant accident (LOCA).

321. Hitachi-GE carried out a Failure Modes and Effects Analysis (FMEA) of the safety relief valve at a component level. I sampled this analysis and was satisfied that it considered appropriate fault scenarios. However, Hitachi-GE concluded that sticking and galling is a potential common cause failure, leading to the valve disc sticking to its seat in the valve body. I consider this the most likely cause of safety relief valves 'failing to open'. UK RGP and OPEX indicate that alternative mechanisms might also prevent valves from opening. For example, the Relief Systems Handbook [89], is a recognised source of RGP with respect to safety relief valves and identifies:

- Damage to sliding surfaces caused by vibration, chatter or corrosion

- Foreign material in the valve bonnet preventing the valve stem from moving

322. Operational experience in the UK also suggests that other mechanisms can cause valve seats to stick. For example, chemical interaction can occur between the seat material and the disc. The Approved Code of Practice (ACOP) accompanying the Pressure Systems Safety Regulations 2000 [90] states that materials used in construction should be suitable for the intended use taking account of foreseeable fault conditions. From the evidence presented, it is not evident to me how Hitachi-GE was able to dismiss these alternative failure mechanisms following its FMEA process.

4.10.7 Safety Relief Valve OPEX

323. Hitachi-GE indicated that some similar plants use pilot operated safety relief valves rather than direct acting spring loaded valves. Hitachi-GE's analysis of OPEX from these plants has led them to consider that the majority of failures occurred in the pilot units of the safety relief valves. Hitachi-GE has indicated [51] that OPEX of spring operated safety relief valves based on 30 years of experience in Japan shows no failures to open due to sticking and galling. Therefore, Hitachi-GE considers that replacing pilot operated valves with direct spring operated valves should eliminate most of the foreseeable faults associated with safety relief valves. I am satisfied with this philosophy based on Hitachi-GE's OPEX.

4.10.8 Safety Relief Valve Qualification

324. ONR SAP EQU.1 – Qualification procedures, states that qualification procedures should be applied to confirm that structures, systems and components would perform their allocated safety function(s) in all normal operational, fault and accident conditions identified in the safety case for the duration of their operational lives. Qualification plans for the safety relief valves are summarised in the BoSC [80] and topic report [82]. Hitachi-GE has re-sized the safety relief valves for the UK ABWR to improve efficiency. In my judgement, the arguments made for qualification of the safety relief valves are adequate for step 4. Based on the review carried out by the TSC, I have sufficient confidence that qualification procedures will be applied following detailed design to align with ONR SAP EQU.1 – Qualification procedures.

4.10.9 Safety Relief Valve Diversity Considerations

325. Hitachi-GE considers that physical changes, to introduce diversity into the safety relief valves are disproportionate compared to the risk reduction they could offer. Hitachi-GE claims that the safety relief valve design is reliable and is supported by many years of OPEX on relevant plant. A design change would not have this supporting OPEX.

326. Hitachi-GE has considered options to improve diversity of safety relief valves against common cause failure in the UK ABWR. However as discussed above the FMEA analysis led Hitachi-GE to limit its focus to countermeasures against "sticking" and "galling". Hitachi-GE concluded that it was unnecessary to introduce diverse countermeasures for sticking and galling. Hitachi-GE has based its claim on evidence from Japan [87] that indicates that there are 427 safety relief valve units in operation in Japanese BWRs and that none of these have experienced a malfunction due to "sticking" or "galling".

327. Hitachi-GE has assumed that during a 'fail to open' fault, the valve stem will be retracted enabling steam pressure to force the disc off its seat or rupture the disc to discharge the pressure. However, based on the UK RGP and OPEX described above, I consider that it is reasonably foreseeable that the stem might remain in contact with the disc preventing it from lifting off its seat.

328. Hitachi-GE argues that designing a diverse safety relief valve potentially increases risk as the technology would be unproven compared to the available OPEX on its proposed designs. I consider that this is a valid argument as using a novel design of safety relief valve or alternative manufacturer could be less reliable.

329. ONR fault studies specialists examined Hitachi-GE's analysis that demonstrates the level of redundancy provided by the existing design of safety relief valves. Similarly, ONR Probabilistic Safety Analysis (PSA) specialists have examined Hitachi-GE's PSA and its contribution to the overall UK ABWR risk. I understand that the ONR specialists concluded that the number of safety relief valves for the UK ABWR offers adequate redundancy with partial diversity for their means of actuation.

330. I am satisfied with Hitachi-GE's arguments that to introduce a diverse safety relief valve with no supporting operating experience could increase the risk of 'failure to open' faults. I therefore consider it would be disproportionate to pursue an alternative valve design as the operating experience (30 years) does not indicate that sticking or galling will be an issue.

4.10.10 Examination, Inspection, Maintenance and Testing (EIMT) for the Safety Relief Valves

331. OPEX examined by Hitachi-GE identified UK failures associated with incorrect maintenance, unsuitable testing programs, and inadequate management of testing. Hitachi-GE claims Japanese BWRs have not identified failures and that this therefore demonstrates that Japanese administrative measures are satisfactory. I do not consider this appropriate evidence to support licensing of the UK ABWR. The hierarchy of protective measures underpinning UK legislation requires that engineered means is provided in preference to reliance on procedures. For example, it is reasonably foreseeable that differences in culture and licensee arrangements might undermine the OPEX from J-ABWR. Based on ONR SAPs EDR.1 to 4, design for reliability, I consider that a diverse design or diverse EIMT practices offer a means of protecting against common cause failure due to maintenance errors.

332. Hitachi-GE considered [87] administrative measures to safeguard against human errors during EIMT. For example, it suggested that different teams could perform EIMT and separate supervisors could then independently verify this. Hitachi-GE also considered safety relief valves manufactured at different times or different production lines could provide some diversity against CCF due to errors in the manufacturing phase.

333. Hitachi-GE has indicated that future licensees in consultation with manufacturers should consider the application of administrative arrangements in the site-specific phase of the project. I concur that such practices can provide diversity and should be the responsibility of the future licensee. However, the licensee may not fully understand the importance of adopting these procedures. I have therefore identified an assessment finding (AF_ABWR_ME_002) aimed at capturing these requirements, as detailed below.

4.10.11 Conclusions

334. Based on my mechanical engineering assessment, I do not consider that Hitachi-GE has provided adequate evidence during GDA to support its conclusion that sticking and galling are the only failure mechanisms within safety relief valves. However, I am satisfied their overall conclusion is appropriate.

335. I am satisfied that Hitachi-GE has presented sufficient justification based on OPEX from J-ABWR and other similar reactor plant that diversity of safety relief valve design

is not necessary. I have also considered the advice of ONR Fault Studies and PSA specialists in reaching this conclusion.

336. Hitachi-GE has provided examples of administrative arrangements for EIMT and for diverse manufacture for the safety relief valves, which might reduce the risk of common cause failure. I have raised an assessment finding to ensure that these aspects are be addressed by the licensee.

337. I am satisfied with Hitachi-GE's evidence that failure of the safety relief valve to close, leading to loss of containment, is bounded by more severe events. My understanding is that ONR fault studies inspectors are also content with this claim.

4.10.12 Regulatory Findings

338. The following assessment finding applies to the safety relief valves:

339. **AF_ABWR_ME_002**- It is established relevant good practice in UK reactor safety cases to demonstrate that at least two diverse structures, systems or components (SSCs) are provided to deliver the key nuclear safety functions required after a frequent design basis fault, so far as is reasonably practicable. In a small number of cases, Hitachi-GE has made adequate arguments that it would be grossly disproportionate to provide fully diverse and independent design provision. In these cases, Hitachi-GE's intention is for other types of diversity such as diverse manufacturing practices, and enhanced EIMT regimes to be considered during detail design. The licensee shall ensure that they consider these alternative methods of achieving diversity during detailed design and implement them wherever reasonably practicable.

4.11 Gloveboxes and Cabinets

4.11.1 Introduction

340. My assessment covers gloveboxes and cabinets whose safety function is to contain radioactive substances.

4.11.2 Overview of Hitachi-GE Safety Case

341. There is a limited requirement for this type of equipment within the UK ABWR. However, Hitachi-GE has stated that it is proposing to design a glovebox for inspecting and sorting radioactive waste in the Solid Waste Facility (SWF). Hitachi-GE has briefly covered the generic design of this glovebox in the Basis of Safety Case (BoSC) on Solid Waste Management Systems [91] .

342. Hitachi-GE has also indicated that it is considering installing containment cabinets for sampling tasks as part of the Liquid radioactive Waste Management System (LWMS), which is part of the radioactive waste building. Hitachi-GE provides a brief description of the generic design of these cabinets in the BoSC on Liquid Waste Processing in the Radioactive Waste Building [92].

4.11.3 Strategy for Assessment of Gloveboxes and Cabinets

343. The ONR TAG on containment in chemical plants [93] offers RGP on specific matters that assessors should consider. I consider that the solid and liquid waste facilities are effectively chemical plant and therefore, as part of my assessment, I used the guidance in this TAG as RGP to assess Hitachi-GE's design and reported my findings below.

344. The TAG states that design and construction standards applicable to radioactive containment should be clearly stated and justified for the particular application. I sampled the evidence for glovebox design based on BoSC for solid waste management facility [91] . I concluded that Hitachi-GE has not presented its detail design for gloveboxes within GDA although it has presented some high-level design claims for the SWF building. I am satisfied that this is sufficient detail at this stage.

345. In the absence of detailed design information, my assessment is limited to determining whether the glovebox design philosophy presented at GDA is adequate. In particular, I have considered whether the generic design be developed into a detailed design that offers an adequate containment barrier to protect workers and members of the public from the risks of a release of radioactive contamination.

4.11.4 Assessment of Design Features for Gloveboxes and Cabinets

346. The following features have been assessed against RGP set out in the ONR TAG on containment in chemical plants TAG [93]hereafter referred to as 'the TAG'

347. The TAG states that containment barriers should be capable of withstanding both internal and external hazards and retaining their duty for the life of the plant and into decommissioning. Hitachi-GE states that glovebox design life would be 60 years, to cover the reactor operational phase, plus an additional period to support decommissioning [92]. I am satisfied with this fundamental philosophy particularly as it is likely that the facility will be required to support decommissioning.

348. The fundamental design philosophy for gloveboxes is to ensure that the physical integrity of barriers is maintained at all times to meet the design and fault conditions. Hitachi-GE states [91] that its design of the structure for the solid waste facility building would withstand any reasonably foreseeable internal and external events. Hence, the building structure will protect gloveboxes so that they maintain their containment safety function. I am satisfied that this design philosophy should be adequate to contain a release of radioactivity within the solid waste facility should any internal events lead to glovebox damage.

349. The TAG states that future licensees should install suitable monitors to detect the presence of radioactive material following loss of containment. Hitachi-GE has indicated that it will design the solid waste facility to provide monitoring arrangements that alert operators of abnormally high activity in air resulting from foreseeable fault conditions. In addition, Hitachi-GE's claims that its design will include monitoring arrangements that alert operators of smoke/fire resulting from fault conditions. Such monitoring proposals should help to underpin foreseeable mechanical fault conditions and should I am satisfied that Hitachi-GE's monitoring proposals are adequate to satisfy UK expectations.

350. The TAG states that ventilation may form part of the containment strategy adopted for a particular process. Hitachi-GE has indicated that it will design the ventilation system in the glove box inspection area to provide adequate containment and filtration during normal operation, fault conditions and filter change tasks. Hitachi-GE proposes a cascaded airflow within the building to prevent the spread of radioactive contamination to other areas under these conditions. I am satisfied with this approach.

351. Hitachi-GE states [91] that design intent is to provide a ventilation system that is capable of providing a level of protection that is greater than that which the predicted operating level requires. Hence, the ventilation should ensure that that even if a glovebox failure were to occur, the building ventilation system is capable of minimising an offsite release of contamination.

352. The TAG states that the licensee's safety case should address EIMT. Hitachi-GE states [91] that this would include duplicate equipment in designated active areas, as required, with local storage provision.

353. Hitachi-GE states [91] that it will design crane control systems to reduce the risk of cranes and their loads, colliding with each other or structures within the solid waste facility. I consider that this to be a reasonable and appropriate means of protecting the gloveboxes from lifting operations.

4.11.5 Conclusions

354. Based on the above generic design features, I am satisfied that from a mechanical engineering perspective, the glovebox design and other measures proposed by Hitachi-GE for the UK ABWR should satisfy UK RGP.

4.11.6 Regulatory Findings and Shortfalls

355. None

4.12 Main Steam Isolation Valves

4.12.1 Introduction

356. The UK ABWR has four main steam lines that deliver high-pressure steam from the reactor to the turbine for the power generation under normal power generation mode.

357. Each main steam line is fitted with two main steam isolation valves, one located inside (inboard) of the containment barrier (the reinforced concrete containment vessel), and another one outside the containment barrier. The main steam isolation valves are required to close rapidly and automatically should an accident involving steam line rupture occur. Their safety function is to limit the loss of reactor coolant and release of radioactive material.

4.12.2 Strategy for Assessment of Main Steam Isolation Valves

358. At step 4, I used a Technical Support Contractor (TSC) to review a sample of the Hitachi-GE submissions for the main steam isolation valve. The objective of this was to review the adequacy of the evidence contained within the Basis of Safety Case [81] (BoSC) submissions and topic report [94] for the main steam isolation valve, from a mechanical engineering perspective. This built upon the work already carried out by the TSC at step 3, including determining whether the findings and observations identified at step 3 are adequately addressed at step 4.

359. The main documents considered by the TSC were the BoSC [81], the topic report [94] and the response to RQ-ABWR-0652 [95]. Hitachi-GE has presented its evidence supporting the safety case in references to these documents. The TSC reviewed a sample of the supporting references to assess the adequacy of this evidence at step 4. The technical support contractor listed the documents sampled in its report [84]. I oversaw the TSC's work through regular technical and progress meetings and by reviewing their final report [84]. I am therefore content that the review completed by the TSC meets my expectations.

4.12.3 Safety Functions for Main Steam Isolation Valves

360. In accordance with guidance in ONR SAP EKP.4 I am satisfied that Hitachi-GE has identified the safety function(s) to be delivered using a structured analysis. The Main

Steam Isolation Valves are deigned to perform the following principal Category A safety functions:

- To isolate the Main Steam lines to limit the release of reactor coolant and radioactive material to the surroundings in the event of a Main Steam pipe rupture. [NB SFC 4-7.2].
- To confine radioactive material within the pressure containment boundary and prevent dispersion during a fault. [NB SFC 4-7.3].
- To contain the reactor coolant during normal operations and form a pressure barrier during fault conditions, the destruction of which would result in a loss of reactor coolant of unmitigated radioactive consequences above the Basic Safety Limit. [NB-SFC-4-1.1]

361. Each of main steam isolation valve safety functions are categorised as Category A (i.e. they play a principal role in nuclear safety) and the main steam isolation valves are designed to meet Class 1 requirements (i.e. they form a principal means of fulfilling the category A safety function). Hitachi-GE's based this on their process for categorisation of safety functions and classification of SSCs [96]. I sampled the categorisation of the safety functions and classification of the Main Steam Isolation Valves and I am satisfied that it is appropriate and aligns with the guidance in ONR SAP ECS.1 and ECS.2.

362. I reviewed the safety function claims, arguments and evidence presented in the Basis of Safety Case [81] and the Topic Report for the Main Steam Isolation Valves [94] and consider them adequate for GDA Step 4. The evidence supporting my assessment is contained in supporting references that the Technical Support Contractor sampled. The Technical Support Contractor concluded that they are acceptable and adequate for Step 4 GDA although some documents will need to be revised and updated. Hitachi-GE has identified these and listed them in Section 7.9 of Reference [94].

363. Based on my review of the Technical Support Contractor's report, I am content that the evidence presented by Hitachi-GE is adequate for GDA Step 4. I recognise that the compilation of the evidence is ongoing.

4.12.4 OPEX for Main Steam Isolation Valves

364. The main steam isolation valves are Y-pattern globe valves that are a tested and proven technology. The design philosophy report [97] presents the philosophy behind the design of the main steam isolation valve and covers the aspects of safety functions, structures, materials and avoidance of through-life degradation.

365. Hitachi-GE selected this type of valve for reliability, supported by considerable OPEX from the large number of the same type valves used in nuclear power stations worldwide. Hitachi-GE states [94] that OPEX in Japan and worldwide has been taken into consideration in the design process. I have reviewed the OPEX report [98] and I am satisfied that issues relating to the reliability of the main steam isolation valves have been adequately considered by Hitachi-GE.

4.12.5 Main Steam Isolation Valve Diversity

366. Independent systems provide the control logic signal, pneumatic power (high-pressure nitrogen and compressed instrument air) and electrical power supplies to the two main steam isolation valves. These should provide common cause failure protection. However, the main steam isolation valves themselves do not provide common cause failure protection as they are of the same mechanical configuration and will be manufactured by the same supplier.

367. I queried the provision of diversity in the main steam isolation valves with Hitachi-GE [99]. It explained [100] that its process does not generally require diversity for Class 1 components unless there is a particular requirement from the PSA to improve reliability against common cause failure. Hitachi-GE explains this approach in its topic report on safety requirements [65]. Hitachi-GE has designed the main steam isolation valves to meet a reliability target that it considers conservative since operating experience (OPEX) indicates that similar isolation valves have much lower failure rates. I am satisfied with Hitachi-GE's response to my query on this topic.

4.12.6 Main Steam Isolation Valves Qualification

368. Qualification procedures provide a means of ensuring that the main steam isolation valves will perform their allocated safety functions in all normal operational, fault and accident conditions, identified in the safety case, for the duration of their operational lives. The specified qualification processes include production tests, safety functionality, dynamic seismic capability, thermal hydraulics performance and ageing and environmental capability. Hitachi-GE has developed the main steam isolation valve equipment qualification plan [101] to ensure that structural and safety functional requirements detailed in the design specification are achievable through the qualification tests and applied Quality Assurance (QA) procedures.

369. I raised a regulatory query [99] on the number of test cycles for the main steam isolation valves to qualify this component for a 60-year design life. Hitachi-GE's response [100] presents an updated number of test cycles, which it considers are sufficient to qualify the UK main steam isolation valve actuation. Hitachi-GE has updated the equipment qualification plan [101] to show these latest figures. I consider that Hitachi-GE has provided an adequate response to this RQ

370. I am satisfied that the main steam isolation valve equipment qualification plan includes adequate information for Step 4 GDA.

4.12.7 Main Steam Isolation Valves Examination Inspection Maintenance and Testing

371. Hitachi-GE describes EIMT of the Main Steam Isolation Valves in References [81] and [94]. Although full details are not available at this stage, I am satisfied that this can be developed by the licensee during detail design. I have captured this as an assumption (AS-ABWR-ME05).

4.12.8 Main Steam Isolation Valve Actuator Handling

372. The main steam isolation valve actuator, together with the valve head, weighs approximately 3 tonnes and is set at an angle of 45° (to vertical) into the valve body. I did not consider the original proposal, of using a manual chain block for extracting the valve, to be UK RGP as the lift is not vertical. Hitachi-GE has changed the removal method by using a jacking system for extracting the actuator and a trolley system with a monorail on the floor for transporting the valve internals to the maintenance area [102]. This revised approach was assessed and accepted by ONR at Reference [103].

373. I consider that Hitachi-GE has adequately considered reducing risks ALARP for extracting the main steam isolation valve internals for maintenance. It has also produced lifting schedules [104], risk assessments [72] and assessed the removal and transportation of main steam isolation valve [102]. The reports submitted provide confidence that Hitachi-GE is identifying risks and continuing to reduce them ALARP. I conclude that this is adequate for GDA step 4.

4.12.9 Main Steam Line Plugs

374. During outages, temporary line plugs are fitted into the main steam lines to isolate the main steam isolation valves from the reactor pressure vessel. This is to allow testing of the main steam isolation valves. These plugs prevent water leakage from the opened reactor pressure vessel into the main steam lines. During Step 4, ONR queried the design proposal for the main steam line plugs and asked Hitachi-GE to explain how the risks have been reduced SFAIRP.

375. Hitachi-GE describes the method of deploying and recovering the plug from the main steam line and the design of the sealing arrangement in its response to RQ-ABWR-0868 [105]. Hitachi-GE also explains the arrangements for lifting the main steam line plug into the reactor pressure vessel in the topic report on operating deck mechanical handling equipment [42]. I have based my assessment on the information in these documents, together with material presented at a technical workshop in January 2017 [106].

376. The design of the main steam line plug includes two diverse seals on a single plug. I consider that this is consistent with UK RGP [107] for a pipe isolation of this type. I am also satisfied with the arguments [105] for rejecting the alternative options of an additional isolation valve in the Main Steam Line or the use of freeze seals. While I am satisfied with the sealing arrangement, I queried several aspects relating to the deployment, removal and potential recovery of the main steam line plug in the event of failure of the deployment mechanism. During the January 2017 technical workshop, Hitachi-GE explained the procedure in detail and a number of potential fault scenarios were described [25]. I am content that Hitachi-GE's response to my queries has provided adequate evidence that the operators should be able to deploy and remove the main steam line plug safely.

377. The topic report [42] considers the consequences of a dropped load for two different designs of the main steam line plug lifting attachment. The selected option for the UK ABWR is based on a design in widespread use within the US fleet of Boiling Water Reactors. This involves inserting the four main steam line plugs sequentially in separate handling operations rather than in a single operation as in the J ABWR design. This design significantly reduces the mass of the lifting attachment and plug compared to the J-ABWR design, which reduces the damage caused by a dropped load. Hitachi-GE has recognised that the increased number of handling operations could increase the likelihood of human errors and it has recognised the need to complete further human factors engineering work during detailed design. Following discussion with a colleague from the human factors specialism, I judge that this position is acceptable at GDA.

4.12.10 Conclusions

378. I am satisfied that the information provided by Hitachi-GE for the Main Steam Isolation Valves is adequate for step 4 GDA. I assume that the licensee will consider EIMT requirements during detail design. I have captured this with assumption AS-ABWR-ME05.

379. Based on this information provided at GDA, I am satisfied that the proposed design for the main steam line plugs provides an ALARP solution that is supported by OPEX from the US fleet of BWRs and that Hitachi-GE has developed an adequate recovery procedure in the event of the device coming stuck. While it is foreseeable that there may be minor changes to the design of the Main Steam Line Plug and operating procedure required for UK ABWR, I am satisfied that these can be accommodated during detailed design.

4.12.11 Regulatory Assumptions

380. **AS-ABWR-ME 05** - ONR acknowledges that during GDA, Hitachi-GE has not sufficiently developed its design to enable the licensee to identify the appropriate EIMT requirements in line with UK expectations. ONR assumes that the licensee shall ensure that they consider EIMT requirements for all mechanical engineering items, which attract a safety classification. ONR also assumes that the licensee shall establish whether these requirements are directly driven by the safety case, are based on manufacturer recommendations or are based on plant operating experience (or appropriate combinations).

4.13 Emergency Core Cooling System

4.13.1 Introduction

381. The role of the Emergency Core Cooling System (ECCS) and the alternative systems for core cooling is to inject water into the Reactor Pressure Vessel (RPV) in the event of a reactor fault. Examples of faults requiring the use of the ECCS include loss of the main condenser, Loss of Coolant Accident (LOCA), Loss of Offsite Power (LOOP), etc. Hitachi-GE has designed the ECCS to prevent serious damage to the core fuel by suppressing zirconium water reactions should the any of the faults identified occur.

382. The ECCS comprises the following safety systems:

(1) **The Reactor Core Isolation Cooling (RCIC)** system provides core-cooling water to the RPV. This occurs when the reactor is in a high-pressure condition to compensate for water loss during transients and LOCA events.

(2) The **High Pressure Core Flooder (HPCF)** System provides core-cooling water supply to the RPV when the reactor is in a high pressure or low-pressure condition to compensate for water loss during transients and LOCA events.

(3) **The Low Pressure Flooder (LPFL)** Mode of the Residual Heat Removal (RHR) system provides core-cooling water supply to the RPV when the reactor is in low-pressure condition to compensate for water loss and remove decay heat in the event of LOCA.

(4) **The Automatic Depressurization (ADS)** system depressurises the RPV to allow operation of the LPFL.

383. A number of backup systems are provided, to perform similar functions in the event that the ECCS failed to perform its functions, as follows:

(5) The **Flooder System of Specific Safety Facility (FLSS)** provides core-cooling water supply to the RPV when the reactor is in low-pressure condition in the event of failure of the primary cooling means (i.e. RCIC, HPCF and LPFL).

(6) The Reactor Depressurization Control Facility (RDCF) depressurises the RPV to allow initiation of the FLSS.

384. I identified the following areas of interest for my assessment:

- Equipment qualification
- Operation Experience (OPEX)
- Claims Arguments and Evidence for Safety Functional Claims

- Safety Property Claims

4.13.2 Overview of Hitachi-GE Safety Case

385. Hitachi-GEs Basis of Safety Case (BoSC) [32] details the design requirements for the ECCS. The safety functional claims made against the ECCS are also present within the BoSC identifying category and class requirements.

386. The BoSC's were assessed by ONR's Technical Support Contractor (TSC) against ONR SAP's and RGP during GDA step 3. The TSC's findings identified shortfalls in the evidence presented by Hitachi-GE at that time. The TSC presented these shortfalls in a report issued via RQ-ABWR-0619. This RQ required Hitachi-GE to:

    1. familiarise itself with the report findings and observations,
    2. confirm the report factual accuracy;
    3. Hitachi-GE to respond to the TSC findings through the formal GDA RQ process for ONR assessment

387. Throughout step 4 ONR has engaged with Hitachi-GE through various level 4 engagements and I am satisfied that Hitachi-GE has adequately discussed the TSC's findings during these engagements. Through my assessment during step 4, I am satisfied that Hitachi-GE's final versions of the BoSC now provide sufficient evidence to satisfy the TSC's initial observations and findings.

388. Hitachi-GE has provided piping and instrumentation diagrams [108] detailing the design of ECCS functions. I am content the proposed design meets the system intent.

389. Hitachi-GE has provided claims arguments and evidence for categorisation and classification of the ECCS. I am satisfied that this aligns with the guidance given in NS-TAST-GD-094 Functions and Classification of Stuctures, Systems and Componentes [37].

390. The Pre-Construction Safety Report (PCSR) summarises the case presented in the BoSC. Topic reports present detailed analysis results including failure mode and effect analysis studies, design assessments and identified modifications to J-ABWR.

391. Hitachi-GE has produced hazard schedules to collate data from hazard studies and present the likelihood and consequence associated with each fault. These schedules identify protection system requirements (e.g. Category and classification).

392. I am satisfied that this approach demonstrates that Hitachi-GE has adequately considered the likely failure mechanisms, consequences of failure and design mitigation in line with the guidance provided in ONR SAP EKP.4 safety function..

4.13.3 Strategy for Assessment of Emergency Core Cooling System

393. The Reactor Core Isolation Cooling (RCIC) system has been a focus of ONR attention throughout GDA and therefore as part of my sampling strategy, I have therefore chosen to sample the RCIC function of the ECCS and specifically the RCIC pump. ONR has undertaken a number of interactions with Hitachi-GE relating to the RCIC system at all stages of GDA via level 4 contacts and Regulatory Queries (RQs).

394. The main documents sampled for the purposes of this assessment were the ECCS BoSC) [32] and equipment qualification plan for reactor core isolation cooling pump [109]. Other references used to inform my judgement include:

- General Design Process Approach for Mechanical Engineering SSCs [18]
- Guidance for SSC's Design Justification [110]
- Reactor Core Isolation Cooling (RCIC) System Piping and Instrumentation Diagram [108]
- Equipment Qualification for RCIC pump (Response to RQ-ABWR-0905,0920,0921) [111]
- Topic Report on Safety Requirements for ME SSC's [65]
- Topic Report on Reactor Core Isolation Cooling Pump Design Justification [112]

4.13.4 Equipment Description

395. The RCIC consists of a steam turbine driven pump, which performs coolant injection into the reactor core in the event the reactor pressure vessel is isolated and the water feed supply is unavailable. The RCIC pump injects water from the condensate storage tank or the suppression pool into the reactor pressure vessel to maintain water level above the active fuel in the event of a fault when the reactor is in high-pressure state and during reactor pressure vessel, depressurisation.

396. The reactor pressure vessel supplies turbine steam to the pump so it can inject coolant without electrical power supply when the core is in a high-pressure state. Coolant for injection comes from either the condensate storage tank or the suppression pool. A suction strainer on the suction line inside the suppression pool prevents pump clogging and injection of foreign substances into the reactor pressure vessel. The RCIC pump has individual piping, valves, instrumentation and controls.

4.13.5 Emergency Core Cooling System Qualification

397. During the GDA process, ONR raised the following RQs:

- RQ-ABWR-0905, Qualification Plans
- RQ-ABWR-0920, RCIC Start-up Test
- RQ-ABWR-0921, Reactor Core Isolation Cooling (RCIC) Pump

398. ONR raised several queries regarding the qualification of the RCIC pump and its alignment with relevant good practice (RGP). Qualification of the RCIC pump was discussed during a regulatory Level 4 workshop with Hitachi-GE and recorded in a contact record [113]. Hitachi-GE provided an equipment qualification plan for the RCIC Pump [111], in response to ONR queries. The responses set out a sufficient level of assurance to satisfy the RQs. I consider that the information provided addresses the guidance in ONR SAP EQU.1 design qualification procedures.

399. ONR SAP EQU.1, qualification procedures is concerned with confirmation that SSCs perform their allocated safety function(s) in all normal operational, fault and accident conditions. I consider that Hitachi-GE's approach achieves this as its design ensures it has the RCIC pump to enable operation during fault conditions such as loss of cooling. This is evident in the system description and fault schedule provided in the BoSC for ECCS) [32].

400. Hitachi-GE states [111] that the concept design for the RCIC pump has been qualified through usage on Pressurised Water Reactors (PWRs) and design for the Lungmen ABWR. Hitachi-GE claims the RCIC pump also complies with internationally recognised standards ASME QME-1, IEEE 323, IEEE344. It will be the responsibility of the licensee evidence of certification during the detailed design phase. However, identification of relevant codes and standards during GDA aligns with ONR SAP ECS.3 codes and standards.

401. Hitachi-GE plans to perform factory acceptance tests and site commissioning tests, simulating operating conditions as outlined in the equipment qualification plan for reactor core isolation cooling pump [109]. The equipment qualification plan identifies detailed design requirements of the pump for testing purposes. The topic report [112] also indicates the testing acceptance criteria, I am content with the information provided and consider that Hitachi-GE's approach should provide adequate qualification procedures for the RCIC pump and aligns with ONR SAP EMT.3 Type-testing.

402. ONR SAPs EAD.1 to EAD.5 describes the expectations for ageing and degradation, which should be evaluated and defined at the design stage as well as reviewing obsolescence of SSCs. The BoSC [32] indicates the life expectancy of the RCIC pump to be 60 years. Hitachi-GE has also identified, in the EIMT section, degradable parts (e.g. O-rings) are replaced at 5 years intervals or less dependent on inspection results.

403. I consider the qualification evidence provided by Hitachi-GE to be adequate for the purposes of GDA. The level of testing and certification is in line with regulatory expectations.

4.13.6 Emergency Core Cooling System Operational Experience

404. Hitachi-GE has identified, in its topic report [112], other nuclear power plants that use pumps of the same type as the proposed J-ABWR RCIC pump. The identification of previous use of the technology provides ONR with assurance of concept qualification in line with ONR SAP EQU.1 Qualification procedures. The previous use of the RCIC pumps in nuclear applications provides confidence in the validity of OPEX statements and design advancement.

405. The original design of the RCIC pump pre dates any current internationally recognised standards. Hitachi-GE states that the design it is proposing, has evolved adopting design standards such as ASME. I consider application of ASME standards combined with past OPEX meets RGP and aligns with ONR SAP ECS.5 use of experience, test or analysis.

406. Hitachi-GE states [111] the RCIC pump design has advanced throughout its use. This development reflects the manufacturer's desire to improve the design incorporating internal OPEX; it has not been in response to any incidents. The RCIC pump topic report [112] and equipment qualification plan [109] identify design changes applied by the manufacture of the pump via their internal processes.

407. The above design advancements provide evidence that Hitachi-GE has addressed and applied OPEX. This is in line with RGP and meets my expectations for the development of equipment design.

4.13.7 Emergency Core Cooling System Safety functional claims

408. The safety functional claims [32] delivered by the RCIC pump are as follows:

- The RCIC is the principal means to provide reactor core cooling as part of the ECCS. This occurs when the RPV is in high-pressure state and in the interval, it is being depressurised. It also provides protection against frequent faults such as loss of the normal feed water supply and infrequent faults such as Loss Of Cooling Accident (LOCA). This function is categorised as Category A (i.e. it plays a principal role in ensuring nuclear safety) and the components to deliver it are designed to meet Class 1 requirements (i.e. a component that forms a principal means of fulfilling a category A safety function).

- RCIC is capable of providing reactor core cooling during at least 8 hours in the event of Loss of Offsite Power Supply (LOOP) and loss of all the AC emergency power sources.

409. In consideration of ONR SAP EKP.4 I am satisfied that Hitachi-GE has identified the safety function(s) using a structured analysis.

410. Hitachi-GE has provided arguments and evidence to support these claims within the BoSC) [32]. An example of evidence provided for the safety functional claims within the piping and instrumentation diagrams [108] is the inclusion of a manual operational ball valve on turbine inlet. This provides manual operation capability on a loss of power fault, allowing the RCIC pump to operate by opening the inlet allowing steam to turn the pump turbine, therefore delivering its safety function. Hitachi-GE also addresses this within the BoSC) [32]. I reviewed the following documents and I am satisfied that Hitachi-GE has substantiated the claims and arguments made for GDA purposes:

- ECCS BoSC [32]
- Piping and instrumentation Diagrams [108]
- RCIC Topic Report [112]
- RCIC Design Description [114]

411. I consider the piping and instrumentation diagrams and evidence of component performance for UK ABWR to be in early stages of maturity. It is my expectation that during detailed design, piping and instrumentation diagrams will be brought in line with the descriptions given within the BoSC, topic report and UK relevant good practice. Detailed design should also provide evidence of component qualification following factory and site acceptance testing. I recognise that further detailed considerations for piping and instrumentation diagrams along with qualification evidence will occur during detailed design. My expectation is that responsibility for making and implementing adequate piping and instrumentation diagrams and qualification arrangements in respect of licence conditions will rest with the licensee. I have captured this expectation as assumptions (AS-ABWR-ME-01, AS-ABWR-ME-02)

4.13.8 Conclusions

412. I consider that Hitachi-GE has adopted a suitable design process that should enable it to satisfy the requirements of UK legislation and RGP.

413. Hitachi-GE has provided adequate evidence of RCIC pump concept qualification.

414. I consider Hitachi-GE has provided sufficient evidence to substantiate the safety claims and arguments made regarding the RCIC pump

415. I am satisfied that Hitachi-GE has considered qualification of the RCIC pump for a UK context in its safety case submission. I have identified two assumptions described below to ensure future design requirements are adequately considered.

4.13.9 Regulatory Assumptions

416. **AS-ABWR-ME-01** – ONR considers that piping and instrumentation diagrams presented during GDA are early drafts requiring further development. ONR assumes that the licensee shall develop these diagrams for all systems so that they are, suitable to facilitate transfer of piping and instrumentation details from the responsible designer

417. **AS-ABWR-ME-02** – ONR acknowledges that during GDA, Hitachi-GE had not sufficiently developed its design to enable the licensee to identify the appropriate design qualifications details. ONR assumes that the licensee shall establish detailed design substantiation, factory acceptance test information, and site acceptance test information for individual mechanical items and their associated systems, which are important to safety. ONR also assumes that the licensee shall generate appropriate evidence that equipment qualification is adequately specified for all mechanical items important to safety in accordance with UK expectations.

4.14 Containment Heat Removal Systems

4.14.1 Introduction

418. The role of the containment heat removal system is to remove decay heat from the primary containment vessel generated by the reactor core following a design basis shutdown. Removing this heat prevents excessive temperatures and pressures within the primary containment and helps to maintain long-term containment integrity.

419. The containment heat removal systems deliver four main safety functions as follows:

- Primary containment vessel cooling
- Residual heat removal
- Atmospheric control
- Filtered containment venting

420. I identified the following areas of interest for my assessment:

- Redundancy, independence and diversity considerations
- Limits and conditions
- Safety functional claims arguments and evidence

4.14.2 Overview of Hitachi-GE's safety case

421. Hitachi-GEs Basis of Safety Case (BoSC) [72] details the design requirements and safety functional claims for the containment heat removal system. Earlier versions of the BoSC were assessed by ONR's Technical Support Contractor (TSC) against ONR SAP's and RGP during GDA step 3. The TSC's findings identified shortfalls in the evidence presented by Hitachi-GE at that time. The TSC presented these shortfalls in a report that was issued via RQ-ABWR-0637. This RQ required Hitachi-GE to:

1. familiarise itself with the report findings and observations,
2. confirm the report factual accuracy;
3. Hitachi-GE to respond to the TSC findings through the formal GDA RQ process for ONR assessment

422. Throughout step 4, I have engaged with Hitachi-GE through various level 4 engagements. During these engagements, I have challenged Hitachi-GE on the shortfalls identified by the TSC and this has resulted in revised submissions of the BoSC. I am now satisfied that Hitachi-GE's revised BoSC provides sufficient evidence to address the TSC's initial observations and findings.

423. Hitachi-GE has produced hazard schedules to collate data from hazard studies and present the likelihood and consequence associated with each fault. These schedules identify protection system requirements (e.g. Category and classification).

424. The individual systems that perform the specified safety function for the containment heat removal also form part of other systems, performing additional safety functions. These have been included in separate BoSC's that provide detail on EIMT and qualification procedures. Therefore, I consider EIMT and qualification for these systems to be out of scope for the purposes of assessment of the containment heat removal system.

425. I am satisfied that this approach demonstrates that Hitachi-GE has adequately considered the likely failure mechanisms, consequences of failure and provides design mitigation. This meets my expectations for ONR SAP EKP.4 safety function use of PSA.

4.14.3 Strategy for Assessment of Containment Heat Removal Systems

426. The main document sampled for the purposes of this assessment was the BoSC for the containment heat removal systems [72]. I have chosen to omit individual system BoSC documents as they contain safety functions out of scope for containment heat removal purposes. Other references used to inform my judgement are:

- Residual Heat Removal System piping and instrumentation diagram [115]
- Residual Heat Removal System Design Description [116]
- Atmospheric Control System Design Description [117]
- Atmospheric Control System piping and instrumentation diagram [118]
- Filtered Containment Venting System Design Description [119]
- Filtered Containment Venting System piping and instrumentation diagram [120]

4.14.4 Equipment Description

427. The containment heat removal system consists of three independent sub systems described below. Each sub system has safety functions to inject water into the Reactor Pressure Vessel (RPV) and provide long-term heat removal from the RPV or the Primary Containment Vessel (PCV). The necessary piping, valves, pumps and heat exchangers are included in each sub system.

Residual Heat removal system

428. The BoSC for containment heat removal system [72] makes safety functional claims on the residual heat removal system. The residual heat removal system delivers the following safety functions for the purposes of containment heat removal:

- **Suppression Pool Cooling**: delivers long-term containment heat removal following 'frequent' faults such as main condenser unavailability and infrequent faults such as anticipated transient without an emergency reactor shut down. It also delivers long-term containment heat removal upon RHR recovery following venting during 'infrequent' faults such as station blackout.

- **Low-pressure flooder mode**: delivers long-term containment heat removal following 'frequent' faults such as main condenser unavailability and infrequent faults such as loss of coolant accident (LOCA).

- **Primary containment vessel Spray Cooling mode**: contributes to suppressing the pressure containment vessel atmosphere and remove fission products from the containment atmosphere following a LOCA.

Atmospheric Control System

429. In the event of design basis faults, which lead to an increase of the pressure inside the primary containment vessel, the atmospheric control system provides overpressure protection for the primary containment vessel by venting through the hardened ventilation line. This directly connects to the main stack via the Stand by gas treatment system.

430. Hitachi-GE, in its BoSC for containment heat removal [72] identifies safety functional claims on the atmospheric control system. The atmospheric control system delivers the following significant safety functions for the purposes of containment heat removal:

- The atmospheric control system is a secondary means to deliver long-term primary containment vessel heat removal and overpressure protection. It is called upon during 'frequent' faults were the primary long-term containment heat removal means (residual heat removal system) has failed.

Filtered Containment Venting System

431. The filtered containment venting system releases primary containment vessel gas to deliver overpressure protection and long-term heat removal during design basis faults, beyond design basis faults or severe accidents. Filters reduce the amount of radioactive iodine and long-half-life fission products, contained in venting gas, in the event of severe accidents.

432. Hitachi-GE, in its BoSC for containment heat removal [72] makes safety functional claims on the filtered containment venting system. The filtered containment venting system delivers the following significant safety function for the purposes of containment heat removal:

- The filtered containment venting system is a secondary means to deliver long-term primary containment vessel heat removal and overpressure protection. It is called upon during 'frequent' faults where the primary long-term containment heat removal system (i.e. residual heat removal) has failed.

4.14.5 Assessment of Containment Heat Removal Systems Redundancy, Diversity & Segregation

Redundancy

433. The residual heat removal system consists of three redundant sub systems with their respective pumps, heat exchangers, piping, valves and instrumentation. Hitachi-GE has designed them so that single failure of components does not prevent the delivery of the containment heat removal safety function. The residual heat removal system piping and instrumentation diagram [115] provides adequate evidence of redundancy

434. Atmospheric control and filtered containment venting are two separate systems with individual components (piping and valves for containment venting) each one having sufficient venting capacity. I am content that these systems provide a level of redundancy in accordance with Hitachi-GE's guidance set out in the topic report on mechanical structures systems and components architecture [121].

435. I consider the application of redundancy described within the BoSC and the designs outlined in the piping and instrumentation diagrams addresses the redundancy aspects of SAP EDR.2 redundancy, diversity and segregation.

Diversity

436. Hitachi-GE state within its BoSC [72] that diversity is taken in to account with the design of alternative systems for long-term heat removal (Residual heat removal system, atmospheric control system and filtered containment venting system). These three systems are all capable of providing the stated safety functions within the BoSC. This meets my expectations and aligns with the diversity expectations of SAP EDR.2 redundancy, diversity and segregation.

Segregation

437. The residual heat removal system has three redundant sub systems segregated by location to prevent a failure in one sub system leading to a failure in another system. I consider that Hitachi-GE has configured the interconnections adequately to maintain segregation SFAIRP. To prevent a fault from a support system affecting the ability of a sub system to deliver its safety function, Hitachi-GE has designed the system to include three redundant, functionally independent and segregated sub divisions:

- Control and instrumentation
- Power supply
- Heating ventilation and air conditioning
- Component cooling

438. Segregation of the systems is evident in the design description for the residual heat removal system [116] supported by the piping and instrumentation diagrams [115]. Therefore, I am satisfied that the design provides adequate functional independence and segregation.

439. The atmospheric control system and the filtered containment venting system are two segregated systems with independent components. The support systems used for each system are also functionally independent.

440. I consider the evidence provided with in the BoSC, design descriptions supported by the piping and instrumentation diagrams provides adequate segregation, satisfying SAP EDR.2 redundancy, diversity and segregation.

4.14.6 Assessment of Limits and conditions for Containment Heat Removal Systems

441. Hitachi-GE in its containment heat removal system BoSC [72], adequately describes the limits and conditions given. Hitachi-GE has set out applicability modes, conditions, required actions, completion time and surveillance requirements. I am content with the structuring of the given limits and conditions which align with SAP EKP.3 on defence in depth.

442. I consider the limits and conditions stated in the BoSC [72] to be of limited scope. In particular, Hitachi-GE has not included details on primary containment temperature monitoring, containment heat removal, testing of systems, start up requirements and valve actuation requirements. I consider the limits and conditions provided to provide insufficient detail that is not in line with my expectations. I acknowledge that further detailed considerations for limits and conditions will occur during detailed design. My expectation is that responsibility for making and implementing adequate limits and conditions arrangements in respect of licence conditions will rest with the licensee. I have identified an assumption that should be considered by a future licensee (AS-ABWR-ME-03)

4.14.7 Safety Functional Claims, Arguments & Evidence for Containment Heat Removal Systems

443. Hitachi-GE has provided arguments and evidence to support the claims made within the BoSC [122]. I have assessed the evidence provided for the containment heat removal system that demonstrates the design is capable of providing long-term heat removal in the event of a design basis fault. I consider the design provides an adequate over view of the design intent and required safety functions. The piping and instrumentation diagrams also meet my expectations in terms of the supporting the evidence to deliver the required safety function. I am content the following documents substantiate the claims and arguments made for GDA purposes:

- Containment heat removal systems BoSC [72]
- Design descriptions for all systems [116] [117] [119]
- Piping and instrumentation diagrams for all systems [115] [118] [120]

444. In consideration of ONR SAP EKP.4 safety function, I am satisfied that Hitachi-GE has identified the safety function(s) using a structured analysis. The safety functional claims given in the BoSC [122] align with that of the system design descriptions [116] [117] [119]. This aligns with ONR SAP.EKP.2, fault tolerance.

4.14.8 Conclusions

445. Hitachi-GE has provided adequate evidence that Hitachi-GE have applied suitable redundancy, diversity and independence.

446. I consider Hitachi-GE has provided sufficient evidence to substantiate its safety claims and arguments.

447. I am satisfied that Hitachi-GE has considered limits and conditions in its safety case submission. I have identified an assumption captured through an assessment finding that detailed design adequately considers limits and conditions requirements.

4.14.9 Regulatory Findings and Regulatory shortfalls

448. **AS-ABWR-ME-03** - ONR acknowledges that during GDA, Hitachi-GE had not sufficiently developed its design to enable the licensee to identify the appropriate plant limits and conditions. ONR assumes that the licensee shall identify plant limits and conditions, from the safety case, covering all mechanical engineering equipment

important to safety. ONR also assumes that the licensee shall generate sufficient safety case information to satisfy the requirements of LC 23, and specifically it shall establish a suitable interface for transferring this information from the responsible designer.

4.15 Standby Gas Treatment System (SGTS)

4.15.1 Introduction

449. The role of the standby gas treatment system (SGTS) is to maintain the secondary containment areas under a negative pressure and remove radioactive particulate.

4.15.2 Overview of Hitachi-GEs safety case

450. Hitachi-GEs Basis of Safety Case (BoSC) [123] details design requirements for the SGTS. Hitachi-GE has made the following safety functional claim:

- The STGS delivers a supportive and secondary function for containment of radioactive materials in the event of a HVAC failure, a LOCA or a Fuel Handling Accident (FHA).

451. Earlier versions of the BoSC were assessed by ONR's Technical Support Contractor (TSC) against ONR SAP's and RGP during GDA step 3. The TSC's findings identified shortfalls in the evidence presented by Hitachi-GE in the earlier versions of the BoSC. The TSC presented these shortfalls in a report that ONR issued via RQ-ABWR-0614. This RQ required Hitachi-GE to:

   1. familiarise itself with the report findings and observations,
   2. confirm the report factual accuracy;
   3. Hitachi-GE to respond to the TSC findings through the formal GDA RQ process for ONR assessment

452. Throughout step 4, I have engaged with Hitachi-GE through various level 4 engagements. During these engagements, I challenged Hitachi-GE on the shortfalls identified by the TSC and this has resulted in revised submissions of the BoSC. I am now satisfied that Hitachi-GE's revised BoSC provides sufficient evidence to address the TSC's initial observations and findings.

453. Hitachi-GE has provided piping and instrumentation diagrams [124] detailing the design of standby gas treatment system functions. I am content that these diagrams support the evidence that these designs meet the description in the BoSC [125].

454. Hitachi-GE has provided claims arguments and evidence for categorisation and classification of the standby gas treatment system functions. I am satisfied that Hitachi has followed the guidance given in NS-TAST-GD-094 Functions and Classification of Structures, Systems and Components [37].

455. Hitachi-GE has produced a HAZOP report on the standby gas treatment system detailing potential hazards and consequences associated with each fault. These schedules identify protection system requirements. I am satisfied that this approach demonstrates that Hitachi-GE has adequately considered the likely failure mechanisms, consequences of failure and design mitigation in accordance with guidance in ONR SAP FA.14 use of PSA.

4.15.3 Strategy for Assessment of Standby Gas Treatment System

456. My assessment sample is based on:

- Design process
- Examination, Inspection, Maintenance & Testing (EIMT)
- Claims, Arguments and Evidence for safety functional claims

457. The main document sampled for the purposes of this assessment was the basis of safety case for the standby gas treatment system [123]. Other documents used to inform my judgement are:

- Standby gas treatment system piping and instrumentation diagram [124]
- Standby gas treatment system HAZOP report [126]
- Standby gas treatment system design description [125]

4.15.4 System Description

458. The SGTS consists of two trains with a fan and a filter in each train as well as the necessary piping, valves, instruments and controllers. The STGS is depicted below in Figure 2.

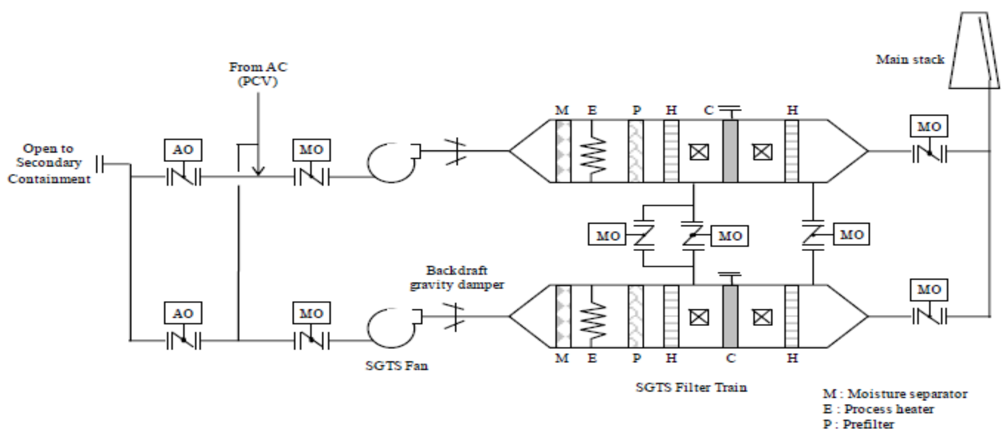

Figure 2 Standby gas treatment system

459. SGTS Filter Train consists of the following components (per one SGTS filter train unit):

- Moisture separator 1 set
- Heating coil 1 set
- Pre filter 1 set (2 units/set)
- High Efficiency Particle Air Filter (HEPA) 2 sets (2 units/set)
- Charcoal filter 1 set
- Space heater 1 set (2 units/set)
- Piping and valves: 1 set
- Instruments and control system: 1 set
- Control panel: 1 set

4.15.5 Design process for Standby Gas Treatment System

460. Hitachi-GE's design process for the STGS is in accordance with its "General design process approach for mechanical SSCs" [18]. I am content that Hitachi-GE has applied its design principles. The BoSC identifies the requirements for redundancy and segregation based on the categorisation of the safety function. I am content that the applied design principles align with SAP EDR.2 redundancy, diversity and segregation.

461. Hitachi-GE undertook a HAZOP process [37], involving independent consultants, to identify areas of concern and provide recommendations regarding the design of the STGS. I sampled these recommendations and identified that Hitachi-GE had given them due consideration and implemented design changes. Hitachi-GE has incorporated temperature monitoring and trips of the charcoal filter along with flow transmitters downstream of the filter trains as a means of indicating flow capacity. I am content that Hitachi-GE has used its HAZOP process to identify such safety measures. This aligns with ONR SAP EKP.5 safety measures.

462. I have assessed the BoSC for the inclusion of codes and standards. I am content that relevant codes and standards have been applied satisfying guidance in SAP ECS.3 codes and standards. Hitachi-GE has also identified ALARP procedures within the BoSC, which meet my expectations.

4.15.6 Examination, Inspection, Maintenance & Testing (EIMT) for Standby Gas Treatment System

463. Hitachi-GE claims that it has designed the STGS with EIMT capability during power operation and/or refuelling outage. This capability is to ensure the system safety functions are delivered throughout the systems operational life.

464. The BoSC [123] sets out surveillance testing requirements and acceptance criteria to ensure delivery of safety functions. The BoSC provides an adequate level of detail for GDA purposes on EIMT of the STGS components. This detail is largely dependent on manufactures maintenance standards. This is in line with my expectations for EIMT. Hitachi-GE claim that those components not designed for a 60-year design life will be replaced as necessary to maintain delivery of the safety functions. I consider the EIMT requirements identified by Hitachi-GE align with SAP EMT.1 identification of requirements.

465. I recognise that further detailed consideration of EIMT will occur during detailed design. My expectation is that responsibility for making and implementing adequate EIMT arrangements in respect of licence conditions will rest with the licensee. I have identified an assumption that should be considered by a future licensee (AS-ABWR-ME-05).

4.15.7 Claims, Arguments and Evidence for safety functional claims for Standby Gas Treatment System

466. In consideration of ONR SAP EKP.4 safety functions, I am satisfied that Hitachi-GE has identified the safety function(s) to be delivered using a structured analysis. The safety functional claims given in the BoSC [123] align with that of the system design description [125].

467. Hitachi-GE has provided arguments and evidence to support the claims made within the BoSC [123]. I consider that the evidence provided for safety functional claims is adequate to substantiate the claims given. I have assessed the calculations provided to determine the volume of airflow required, I consider this demonstrates the design specifications are capable of delivering the safety function. Piping and instrumentation diagrams also adequately demonstrate that the system design should deliver the required safety functions.

468. I am content the following documents substantiate the claims and arguments made for GDA purposes:

- Standby gas treatment system basis of safety case [123]
- Standby gas treatment system piping and instrumentation diagram [124]
- Standby gas treatment system design description [125]

4.15.8 Conclusions

469. Hitachi-GE has provided adequate evidence that it has applied suitable design processes.

470. I am satisfied that Hitachi-GE has considered EIMT requirements in its safety case submission. I have identified an assumption, captured through a generic ONR assessment finding, which assumes that detailed design adequately consider EIMT requirements.

471. I consider Hitachi-GE has provided sufficient evidence to substantiate the claims and arguments made regarding the safety functions of the standby gas treatment system.

4.15.9 Regulatory Findings

472. **AS-ABWR-ME 05** - ONR acknowledges that during GDA, Hitachi-GE has not sufficiently developed its design to enable the licensee to identify the appropriate EIMT requirements in line with UK expectations. ONR assumes that the licensee shall ensure that they consider EIMT requirements for all mechanical engineering items, which attract a safety classification. ONR also assumes that the licensee shall establish whether these requirements are directly driven by the safety case, are based on manufacturer recommendations or are based on plant operating experience (or appropriate combinations).

4.16 Emergency Power Supply

4.16.1 Introduction

473. Redundant and independent electrical power supply systems are an important safety requirement for nuclear power plants. The predominant means of providing this emergency (standby) electrical power is the use of onsite diesel driven generators. Their role is to supply power to essential safety systems needed for safe shut down of the reactor following a Loss of Off-site Power (LOOP) event.

474. ONR's Safety Assessment Principles (SAPs) set out ONR's assessment guidance for nuclear power plants in relation to emergency (standby) electrical power generation.

4.16.2 Overview of Hitachi-GE Safety Case for Emergency Power Supply

475. Hitachi-GE's Basis of Safety Case (BoSC) [127] details the design requirements for an emergency power supply. From a mechanical engineering perspective, the key system design requirements in the event of a LOOP are to:

- Supply the rated power and capacity within a prescribed time;
- Provide an uninterrupted supply of power for seven days;

- Provide reliable supply and management of essential services for the diesel engines (i.e. fuel, coolant, lubricating oil and compressed air).

476. The Pre-Construction Safety Report (PCSR) summarises the case presented in the BoSC. The topic report [128] outlines a high-level Failure Mode and Effects Analyses (FMEA) and hazard schedule for the emergency power supply system. The hazard schedule collates hazards identified from the FMEA and presents the likelihood, consequence of each fault and protection system requirements (e.g. safety category and classification).

477. The safety case provides claims that the emergency power supply system will be capable of reducing risk So Far As Is Reasonably Practicable (SFAIRP). Detailed design will commence after GDA with support from UK consultants and diesel generator manufacturers to qualify equipment to a level commensurate with its category and classification.

478. I am satisfied that Hitachi-GE has adequately considered the likely failure mechanisms, consequences of failure and design mitigation which aligns with ONR SAP EKP.4 safety function., use of probabilistic safety assessment.

4.16.3 Strategy for Assessment of Emergency Power Supply

479. ONR has undertaken a number of interactions with Hitachi-GE relating to the emergency power supply at all stages of GDA via Level 4 meetings, Regulatory Queries (RQs) and Regulatory Observations (ROs).

480. Two key documents sampled from a mechanical engineering perspective were Hitachi-GE's BoSC on emergency power supply system [127] and a topic report on design justification for the emergency diesel generator [128]. I also assessed relevant system description documents, references [129], [130] and [131] to inform my judgment. In addition, ONR's Technical Support Contractor (TSC) assessed the version of the BoSC, submitted at GDA Step 3, against ONR SAP's and RGP. The TSC's findings identified shortfalls in the evidence presented by Hitachi-GE. The TSC presented these shortfalls in their report issued via RQ-ABWR-0638 [132]. This RQ required Hitachi-GE to:

- familiarise itself with the report findings and observations;
- confirm the report factual accuracy;
- prepare to discuss the findings, observations and expectations in detail as part of the planned mechanical Engineering technical workshops;
- in advance of the planned technical workshops, develop and advise its strategy to address the findings, observations and expectations.

481. Throughout GDA Step 4 ONR has engaged with Hitachi-GE through various level 4 engagements and I am satisfied that Hitachi-GE has adequately discussed the TSC's findings during these engagements. Through my assessment during step 4, I am satisfied that Hitachi-GE's final versions of the BoSC now provide sufficient evidence to satisfy the key TSC observations and findings.

4.16.4 Equipment Description

482. The emergency power supply system consists of;

- Three independent Emergency Diesel Generators (EDGs). The EDGs safety function is Category A (any function that plays a principal role in ensuring nuclear safety) and the components are Class 1 (any SSC that forms a principal means of fulfilling a Category A safety function). The EDGs provide the first line of protection against short, medium and long-term (i.e. two hours, one day and seven days respectively) LOOP events.

- Two independent Back-up Building Generators (BBGs). The BBGs safety function is Category A and the design of systems, structures and components that deliver it meet Class 2 requirements. The BBGs are the primary line of protection against infrequent design basis LOOP events as a means of protection against Common Cause Failure (CCF) of the EDGs.

- One Diverse Additional Generator (DAG). The DAGs safety function is category B and the design of systems, structures and components to deliver it meet Class 3 requirements. The DAG is a tertiary line of defence providing power to essential safety systems to shut down the reactor in the event of a LOOP coupled with failure of both the EDGs and BBGs. Hitachi-GE claims that the design of the DAG will be diverse (e.g. the prime mover will be a gas turbine), in order to minimise the potential for CCF. However, Hitachi-GE has not specified the final design configuration of the DAG within GDA.

483. The EDGs are continuously primed in stand-by mode so that they can be operational on demand, in the event of a LOOP. Continuous pre-lubrication and pre-heating facilitate rapid start-up of the engine. A low voltage signal or detection of a loss of coolant accident automatically starts the engines. Three EDGs are sent start signals which attempts to start them in series, such that failure of one EDG leads to initiation of the next. Failure to start all three leads to initiation of the first BBG.

484. Each of the EDG and BBG sub-systems consists of the following main components;

(a) Engine - The EDG/BBGs have engines that supply motive power to the generator. The engines are designed to be automatically started by compressed air upon detection of a fault condition and therefore do not require external power. Operators can use the main control panel or local control panel to remotely start or shut down the engine manually.

(b) Generator – The generator provides the electricity needed for safe shut down of the reactor in the event of a LOOP. A synchronous generator is directly coupled to the engine.

(c) Fuel Oil System - Each EDG division has its own independent fuel oil system designed to supply sufficient fuel to the engine for seven days of continuous operation at rated power. The system consists of fuel oil transfer pumps, tanks and fuel oil transfer lines all designed to meet Class 1 safety requirements. Each EDG division has 1 day tank holding enough fuel for eight hours of engine operation. A main storage tank supplies fuel to the day tank when the fuel oil level falls below a specified level. There are six main storage tanks, two per division, providing redundancy.

(d) Cooling Water System – During operation the engine is cooled by the cooling water system and in stand-by mode, the engine is pre-heated via the pre-heater and the pre-heating pump to shorten the start-up time. Each generator has its own independent cooling water system with automated coolant supply. Each system consists of an expansion tank, heat exchanger, pre-heater, level switches for automatic supply control, pumps, piping and valves.

(e) Lubricant Oil System - The purpose of the lubricant oil system is to lubricate and cool the engine. Each engine has its own independent lubricant oil system that includes a cooling unit, heat exchanger and pre-lubricating pump, piping and valves. Hitachi-GE has designed the lubricating capacity of the system to be sufficient for continuous operation for seven days at rated power.

(f) Compressed Air System - The purpose of the compressed air system is to start the engine without external electrical power. Each compressed air system has enough air storage capacity for five starts.

(f) Air Intake and Exhaust Gas System – This system provides fresh air to the engine and removes exhaust products to outside the generator buildings. The exhaust gas also drives the turbocharger to increase the air intake pressure.

4.16.5 Assessment of Hitachi-GEs emergency power supply mechanical engineering arrangements

485. **Examination, Inspection, Maintenance and Test** (EIMT): The EDG BoSC sets out high-level EIMT proposals and states that the licensee will incorporate detailed EIMT schedules into the operation and maintenance manuals and plant maintenance instructions. However, as the safety case has yet to define specific requirements for EIMT I consider this a shortfall. ONR SAP EMT.2 states SSCs should receive regular and systematic examination, inspection, maintenance and testing as defined in the safety case. I have identified an assumption that should be considered by a future licensee (AS-ABWR-ME05).

486. Equipment Qualification (EQ): ONR SAP EQU.1 states that design qualification procedures should be applied to confirm that SSCs perform their allocated safety function(s) in all normal operational, fault and accident conditions. During Step 3 GDA ONR raised RO-ABWR-0051 – Mechanical Engineering SSC Qualification. This was to ensure that the UK ABWR SSC qualification arrangements were adequately developed to demonstrate and substantiate the SSC's design basis through life. Hitachi-GE has demonstrated that it has high-level arrangements in place to develop EQ plans to qualify its SSCs. However, shortfalls exist in the sample EQ plans developed in GDA to facilitate closure of RO-ABWR-0051. The EQ plans submitted in GDA were more aligned to factory acceptance test plans. It my expectation that EQ arrangements are further developed to include but not limited to;

- statistical sampling of SSC components to provide confidence in the robustness of the design over its intended operational life. Less reliance on single component single test;
- sequential or synergistic testing to verify SSC performance following interrelated or sequential fault conditions;
- consideration of sub-assembly and piece part component qualification not just system level qualification;
- test methods, test specifications, codes and standards should be clearly defined;
- qualification against UK specific operational, environmental, fault and accident conditions. Historic J-ABWR test data should not be assumed to be bounding;
- the use of J-ABWR OPEX (currently only 17 years) to underpin the UK ABWR 60 year qualified life should be justified. OPEX evidence should be complementary to physical test data and more reliance placed on accelerated ageing;
- consideration of ONR SAPs EAD.1 to EAD.5 - ageing and degradation.

The above requirements are captured by the following assessment findings:

- Equipment Qualification: GDA Assessment finding **AF_ABWR_ME_001**.
- Aging and degradation: GDA mechanical engineering **AF-ABWR-ME-07.**

4.16.6 System safety functions for emergency power supply

487. The key system safety functions claimed by Hitachi-GE are as follows:

- The EDG and BBG will supply power to the essential safety systems necessary for safe shut down of the reactor in the event of Loss of Off-site Power (LOOP). The EDG and BBG must develop the necessary supply frequency and voltage within a prescribed time from receiving an automatic or manual starting signal to achieve this.
- The EDG and BBG are capable of continuous operation for seven days at rated power.

488. Hitachi-GE has assigned the emergency power supply systems with safety categorisations, which align with ONR SAP ECS.2, safety classification. Hitachi-GE is proposing a probability of failure on demand (pfd) of 10^{-4} for the Class 1 EDGs and 10^{-3} pfd for the Class 2 BBGs. I consider this to align with the ONR TAG on categorisation and classification [37] that sets out target reliability figures for Class 1 and 2 SSCs. Detail design of the diesel generators will be necessary to substantiate these claims but I consider that identifying the appropriate safety classification for SSCs should enable the future licensee to undertake detailed design, manufacture and procurement of SSCs to the required safety classification.

4.16.7 Comparison of UK ABWR with J-ABWR reference design

489. Hitachi-GE has not detail designed the emergency power supply during GDA. Hitachi-GE proposes to communicate a number of design changes to the licensee during detailed design. Hitachi-GE has identified these through analysis of OPEX and changes to legislation since the conception of J-ABWR. For example;

- In the Fukushima Dai-ichi accident, all EDGs lost functionality due to water ingress from the tsunami. This highlighted the necessity of enhanced diversity and segregation to avoid the effects of common cause failure (ONR SAP EDR.2). The design basis for UK ABWR is that the EDGs and BBGs are located in different areas of the site and at different elevations. Control power sources for both systems are also separated and independent.

490. Hitachi-GE is proposing to introduce the Diverse Additional Generator (DAG). Although not confirmed in GDA, the proposed design will be diverse from the EDGs as it has a gas turbine as its prime mover, is protected against water ingress and is air (not water) cooled.

491. Hitachi-GE has produced a report [98] which relates to OPEX and learning from EDG operations worldwide, which identifies potential failure modes. I am satisfied that Hitachi-GE has identified appropriate OPEX that will inform the detailed design for UK ABWR EDGs.

4.16.8 Assessment of the Emergency Power Supply system.

492. As part of my assessment, I sampled the system design calculations presented in [129]. The example I chose is the capacity of the EDG fuel tank.

493. Hitachi-GE has proposed a capacity for the EDG fuel tanks, which should provide sufficient fuel for maintenance testing and seven days continuous operation during a LOOP event together with a suitable ullage space.

494. I could not identify the source of some fundamental input values used in the calculations. For example, Hitachi-GE claims that the fuel consumption rate of the EDG is 240g/kWh but there is no justification or identified origin for this figure. It is not clear if this value is taken from the J-ABWR reference design, or takes into account the uprated power and capacity of the UK ABWR design (5300kW to 7600kW and 6250kVA to 9000kVA respectively). In this instance, the implications of using provisional data maybe that a larger tank capacity is required for the UK ABWR. This may affect the overall size of the EDG building and site layout provision, which is considered in ONR SAP ELO.1.

495. At the fifth mechanical engineering technical workshop [133] Hitachi-GE accepted that EDG performance claims must be underpinned by objective evidence. The detail provided above is a single example of where the safety case does not yet meet this expectation. I am satisfied that this detail can be addressed by the licensee during detail design to demonstrate how performance claims have been appropriately applied for UK ABWR EDG design.

4.16.9 Diesel generator reliability and the use of biofuel.

496. I sought evidence of how Hitachi-GE has considered the amendment to the Motor Fuel (Composition and Content) Regulations 1999. The amendment is concerned with implementing stringent control of fuel parameters, which have an environmental impact. One means of complying with the regulations is the use of biofuels.

497. Bio diesel degrades over a relatively short time and can cause reliability problems with fuel systems. This has the potential for common cause failure of all EDG and BBG trains. Furthermore, the choice of fuel may have an impact on the accuracy of the EDG and BBG fuel tank capacity calculations as different fuels will have different densities requiring different capacity tanks.

498. I have assumed that the licensee will consider the requirements for handling and storage of fuel. This is captured under GDA mechanical engineering assumption (AS-ABWR-ME07).

4.16.10 Start-up time claims for the diesel generators.

499. I queried the design basis and safety margins associated with the prescribed start up time for the diesel generators and Hitachi-GE's response is provided in [134]. The safety analysis for the Emergency Core Cooling System (ECCS) compares the delay time from receiving the LOOP signal to the start of the ECCS injecting coolant into the core. The start-up time of the emergency diesel generator (EDG) meets the safety requirement for the ECCS to deliver its safety function. The diagram below shows the ECCS start up sequence with the demands placed on the EDG in blue.

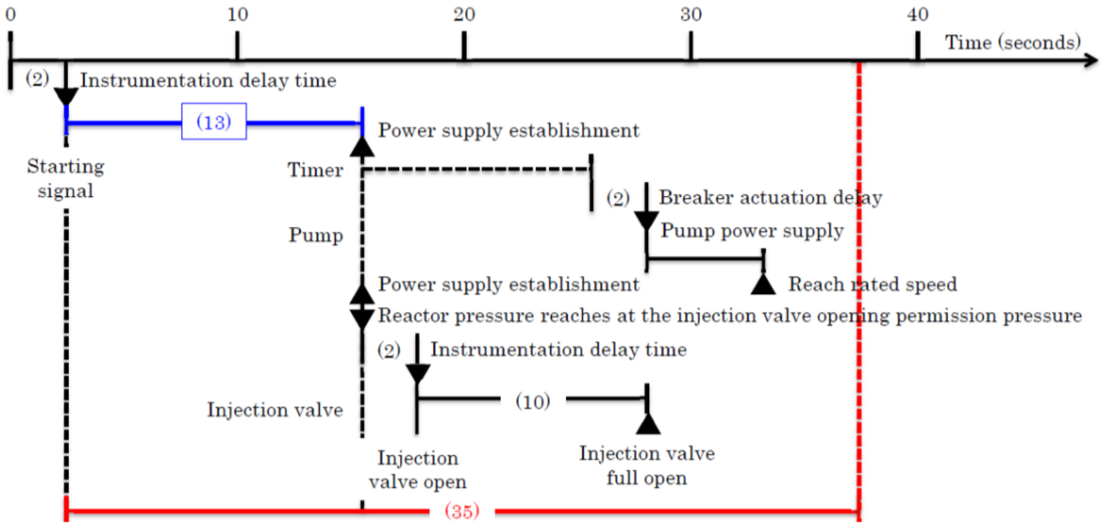

Figure 3 ECCS Start up Sequence

500. I consider the above to provide adequate justification for the EDG start up time.

4.16.11 Emergency power supply system availability during post power generation lifecycle phases

501. Hitachi-GEs emergency power supply safety case makes no claim on the availability during the post power generation life cycle phases. ONR SAP SC.3 states that for each lifecycle stage, control of the hazard should be demonstrated by a valid safety case that takes into account the implications from previous stages and for future stages.

502. To ensure that the future licensee considers the availability of on-site emergency power for post-operational clean out and decommissioning during detailed design I have raised GDA mechanical engineering assumption (AS-ABWR-ME04).

4.16.12 Conclusions

503. I am satisfied that my assessment strategy provides me with an adequate sample to identify the SSCs that are important for safety.

504. I consider that Hitachi-GE has adopted a suitable design process that should enable them to satisfy the requirements of relevant UK legislation and RGP.

505. Hitachi-GE has provided adequate evidence that its case has been prepared in accordance with its safety case development manual and GDA ALARP Methodology. I am satisfied that Hitachi-GE has a process which robustly enables it to consider normal operating and potential fault conditions including internal and external hazards, conventional safety and human factor influences that could affect safety.

506. I recognise that there will be further development of the emergency power supply system design during the detailed design phase. My expectation is that the responsibility to develop the emergency power supply system safety case, to include adequate arrangements in respect of licence conditions, will be the responsibility of any future licensee. The findings and assumptions identified during my assessment are set out below.

4.16.13 Regulatory Findings and Assumptions

507. Assessment finding relating to the emergency power supply system.

- **AF_ABWR_ME_01:** Hitachi-GE has not provided sufficient evidence that equipment qualification plans meet UK expectation. In particular, the plans failed to demonstrate a suitable sample size and testing regime both at system and component level. Furthermore, Hitachi-GE had not identified test standards in its equipment qualification plans or applied UK test conditions and timescales commensurate with UK ABWR expected lifetime. To address these shortfalls, the licensee shall develop equipment qualification plans, which consider these issues.

- **AF_ABWR_ME_07:** The J-ABWR design assumes a 40-year operational lifetime for SSC equipment qualification. In some cases, Hitachi-GE safety case claims that no further equipment qualification is required for UK ABWR despite it having a 60-year operational lifetime. The licensee shall identify and qualify those SSCs that they will not maintain or replace during the assumed lifetime of the plant.

508. Assumptions relating to the emergency power supply system.

- **AS-ABWR-ME04 -** During GDA, Hitachi-GE identified certain equipment that is required to perform activities associated with decommissioning of the plant at the end of its 60-year life. ONRs expectation is that any future licensee will provide suitable plant from the outset to avoid unnecessary modifications to plant in future prior to decommissioning. ONR assumes that the licensee shall identify and confirm the use of equipment for decommissioning shall consider design features to facilitate decommissioning and reduce future dose uptake by workers and where reasonably practicable include any necessary design features in the final design.

- **AS-ABWR-ME05 -** ONR acknowledges that during GDA, Hitachi-GE has not sufficiently developed its design to enable the licensee to identify the appropriate EIMT requirements in line with UK expectations. ONR assumes that the licensee shall ensure that they consider EIMT requirements for all mechanical engineering items, which attract a safety classification. ONR also assumes that the licensee shall establish whether these requirements are directly driven by the safety case, are based on manufacturer recommendations or are based on plant operating experience (or appropriate combinations).

- **AS-ABWR-ME07 -** During GDA Hitachi-GE did not describe the effective use, management and storage of Bio fuel for emergency diesel generators. ONR assumes that the licensee shall ensure that any diesel combustion plant used for the UK ABWR is designed to take into account the regulation amendment in respect of fuels, (Motor Fuel (Composition and Content) Regulations 1999), in terms of meeting their safety functional requirements

4.17 Heating, ventilation and air conditioning (HVAC) system

4.17.1 Introduction

509. Key safety claims made by Hitachi-GE in relation to its HVAC systems are;

- maintain design basis environmental conditions for systems, structures and components;
- limit the spread of radioactive contamination in normal and accident conditions;
- direct any discharges to suitably filtered routes;
- provide suitable environmental conditions for workers.

4.17.2 Overview of Hitachi-GE Safety Case

510. Hitachi-GE's Basis of Safety Case (BoSC) [135] details design requirements for HVAC. In terms of mechanical engineering, the key system safety functions identified in the BoSC are to:

- Provide contamination control and confinement through:
- controlled discharge and exhaust filtering to prevent radiological release above permissible limits;
- intake filtering and maintaining positive pressure to prevent radioactive material ingress in fault conditions;
- maintaining internal negative pressure requirements in controlled areas;
- HVAC system isolation requirements in fault conditions;
    - Manage internal environmental conditions for adequate equipment operation through:
    - heat removal for adequate equipment performance in normal and fault conditions;
    - filtering for dust and particulate removal.
    - Manage internal environmental conditions for adequate worker comfort and safety during normal and fault conditions through;
    - ventilation and air quality control;
    - temperature and humidity control;
- purge of nitrogen atmosphere from the primary containment for safe worker access during refuelling outages;
- maintain positive pressure in protected staircases to prevent smoke ingress for safe access and egress of fire-fighting brigades.

511. The Pre-Construction Safety Report (PCSR) summarises the case presented in the BoSC. The BoSC, in section 6, provides the link to the risk assessments (Failure Mode Effects Analysis (FMEA), Hazard and Operability studies (HAZOP) and Probabilistic Safety Assessment (PSA)) that Hitachi-GE has undertaken. This provides the initial means against which the HVAC SSCs are to be classified and categorised and will be capable of reducing risk so far as is reasonably practicable (SFAIRP). Detailed design will commence after GDA with support from UK consultants and HVAC system manufacturers to design develop and qualify equipment to a level commensurate with its category and classification.

512. In consideration of ONR SAP EKP.4 I am satisfied that the above provides evidence that Hitachi-GE has identified that delivery of the safety function(s) has followed a structured analysis.

4.17.3 Strategy for Assessment of Heating, ventilation and air conditioning (HVAC) system

513. My assessment only focuses on the HVAC systems, which support the delivery of a nuclear safety function.

514. The key documents that I sampled throughout GDA were the revisions of Hitachi-GE's HVAC BoSC [135] as it developed. I also assessed a number of related references that support the BoSC information [136], [137], [88], [138], [139], [140] [141].

515. ONR's Technical Support Contractor (TSC) assessed the version of the BoSC submitted at GDA Step 3, against ONR SAP's and RGP. The TSC's findings identified shortfalls in the evidence presented at that time by Hitachi-GE. The TSC presented these shortfalls in a report issued via RQ-ABWR-0636. This RQ required Hitachi-GE to:

1. familiarise itself with the report findings and observations;
2. confirm the report factual accuracy;
3. prepare to discuss the findings, observations and expectations in detail as part of the planned mechanical Engineering technical workshops;
4. in advance of the planned technical workshops, develop and advise its strategy to address the findings, observations and expectations.

516. Throughout GDA Step 4 I have engaged with Hitachi-GE through various level 4 engagements and I am satisfied that Hitachi-GE has adequately discussed the TSC's findings during these engagements. Through my assessment during step 4, I am satisfied that Hitachi-GE's final versions of the BoSC now provide sufficient evidence to satisfy the TSC's initial observations and findings.

517. ONR has undertaken a number of interactions with Hitachi-GE relating to HVAC at all stages of GDA via Level 4 contacts, Regulatory Queries (RQs) and Regulatory Observations (ROs).

4.17.4 Equipment Description

518. The HVAC system comprises numerous systems each supporting different buildings and supplying the HVAC safety functions. Typically, a category A and class 1 system is comprised of 3 divisions. Each division features supply fans, ductwork, dampers, air treatment facility including 2-stage HEPA filtration and exhaust fans. The primary safety functions of the HVAC system are to provide;

- contamination control;
- cascaded negative differential pressure to prevent gaseous or airborne contamination from spreading to areas of lower potential contamination or to outside the controlled areas;
- suitable environmental conditions (temperature and humidity) for SSCs housed within the reactor area.

519. The above functions are maintained for normal operations and faults outside the design basis. Under high heat loads, local cooling units activate in order to remove excess heat.

4.17.5 Assessment of Hitachi-GE's HVAC system mechanical engineering arrangements

520. **Nuclear Ventilation Codes and standards** – During step 2 GDA, ONR raised RO-ABWR-0017 – Nuclear Ventilation Codes and Standards. ONR considered that Hitachi-GE had not provided adequate assurance that it had designed SSCs in accordance with UK RGP. Hitachi-GE generated document [142] - List of Applicable

Legislation and RGP to Mechanical Engineering (ME) SSC. The document lists Relevant Statutory Provisions (RSP) and their applicable Approved Codes of Practice (ACoPs), RGP and guidance. I consider that this document along with the BoSC adequately identifies the relevant codes and standards applicable to the safe design of the HVAC components. Furthermore, I consider that this aligns with SAP ECS.3 that states that SSCs important to safety should be designed to appropriate codes and standards.

521. **Examination, Inspection, Maintenance and Testing (EIMT)** – During step 2 GDA ONR raised RO-ABWR-0018 relating to the adequacy of Hitachi-GEs arrangements for SSC EIMT. In response, Hitachi-GE generated document [33] - Strategy on Examination, Inspection Maintenance and Testing (EIMT) Isolations and Configurations. ONR considered this submission was adequate to close RO-ABWR-0018 on the basis that Hitachi-GE;

- demonstrated adequate progress in developing an overarching EIMT strategy;
- would implement this EIMT strategy across the whole plant during detailed design.

522. The HVAC BoSC sets out high-level EIMT proposals and states that detailed EIMT schedules will be incorporated into the operation and maintenance manuals and plant maintenance instructions. However, as the safety case has yet to define specific requirements for EIMT I consider there to remain a shortfall when compared with ONR SAP EMT.2, which states SSCs should receive regular and systematic examination, inspection, maintenance and testing as defined in the safety case. I have captured this under a GDA mechanical engineering assumption (AS-ABWR-ME05).

523. **Equipment Qualification (EQ)** - ONR SAP EQU.1 states that design qualification procedures should be applied to confirm that SSCs perform their allocated safety function(s) in all normal operational, fault and accident conditions. During Step 3 GDA ONR raised RO-ABWR-0051 – Mechanical Engineering SSC Qualification. This was to ensure that the UK ABWR SSC qualification arrangements were developed to demonstrate and substantiate the SSC's design basis through life. In relation to the HVAC system, Hitachi-GE developed [141] – Equipment Qualification Plan for Main Control Room HVAC supply fan. I consider this sample EQ plan to be equivalent to a Factory Acceptance Test plan. However, Hitachi-GE has demonstrated that high-level arrangements do exist to develop EQ plans to qualify its SSCs and it was on this basis that I was content to close RO-ABWR-0051. However, as I consider there to remain shortfalls in the specific EQ plans developed in GDA it is an expectation that EQ plans are further developed to include but not limited to;

- statistical sampling of SSC components to provide confidence in the robustness of the design over its intended operational life. Less reliance on single component single test;
- sequential or synergistic testing to verify SSC performance following interrelated or sequential fault conditions;
- consideration of sub-assembly and piece part component qualification not just system level qualification;
- test methods, test specifications, codes and standards should be clearly defined;
- qualification against UK specific operational, environmental, fault and accident conditions. Historic J-ABWR test data should not be assumed to be bounding;

- the use of J-ABWR OPEX (currently only 17 years) to underpin the UK ABWR 60 year qualified life should be justified. OPEX evidence should be complementary to physical test data and more reliance placed on accelerated aging;
- consideration of ONR SAPs EAD.1 to EAD.5 - ageing and degradation.

524. The above requirements are jointly captured under;

- Equipment Qualification: GDA Assessment finding AF_ABWR_ME_001;
- Ageing and degradation: GDA mechanical engineering AF_ABWR_ME_007**.**

4.17.6 System safety functions for Heating, Ventilation and Air Conditioning (HVAC) system

525. This section outlines the HVAC system safety functions as claimed in the safety case. Hitachi-GE uses the terminology Safety Functional Claims (SFC) and Safety Property Claims (SPC).

Safety Functional Claims (SFC)

526. Safety functional claims relate to the required function of the HVAC systems as derived from the fault analysis and from the identified High Level Safety Functions. Sections 4.1 to 4.13 of [135] provide the detailed safety case showing the relation between the High Level Safety Functions, the SFC and their arguments and evidence. The main safety functions the HVAC systems are designed to deliver are set out below:

   - High Level Safety Functional Claim 1: Confinement of radioactive material and contamination spread control

527. The HVAC systems in controlled areas are designed to prevent gaseous or airborne contamination spreading from higher contamination to areas to of lower contamination areas, or to outside of the controlled area. The exhausts also include filters to prevent contamination from discharging above acceptable levels.

528. The claims relevant to this High Level Safety Function are summarised in Figure 4 below [135].

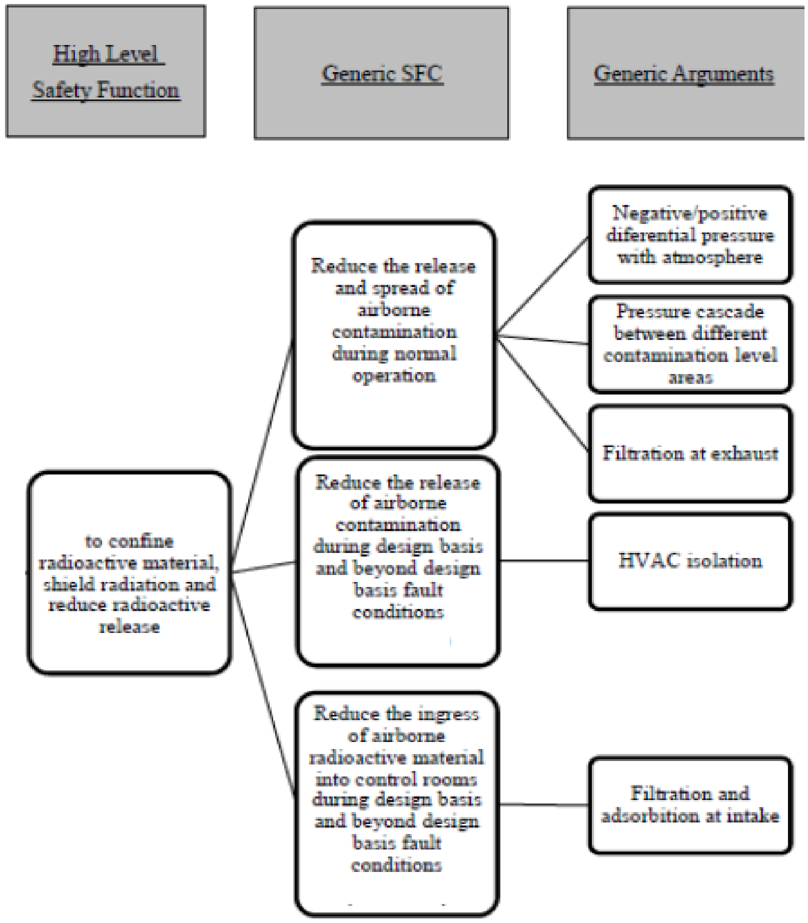

Figure 4 Claims Relevant to High Level Safety Functions

High Level Safety Function Claim 2 - Control of internal building environmental conditions

529. The HVAC systems ensure SSCs are under controlled environmental conditions for normal operation or during fault conditions and control the environmental conditions to maintain appropriate parameters for worker comfort. The HVAC systems also support the conventional fire protection strategy for the building.

530. The claims relevant to this High Level Safety Function are summarised in Figure 5 below [135].

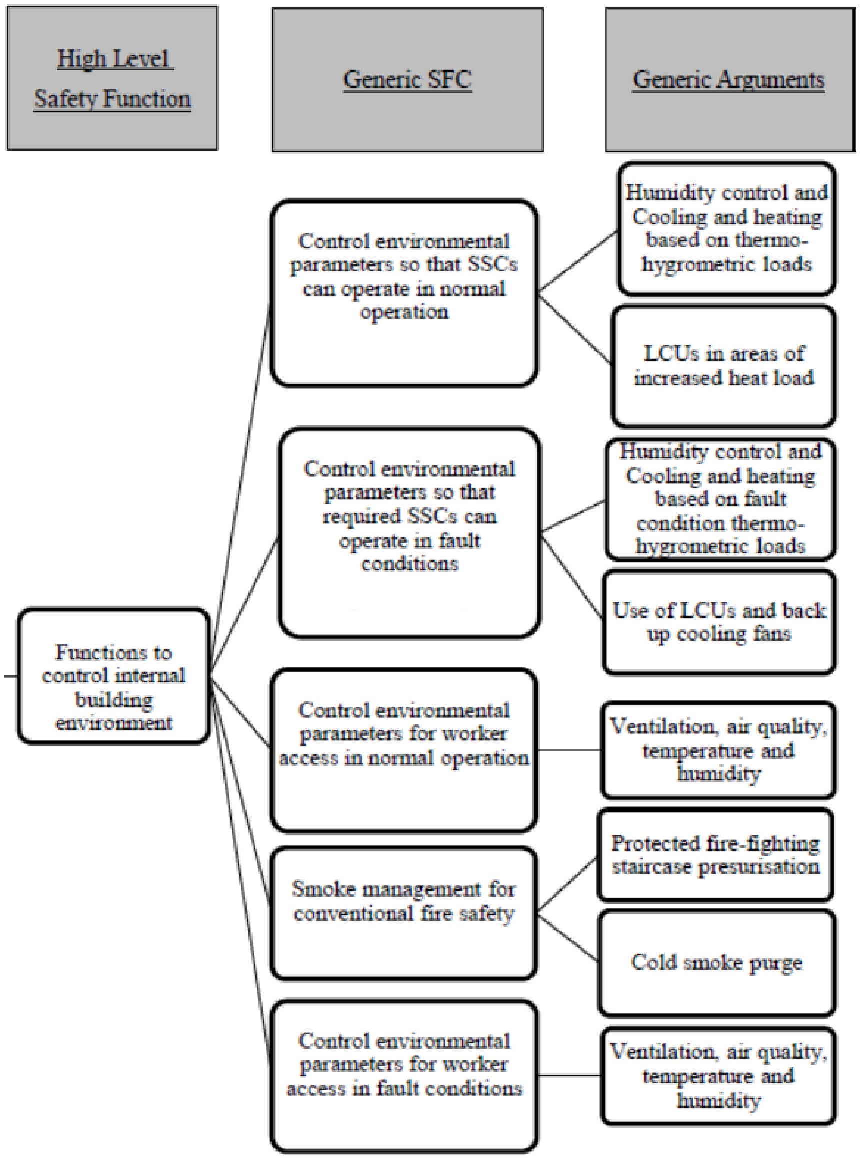

Figure 5 Claims Relevant to High Level Safety Functions

531. In consideration of ONR SAP EKP.4 I am satisfied that the above provides evidence that Hitachi-GE has identified that delivery of the safety function(s) has followed a structured analysis.

Safety Property Claims (SPC)

532. Hitachi-GE has assigned the HVAC systems with safety categorisations, which take account of ONR SAP ECS.2, Safety classification of SSCs. Hitachi-GE is proposing a reliability of 10^{-4} probability of failure on demand (pfd) for Class 1 HVAC SSCs and 10^{-3} pfd for Class 2 [143]. I consider this to align with the ONR TAG on categorisation and classification [37] which sets out target reliability figures for Class 1 and 2 SSCs. Detail design of the HVAC systems will be necessary to substantiate these claims but I consider that this approach should enable future licensees to undertake detailed design, manufacture and procurement to an appropriate classification.

4.17.7 Comparison of UK ABWR with J-ABWR reference design

533. As part of RO-ABWR-0017, Hitachi-GE undertook a gap analysis of extant UK RGP applicable to nuclear ventilation systems against the baseline J-ABWR design. The output is summarised in [144] and [145]. Hitachi-GE then used the results of the gap analysis to inform a HVAC multidisciplinary design review and HAZOP study [136]. The Hitachi-GE HVAC design team undertook this review with input from UK consultants. I consider the output of the HAZOP study has served to better align the J-ABWR reference design with UK RGP through the implementation of a number of design changes. Of particular note are;

- inclusion of safe change High Efficiency Particulate Absorption (HEPA) filters;
- controlled area Air Change Rates (ACH) are in accordance with;
  - ES_0_1738_1: Ventilation Systems for Radiological Facilities - Design Guide; and;
  - BS ISO 26802:2010: Nuclear facilities -- Criteria for the design and the operation of containment and ventilation systems for nuclear reactors.
- coalesce filters, used to separate liquid water and oil from compressed air;
- frost protection heating coils are incorporated into the supply air treatment facilities;
- additional instrumentation for fans;
- damper design changed to air-operated or motorized damper from a backdraft damper design;
- staircase pressurising system to generate 50Pa across closed doors as part of the fire protection system;
- internal and external environmental conditions aligned to UK generic site envelope.

4.17.8 Assessment of Heating, ventilation and air conditioning (HVAC) system

534. ONR issued RO-ABWR-0075 [146] following ONRs multi-disciplinary assessment of revision 2 of Hitachi-GEs HVAC BoSC. The RO identified perceived gaps in Hitachi-GEs safety case and grouped these into 13 separate areas.

535. Following the issue of RO-ABWR-0075 Hitachi-GE undertook a major rewrite of its HVAC BoSC. Following my detailed assessment of the submissions made in relation to RO-ABWR-0075 I consider that Hitachi-GE has adequately developed its HVAC system safety case for GDA. However, there remain a number of shortfalls, which need addressing by the future licensee and tracked into the detailed design phase. The latest version of the BoSC [135] identifies these as further work to be actioned by any future licensee, and these are summarised below;

   a. Validity of design requirements - Using UK met office data the peak external temperature which may occur during the operational lifetime of the UK ABWR is predicted at 45.9°C. One of the inputs informing the HVAC concept design is this peak temperature although Hitachi-GE considers this overly conservative. ONR expects accurate source data to be specified prior to the commencement of detailed design to ensure the requisite degree of nuclear safety can be delivered by the system. In addition, the validity and accuracy of the source data and its design parameters should be substantiated by robust claims, arguments and evidence.

b. Dangerous Substances and Explosive Atmospheres Regulations (DSEAR) – The future licensee is to provide an adequate safety justification to demonstrate that the HVAC system design is compliant with the requirements of DSEAR.

c. Limited safety case scope – within GDA the HVAC safety case attempts to provide concept design safety assessment of only a sample of the overall UK ABWR HVAC systems. Following GDA, the future licensee should adequately develop the safety case to provide detailed safety justification of all UK ABWR HVAC systems.

d. Local Exhaust Ventilation (LEV) – Hitachi-GE has identified within GDA that LEV systems may be required during maintenance, post fault scenarios and decommissioning. Hitachi-GE has not provided a safety justification on these systems within GDA. It is an expectation that the future licensee develops an LEV safety case.

e. Filter bank design – To reduce dose to workers during EIMT, Hitachi-GE has incorporated safe change HEPA filters into the UK ABWR design. However, the filter bank design layouts are in a ‘U’ configuration, which I do not consider relevant good practice as it potentially exposes the operator to a radiation source during filter changes. The layout of a filter bank is a key aspect of the design, which affects worker radiation dose. The future licensee should provide further ALARP justification for filter bank design.

f. Diversity – To minimise the effects of common cause failure (CCF) adequate diversity within SSCs should be provided. Achieving diversity down to component level for HVAC systems may be to the detriment of component quality and reliability, as there are a limited number of HVAC SSC suppliers. The future licensee should consider the quality, reliability and diversity requirements of HVAC systems during detailed design.

4.17.9 Availability of the HVAC system during post power generation lifecycle phases

536. ONR SAP SC.3 states that; for each lifecycle stage, control of the hazard should be demonstrated by a valid safety case that takes into account the implications from previous stages and for future stages.

537. The HVAC system safety case does not adequately consider the availability of the HVAC system during post-operational clean out and decommissioning life cycle phases. More onerous safety demands may be placed upon the HVAC systems (e.g. to control the spread of radioactive contamination) than in normal (power generation) operations.

538. The requirement for the HVAC systems to support post-operational clean out and decommissioning life cycle phases needs to be addressed. To ensure that any future licensee considers the availability of HVAC systems for post-operational clean out and decommissioning I have raised GDA mechanical engineering assumption (AS-ABWR-ME04).

4.17.10 Conclusions

539. I am satisfied that my assessment strategy provides me with an adequate sample to identify the SSCs that are important for safety.

540. I consider that Hitachi-GE has adopted a suitable design process that should enable it to satisfy the requirements of UK legislation and RGP.

541. Hitachi-GE has provided adequate evidence that its safety case has been prepared in accordance with its safety case development manual and GDA ALARP Methodology. I

am satisfied that Hitachi-GE has a process which enables it to consider normal operating and potential fault conditions; including internal and external hazards, conventional safety and human factor influences that could affect safety.

542. Hitachi-GE has made significant progress in the development of its HVAC system safety case within Step 4 of GDA. I consider the HVAC system design to be adequate and to meet my expectations for GDA.

543. I recognise that further development of the HVAC system design will occur during the detailed design phase. My expectation is that the responsibility to further develop the HVAC system safety case and include adequate arrangements in respect of licence conditions will rest with any future licensee. The findings and assumptions identified during my assessment for management by the licensee are set out below.

4.17.11 Regulatory Findings and Assumptions

544. Assessment findings relating to the HVAC system.

- **AF_ABWR_ME001:** Hitachi-GE has not provided sufficient evidence that equipment qualification plans meet UK expectation. In particular, the plans failed to demonstrate a suitable sample size and testing regime both at system and component level. Furthermore, Hitachi-GE had not identified test standards in its equipment qualification plans or applied UK test conditions and timescales commensurate with UK ABWR expected lifetime. To address these shortfalls, the licensee shall develop equipment qualification plans, which consider these issues.

- **AF-ABWR_ME_007:** The J-ABWR design assumes a 40-year operational lifetime for SSC equipment qualification. In some cases, Hitachi-GE safety case claims that no further equipment qualification is required for UK ABWR despite it having a 60-year operational lifetime. The licensee shall identify and qualify those SSCs that they will not maintain or replace during the assumed lifetime of the plant.

545. Assumptions relating to the HVAC system.

- **AS-ABWR-ME04 -** During GDA, Hitachi-GE identified certain equipment that is required to perform activities associated with decommissioning of the plant at the end of its 60-year life. ONRs expectation is that any future licensee will provide suitable plant from the outset to avoid unnecessary modifications to plant in future prior to decommissioning. ONR assumes that the licensee shall identify and confirm the use of equipment for decommissioning shall consider design features to facilitate decommissioning and reduce future dose uptake by workers and where reasonably practicable include any necessary design features in the final design.

- **AS-ABWR-ME05 -** ONR acknowledges that during GDA, Hitachi-GE has not sufficiently developed its design to enable the licensee to identify the appropriate EIMT requirements in line with UK expectations. ONR assumes that the licensee shall ensure that they consider EIMT requirements for all mechanical engineering items, which attract a safety classification. ONR also assumes that the licensee shall establish whether these requirements are directly driven by the safety case, are based on manufacturer recommendations or are based on plant operating experience (or appropriate combinations).

4.18 Through Wall Penetrations

4.18.1 Introduction

546. During step 3 GDA, ONR radiological protection specialists raised Regulatory Observation RO-ABWR-0065 – Demonstration of adequate design and implementation of inherently safe techniques and structure to minimise dose rates via through wall penetrations. This observation required Hitachi-GE to:

- Demonstrate that it has adequate arrangements for the design and implementation of inherently safe techniques and structures, to minimise radiation dose rates, via through wall penetrations; and
- Minimise the use of lead wool during all operating modes, for the lifetime of the generic design.

4.18.2 Overview of Hitachi-GE Safety Case

547. Hitachi-GE submitted its UK ABWR penetration design guidelines [147]. The purpose of this document is to demonstrate UK ABWR penetration design rules, and how to apply them to specific penetration designs using representative penetrations as examples. Hitachi-GE indicated that to develop these design rules the following steps were taken:

- Hitachi-GE produced its penetration design guidelines [147] that take into account UK RGP.
- The penetration design rules for UK ABWR were also developed in line with UK RGP, taking into account Hitachi-GE's penetration design guideline [147].

4.18.3 Strategy for Assessment of Through Wall Penetrations

548. ONR's radiological protection specialists led on assessing Hitachi-GE's penetration design guidelines and in judging the adequacy of the response to RO-ABWR-0065. ONR now considers that Hitachi-GE's response to RO-ABWR-0065 is satisfactory and the RO has been closed.

549. My mechanical engineering assessment was limited to assessing Hitachi-GE's design guidelines [147].

4.18.4 Equipment Description

550. The UK ABWR design incorporates numerous penetrations to allow a variety of services such as pipes, ducts, cables to enter radioactive areas. Penetrations that form part of a boundary between designated radiation zones should provide suitable shielding from potentially harmful radiation.

551. There are means of providing radiation shielding within penetrations, incorporating a combination of shielding materials and penetration geometry. Hitachi-GE intends to consider the design of each individual penetration separately using a range of options specified within its penetration design guidelines [147].

552. Hitachi-GE has stated that datasheets showing penetration specifications, such as the penetration location and size, will be developed for each penetration during the detailed design phase. In addition to penetrations for services, a number of personnel and/or equipment access penetrations are required. Hitachi-GE does not discuss this type of penetration in detail in the penetration guidelines document [147]. However, the principles of the design approach are the same for all penetrations.

4.18.5 Assessment of Design Process for Through Wall Penetrations

553. Hitachi-GE's penetration design guidelines [147] cover all relevant technical aspects in relation to the penetrations during normal operating conditions, fault conditions and

decommissioning, e.g. radiological protection, contamination control, fire resistance and flooding protection.

554. Hitachi-GE has developed the UK ABWR penetration design rules, and demonstrated the application of such design rules by using typical examples in the GDA phase. The individual and specific penetrations will be designed in detail in the site license phase based on the design rules developed in the GDA phase.

555. I recognise that further detailed consideration of the penetration design guidelines will occur during detailed design. My expectation is that responsibility for implementing these will rest with the licensee. I have raised a generic assumption to capture this expectation, which Hitachi-GE should communicate to any future licensee. I have captured this assumption as AS-ABWR-ME06.

4.18.6 Comparison of UK ABWR with J-ABWR reference design

556. Historically, lead based materials such as lead wool, lead shot and lead sheet have been used to provide radiation shielding in the nuclear industry and were used in the J-ABWR. However, Lead is a toxic material and UK legislation seeks to eliminate its use from new designs. In addition, lead and lead wool in particular are challenging and expensive to dispose of. Hence, Hitachi-GE's penetration design guidelines [147] seek to offer alternative design options eliminating lead wherever possible.

557. I consider that eliminating lead from penetrations in the J-ABWR reference design could potentially result in larger penetrations that might affect the J-ABWR reference design and layout for UK ABWR. This might affect radiological risk and EIMT. Based on my assessment of Hitachi-GE's penetration design guidelines [147]**,** I established that Hitachi-GE has identified a wide range of options as a means of providing adequate radiation shielding. From discussions with ONR radiation protection specialists, I consider that there are sufficient options available so that penetration size is not significantly affected for UK ABWR. Furthermore, I established that the penetration design guidelines do consider penetration removal and replacement, EIMT and decommissioning requirements. I have assumed that the licensee will undertake detailed design in accordance with Hitachi-GE's design guidelines and I have captured this assumption as detailed below (AS-ABWR-ME06).

4.18.7 Conclusions

558. I am satisfied that Hitachi-GE's penetration design guidelines provide sufficient guidance and options for future designers to detail design penetrations that provide adequate radiation shielding. Therefore removing the use of lead, should not significantly affect the J-ABWR reference design for the UK ABWR.

559. I have assumed that the detail design of penetrations for UK ABWR will ensure that the overall geometry of penetrations is not significantly changed from those used on J-ABWR. I have captured this assumption within AS-ABWR-ME06

4.18.8 Regulatory Assumptions

560. **AS-ABWR-ME06** – Hitachi-GE's change to the J-ABWR reference design (to facilitate removal of all lead materials where reasonably practicable) has the potential to alter the geometry and size of through wall penetrations. ONR assumes that the licensee shall ensure that these changes are made in accordance with Hitachi-GE's penetration design guidelines to minimise the impact on the reference design parameters.

4.19 Reactor Recirculation System

4.19.1 Introduction

561. The forces the reactor coolant to circulate through the reactor core. This transfers heat, generated by the nuclear fission reaction in the core, to the coolant. The system adjusts the core flow rate to control the reactor power.

562. The system supports the following:

- Reactivity control
- Heat transfer
- Confinement

4.19.2 Overview of Hitachi-GEs safety case

563. Hitachi-GEs Basis of Safety Case (BoSC) [148] details design requirements for the RRS. Hitachi-GE has made a number of safety functional claims given below, which are of interest to mechanical engineering assessment

- The reactor recirculation system in conjunction with the recirculation flow control system provides reactor coolant forced recirculation for power generation in plant normal operation conditions whose failure could lead to a total loss of reactor coolant flow.
- The reactor recirculation system in conjunction with the reactor flow control system provides reactor coolant forced recirculation for power generation in plant normal operation conditions whose failure could lead to a partial loss of reactor coolant flow.
- The reactor recirculation system portions within the reactor coolant pressure boundary contain reactor coolant in plant normal conditions and form a pressure barrier during fault conditions. The failure would result in a loss of reactor coolant with consequences above the baseline safety limit.

564. The following safety functions are developed and justified under different systems and therefore fall out of the scope of this assessment:

- The reactor internal pumps of the reactor recirculation system are tripped by the recirculation pump trip system by a signal from the hardwired backup system as part of the actions to perform alternative shutdown of the reactor in the event of anticipated transient without automatic shutdown.
- The reactor internal pumps of the reactor recirculation system are tripped by a signal from the Plant Control System as part of the actions used to deliver mitigation of power increases.
- The reactor recirculation components penetrating the primary containment form a barrier to confine the radioactive material within the containment boundary and prevent its dispersion to the environment in the event of faults.

565. Earlier versions of the BoSC were assessed by ONR’s Technical Support Contractor (TSC) against ONR SAP’s and RGP during GDA step 3. The TSC’s findings identified shortfalls in the evidence presented by Hitachi-GE in the earlier versions of the BoSC. The TSC presented these shortfalls in their report issued via RQ-ABWR-0616. This RQ required Hitachi-GE to:

1. familiarise itself with the report findings and observations,

2. confirm the report factual accuracy;

3. Hitachi-GE to respond to the TSC findings through the formal GDA RQ process for ONR assessment

566. Throughout step 4, ONR I have engaged with Hitachi-GE through various level 4 engagements. During these engagements, I have challenged Hitachi-GE on the shortfalls identified by the TSC and this has resulted in revised submissions of the BoSC. I am now satisfied that Hitachi-GE's revised BoSC provides sufficient evidence to address the TSC's initial observations and findings.

567. Hitachi-GE has provided piping and instrumentation diagrams [149] to support its evidence and provide an overview of the design of reactor recirculation system functions. I am content the proposed design meets that described in the BoSC and design description system intent [148] [150].

568. Hitachi-GE has provided claims arguments and evidence for categorisation and classification of the reactor recirculation system functions. I am satisfied that Hitachi has satisfied the guidance given in NS-TAST-GD-094 [37].

569. Hitachi-GE has provided a topic report for design justification of the reactor internal pump [151]. I am content this document adequately demonstrates justifications against safety function claims.

570. I am satisfied that this approach demonstrates that Hitachi-GE have adequately considered the likely failure mechanisms, consequences of failure and design mitigation in accordance with ONR SAP EKP.4 safety function.

4.19.3 Strategy for Assessment of Reactor Recirculation System

571. My assessment sample is based on:

- Single failure and common cause failure
- Qualification of the design
- Category and classification

572. The main document sampled for the purposes of this assessment was the basis of safety case for the reactor recirculation system [148]. Other documents used to inform my judgement are:

- Topic report on reactor internal pump design justification [151]
- Reactor internal pump piping and instrumentation diagram [149]
- Reactor internal pump system design description [150]
- Hitachi RQ responses

4.19.4 System description

573. The reactor recirculation system consists of 10 reactor internal pumps installed at the bottom of the reactor pressure vessel to provide forced circulation enabling recirculation of the reactor coolant for heat transfer. Their rotational speed, controlled

by the recirculation flow control system, changes flowrate to control reactor power without changing the position of the control rods.

4.19.5 Reactor Recirculation System Single failure and common cause failure

574. During GDA step 3 ONR required clarity on Hitachi-GEs rational, as to why the reactor internal pumps were not subject to single and common cause failure analysis. ONR requested this information in RQ-ABWR-0464 [152].

575. Hitachi-GE's response [153] states that reactor internal pumps are a class 3 component as per ONR guidance on categorisation and classification [37] and as such are not required to protect against single or common cause failure due to their low nuclear significance. However, the internal pumps are integral to the power output so Hitachi-GE has provided a degree of redundancy and segregation.

576. Hitachi-GE has designed the reactor internal pumps to ensure that the failure of one pump does not affect the desired power output, as the reactor can meet its design criteria with nine pumps. Hitachi-GE claims this in the BoSC [148], though the evidence to underpin this claim is unclear. I have held discussions with the fault studies discipline and I am satisfied that analysis of pump failures has been undertaken and assessed under this specialism area. The reactor internal pumps as also divided in to sub-systems, with separate power supplies. This adds a degree of redundancy to the design of the reactor recirculation system as identified in the piping and instrumentation diagram [149]. I am satisfied for the purposes of GDA that Hitachi-GE has addressed SAP EDR.3 common cause failure.

4.19.6 Reactor Recirculation System Qualification

577. I assessed the adequacy of the qualification for the reactor internal pump. Hitachi-GE claim the reactor recirculation system components are qualified to demonstrate that design intent is met under the operation and environmental conditions associated with the safety functions. This is in line with expectations set out in ONR TAG on Design safety assurance [14].

578. Hitachi-GE claim within its topic report on the reactor internal pump design justification [151] that the concept qualification for the reactor internal pumps is complete. As the design for the UK ABWR is almost unchanged to that of the J ABWR concept qualification Hitachi-GE has qualified the design based on a combination of testing and operational experience. The topic report [151] lists testing parameters met during full scale continuous operational testing for the J ABWR these include:

- Flow rate
- Differential pressure
- Temperature of motor cooling water
- Motor casing vibration
- Impeller vibration

579. The reactor internal pumps have been subject to commissioning tests along with operation on J-ABWRs. I am content that Hitachi-GE has sufficiently demonstrated that the concept design for the reactor internal pumps aligns with the guidance set out in ONR SAP EQU.1 qualification procedures.

580. Hitachi-GE has sentenced commissioning tests with suitable acceptance criteria for the UK ABWR with in the BoSC [148]. The proposed testing aims to demonstrate the

reactor recirculation's systems ability to meet the required safety functions. The acceptance criteria should confirm that the pump meets its operational conditions and set points. I am content with the proposed commissioning testing for the purposes of GDA. I am also content that Hitachi-GE's proposals align with SAP ECM.1 commission testing.

581. It is ONRs expectation that the licensee shall define testing parameters in more detail in terms of temperatures and pressures for UK conditions during licensing. I recognise that further detailed considerations for qualification will occur during detailed design. My expectation is that responsibility for making and implementing adequate qualification arrangements in respect of licence conditions will rest with the licensee. I have captured this as an assumption (AS-ABWR-ME02).

582. Hitachi-GE proposes that UK ABWR operational life is to be 60 years rather than the 40 years of the J-ABWR, with plant operating cycles extending from 12 to 18 months. Hitachi-GE claims in the BoSC [148] that the effects of the additional life span on main component parts are negligible and consumables are replaced at regular defined periods based on operation experience. I am content with Hitachi-GEs arguments regarding ageing of the reactor internal pumps main components. However Hitachi-GE has yet to define the consumables within the reactor recirculation system or their proposed replacement periodicity, for example the 'O' ring within the reactor internal pump motor is susceptible to ageing under operating conditions.

583. It is my expectation that the licensee will define consumables and replacement periods within the reactor recirculation system. I recognise that further detailed considerations for EIMT will occur during detailed design. My expectation is that responsibility for making and implementing adequate EIMT arrangements in respect of licence conditions will rest with the licensee. I have raised a regulatory assumption to capture this expectation that should be communicated by Hitachi-GE to any future licensee (AS-ABWR-ME05).

584. Hitachi-GE has undertaken an ALARP risk review on the reactor internal pump design. In order to apply relevant good practice to the design, the upper plug seal requires double isolation. Hitachi-GE has proposed to include an additional O-ring to the reactor internal pump plug design in order to provide double isolation during maintenance. Feasibility of the reactor internal pump plug design is to be considered during detailed design. It is ONRs expectation that the concept design of the upper plug design is qualified during detailed design and this has been captured as a regulatory finding below.

4.19.7 Reactor Recirculation System Category and classification

585. During GDA step 3, ONR requested information on the category and classification given to the reactor recirculation system via RQ-ABWR-0468 [154]. ONR also asked for Hitachi-GE's arguments as to why the classification of components was "3" (SSC contributing to a categorised safety function) and not "2" (makes a significant contribution to a nuclear safety function). Hitachi-GE's response [155] presents arguments that the reactor recirculation system is a power generation system and it does not make claims for fault conditions such as a loss of coolant action. Fault conditions would lead to the need for a category A system to be implemented therefore the reactor recirculation system is a preventative system given a category B safety function. The argument is substantiated within the BoSC [148], were it is stated that if the reactor recirculation system was lost a rapid reactor shutdown would be initiated with the following category A systems being implemented:

- Reactor Core Isolation Cooling System

- Safety Relief Valves
- Residual Heat Removal System

586. The classification of the reactor recirculation is justified as the loss of one or more of the reactor internal pumps would not significantly affect the ability of the system to provide its safety functions or its power functions. Hitachi-GE substantiates this claim within the BoSC [148] and design description [150]. I am content that the evidence provided meets the requirements set out in SAPs ECS.1 Safety categorisation and ECS.2 Safety classification of structures, systems and components.

4.19.8 Conclusions

587. I consider that Hitachi-GE has adopted a suitable design process that has enabled it to adequately categorise and classify the reactor recirculation system.

588. I am satisfied that Hitachi-GE has considered EIMT in its safety case submission. I have identified an assumption captured through an ONR generic assessment finding that detailed design adequately considered EIMT requirements.

589. I am satisfied that Hitachi-GE has considered equipment qualification in their safety case submission. I have identified an assumption captured through an assessment finding that detailed design adequately considers equipment qualification requirements.

590. I am satisfied that Hitachi-GE has adequately identified ALARP considerations and the incorporation of UK relevant good practice for the design of the reactor internal pumps.

4.19.9 Regulatory Findings and Assumptions

591. Regulatory Findings for Reactor Recirculation System

- **AF-ABWR-ME-005**: Hitachi-GE in its ALARP study identified that the design of the reactor internal pump upper plug does not meet UK RGP since it relies single isolation. The licensee shall develop the design of the reactor internal pump upper plug, meeting the identified requirements of UK relevant good practice. The concept design shall be qualified with evidence provided to substantiate any design changes made.

592. Regulatory Assumptions for Reactor Recirculation System

- **AS-ABWR-ME02**: ONR acknowledges that during GDA, Hitachi-GE had not sufficiently developed its design to enable the licensee to identify the appropriate design qualifications details. ONR assumes that the licensee shall establish detailed design substantiation, factory acceptance test information, and site acceptance test information for individual mechanical items and their associated systems, which are important to safety. ONR also assumes that the licensee shall generate appropriate evidence that equipment qualification is adequately specified for all mechanical items important to safety in accordance with UK expectations.

- **AS-ABWR-ME05:** ONR acknowledges that during GDA, Hitachi-GE has not sufficiently developed its design to enable the licensee to identify the appropriate EIMT requirements in line with UK expectations. ONR assumes that the licensee shall ensure that they consider EIMT requirements for all mechanical engineering items, which attract a safety classification. ONR also assumes that the licensee shall establish whether these requirements are directly driven by the safety case, are based on manufacturer recommendations or are based on plant operating experience (or appropriate combinations).

4.20 Fine Motion Control Rod Drive (FMCRD) system

4.20.1 Introduction

593. The FMRCD system inserts control rods from beneath the reactor and is a key safety system in the UK ABWR. Its primary safety function is to provide emergency shutdown of the reactor and maintain sub-criticality in the core in the event of a fault. In normal power operation, the system is used, in conjunction with the reactor recirculation system to control the reactivity of the reactor core.

4.20.2 Overview of Hitachi-GE Safety Case

594. During power operation, the FMCRD system controls changes in core reactivity by movement and positioning of neutron absorbing control rods within the core in response to signals from the control system. Fully inserting the rods shuts down the reactor fully.

595. Hitachi-GE's Basis of Safety Case (BoSC) [156] details the safety functional requirements and design requirements for the FMRCD system. I have identified the key system safety functions, relative to my mechanical engineering assessment, below:

- During transients, the control rods are driven rapidly into the core using pressurised water from a hydraulic control unit.
- Following a rapid shutdown and once the control rods are fully inserted, a latch engages to prevent the control rod from dropping away from the core under gravity. The actuation of the latch is automatic in nature.
- In the event the normal reactor protection system fails to insert the rods and shutdown the reactor, an Alternative Rod Insertion (ARI) system provides a diverse means of inserting the control rods into the reactor core.
- The ABWR reactor pressure vessel (RPV) undergoes periodic hydrostatic testing to check for leaks, the control rod drive pumps are used to pressurise the RPV during these tests.
- Pressure Boundary – 205 FMCRDs penetrate the reactor coolant pressure boundary (i.e. RPV). These penetrations must be capable of withstanding the full range of design basis pressures in order to maintain containment of the reactor coolant.

596. Chapter 12 of the Pre-Construction Safety Report (PCSR) summarises the case presented in the FMCRD BoSC [156] . The BoSC, in section 6, provides the link to the risk assessments (FMEAs, HAZOP studies and PSA) that Hitachi-GE has carried out from a nuclear, conventional and environmental safety perspective. This provides evidence that the concept design of the FMCRD system has followed Hitachi-GE's process to demonstrate that safety functions and systems have been appropriately classified and categorised and risk is reduced so far as is reasonably practicable (SFAIRP). Detailed design will commence after GDA by the UK licensee to design, develop and qualify equipment to a level commensurate with its assigned category and classification.

597. I am satisfied, being proportionate to the scope of GDA, that Hitachi-GE has adequately considered the likely failure mechanisms, consequences of failure and

design mitigation. In consideration of ONR SAP EKP.4, I am satisfied that the above provides evidence that Hitachi-GE has identified that delivery of the safety function(s) has followed a structured analysis.

598. Strategy for Assessment of Fine Motion Control Rod Drive (FMCRD) system

599. ONR has undertaken a number of interactions with Hitachi-GE relating to the FMCRD system at all stages of GDA via Level 4 meetings, Regulatory Queries (RQs) and Regulatory Observations (ROs).

600. Throughout my GDA assessment, I have sampled the revisions of Hitachi-GE's FMCRD system BoSC [156] as it developed and its supporting references.

601. ONR's Technical Support Contractor (TSC) assessed the version of the BoSC submitted at GDA Step 3 against ONR SAP's and RGP. The TSC's findings identified shortfalls in the evidence presented by Hitachi-GE at that time. The TSC presented these shortfalls in a report that was issued via RQ-ABWR-0653 [157]. This RQ required Hitachi-GE to:

- familiarise itself with the report findings and observations;
- confirm the report factual accuracy;
- prepare to discuss the findings, observations and expectations in detail as part of the planned mechanical Engineering technical workshops;
- develop and advise its strategy to address the findings, observations and expectations in advance of the planned technical workshops.

602. Throughout GDA Step 4 I have engaged with Hitachi-GE through various level 4 engagements and I am satisfied that Hitachi-GE has adequately discussed the TSC's findings during these engagements. Through my assessment during step 4, I am satisfied that Hitachi-GE's final versions of the BoSC generally provide sufficient evidence to satisfy the TSC's initial observations and findings.

4.20.3 Equipment Description

603. The FMCRD system is described in detail in section 3 of the BoSC [156]. The UK ABWR has 205 FMCRDs positioned below the Reactor Pressure Vessel (RPV). The FMCRDs insert and remove the control rods from the reactor core. Each FMCRD has two components (i.e. upper component and lower component). Both the lower and upper components contain reactor water, which needs draining during EIMT activities. The FMRCD is shown in Figure 6 below;

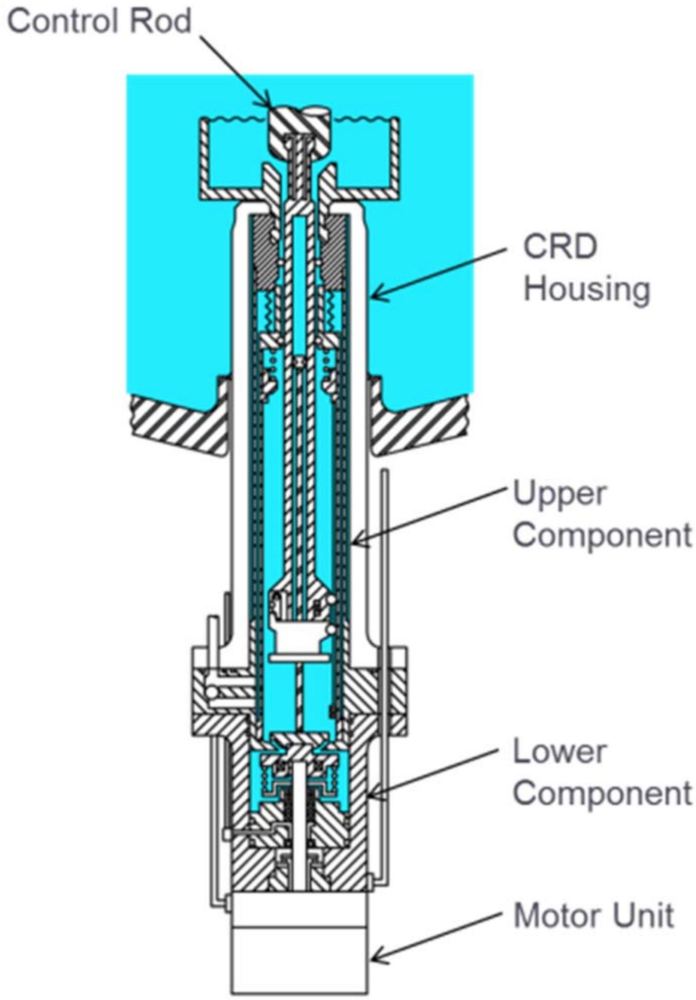

Figure 6 Fine Motion Control Rod Drive assembly

604. The FMCRDs utilises an electric motor to position control rods for normal insertion and withdrawal. The FMCRD can also insert control rods using hydraulic power under abnormal operating conditions when a rapid reactor shutdown is required. Each FMCRD is mounted in a housing welded into the reactor pressure vessel.

4.20.4 Assessment of Hitachi-GEs FMCRD system mechanical engineering arrangements

605. **Examination, Inspection, Maintenance and Testing (EIMT)** – Hitachi-GEs FMCRD safety case sets out high level requirements for in-service testing, inspection and other maintenance procedures and also proposes indicative frequencies. However, the level of detail relating to EIMT does require further development. I consider there to remain a shortfall when compared with ONR SAP EMT.2, which states SSCs should receive regular and systematic examination, inspection, maintenance and testing as defined in the safety case. I have captured this under GDA mechanical engineering assumption (AS-ABWR-ME05).

606. **Equipment Qualification (EQ)** - ONR SAP EQU.1 states that design qualification procedures should be applied to confirm that SSCs perform their allocated safety function(s) in all normal operational, fault and accident conditions. During Step 3 GDA ONR raised RO-ABWR-0051 – Mechanical Engineering SSC Qualification. This was to ensure that the UK ABWR SSC qualification arrangements were adequately developed to demonstrate and substantiate the SSCs design basis through life. In

relation to the FMCRD system, Hitachi-GE developed [158] - Equipment Qualification (EQ) Plan for Fine Motion Control Rod Drive. I consider this sample EQ plan to present shortfalls against ONR SAP EQU.1. However, being proportionate to GDA, Hitachi-GE has demonstrated that it has high-level arrangements in place to develop EQ plans to qualify its SSCs. It was on this basis that I was content to close RO-ABWR-0051. I have made an assumption that Hitachi-GE's FMCRD EQ plan is further developed by the future licensee to include but not limited to:

- statistical sampling of SSC components to provide confidence in the robustness of the design over its intended operational life. Less reliance on single component single test;
- sequential or synergistic testing to verify SSC performance following interrelated or sequential fault conditions;
- consideration of sub-assembly and piece part component qualification not just system level qualification;
- test methods, test specifications, codes and standards should be clearly defined;
- qualification against UK specific operational, environmental, fault and accident conditions. Historic J-ABWR test data should not be assumed to be bounding;
- the use of J-ABWR OPEX (currently only 17 years) to underpin the UK ABWR 60 year qualified life should be limited. OPEX evidence should be complementary to physical test data and more reliance placed on accelerated ageing.

607. The above requirements are captured under GDA Assessment finding AF_ABWR_ME_001.

608. **Operational Experience** – During step 3 GDA ONR raised RO-ABWR-0045 - Operational Experience (OPEX). ONR considered that Hitachi-GE had not adequately taken into account OPEX from BWR plants from around the world during the development of its UK ABWR concept design. It is ONR's expectation (SAP Para 36) that seeking and applying lessons learned from events, new knowledge and experience, both nationally and internationally, must be a fundamental feature of the safety culture of the nuclear industry. It is further stated (SAPs Para 110), that reviews of incidents, operating experience and other sources of information should not be restricted to the facility or site in question. They should include similar facilities or equipment and a range of nuclear and non-nuclear experience, both nationally and internationally.

609. In responding to this observation, Hitachi-GE produced [98] that identifies OPEX relating to control rod drive system failure modes. It is my expectation that this learning informs the detailed designs of the UK ABWR FMCRD systems.

4.20.5 Control Rod Drive (CRD) system safety functions

610. This section outlines the CRD system safety functions as claimed in the safety case. Hitachi-GE uses the terminology Safety Functional Claims (SFC) and Safety Property Claims (SPC);

<u>Safety Functional Claims (SFC)</u>

611. These are claims relating to the required function of the CRD system derived from the fault analysis. The relationship between the safety functional claims for this system and the faults against which Hitachi-GE expects each to be delivered, are shown in the claim tree in Section 6.3 of [156]. In consideration of ONR SAP EKP.4, I am satisfied

that Hitachi-GE has identified the safety function(s) to be delivered by the CRD system using a structured analysis.

Safety Property Claims (SPC)

612. These are principles applied by Hitachi-GE when designing SSCs in accordance with their Nuclear Safety and Environmental Design Principles (NSEDP). Hitachi-GE has provided evidence in its BoSC that its NSEDPs specify similar requirements to ONR's SAPs. This has provided me with sufficient evidence that Hitachi-GE has a process that will align with the relevant ONR SAPs.

613. Hitachi-GE has assigned the CRD system with safety categorisations that align with ONR SAP ECS.2, Safety classification. Hitachi-GE is proposing a probability of failure on demand reliability of 10^{-4} for Class 1 RCD components and 10^{-3} for Class 2 components [143]. I consider this comparable with the ONR TAG on categorisation and classification [37] which sets out target reliability figures for Class 1 and 2 SSCs. Detailed design of the CRD system will be necessary to substantiate these claims but I consider that this approach should enable the future licensee to undertake detailed design, manufacture and procurement to an appropriate classification.

4.20.6 Assessment of Control Rod Drive Mechanism

614. The CRD safety case makes specific performance claims on the system safety functions. One example of this is "60% insertion of the control rods within 1.44 seconds". During my assessment, I sought assurance that the evidence exists to underpin the system design basis. I sampled development testing data for the J-ABWR for the following Class 1 safety functions;

- Hydraulic control unit valve opening time;
- Control rod insertion time.

615. The test reports I sampled provided me with assurance that the design basis is in place for the CRD system. Hitachi-GE did not submit test data to ONR within GDA, this was only available for sampling at Hitachi works. The ONR contact record for the visit to Hitachi works contains the detail of the evidence sampled [133] and shows that the Class 1 safety functions above were met. I advised Hitachi-GE that for completeness of the safety case, it should supply all underpinning design and test data to any future licensee. I have raised a regulatory finding to capture this future requirement (**AF_ABWR_ME01).**

Material aging and degradation

616. Hitachi-GE's CRD equipment qualification plan [158] claims that the CRD has been qualified for 40 years for the J-ABWR, that 17 years' service has been completed and therefore that no further qualification is required over that already carried out historically. I challenged this claim as the UK ABWR design has a 60 years design life compared to only 40 years for the J-ABWR, an increase of 50%. Hitachi-GE stated that this claim is based on the number of control rod drive times anticipated in 40 years for the J-ABWR bounding the number of drives anticipated in 60 years the for UK ABWR. Hitachi-GE undertakes qualification as a function of drive distance travelled during rapid reactor shutdown not age in years. For the UK ABWR, fewer rapid shutdowns are anticipated due to the lower seismic activity in the UK geographical area. Hitachi-GE's qualification has considered the number of motor steps corresponding to a high number of rapid reactor shutdowns.

617. I examined Hitachi-GE's test report on "CRD separation countermeasure verification test (40 years life)". This report provided test results for a single CRD unit life test and

I am satisfied that Hitachi-GE has adequately compared this with its design qualification for 60 years life. .

618. I questioned what measures would be implemented to ensure that UK ABWR CRDs will not exceed this qualified distance over a 60 year period. Hitachi-GE claimed that a system will be put in place to monitor and record drive distance during plant operations and if the qualified life is exceeded the CRD will be replaced. I consider this qualification strategy to present a potential shortfall against ONR SAP EAD.1 and EAD.2 as it fails to take into account the aging and degradation that may occur in materials subjected to a 50% increase in operational life.

619. I have captured this requirement under assessment finding AF_ABWR_ME_07.

CRD upper component maintenance

620. During a reactor outage, a sample of CRD components is subject to EIMT. If removal of the upper component is required, a single metal-to-metal seal provides a temporary reactor pressure vessel containment barrier for the contaminated reactor coolant. However, there appears to be no positive means of confirming the integrity of the single seal prior to commencing EIMT operations. Operator actions confirm seal integrity only during the removal of the upper component. Any leak during this operation, requires remedial action by the operator [159]. This introduces the risk of generating radioactive waste through the spread of contamination. There is also a risk of contamination to workers present in the lower drywell during EIMT operations.

621. I consider the above arrangement to present the following shortfalls against UK RGP:

- HSG253 [160] - The safe isolation of plant and equipment [160], states that multiple seal arrangements should be adopted for temporary isolations;
  - Detection of a leak through operator action presents a potential shortfall against ONR SAP EKP.5. ONR SAPs Para 155 sets out a hierarchy of safety measures to be identified to deliver safety functions. Top of the hierarchy are:
    - Passive safety measures that do not rely on control systems, active safety systems or human intervention, and
    - automatically initiated active engineered safety measures.
- ONR SAP ECV.1 states that radioactive material should be contained and the generation of radioactive waste through the spread of contamination by leakage should be prevented.
- ONR SAP ECV.3 states that the primary means of confining radioactive materials should be through the provision of passive sealed containment systems and intrinsic safety features, in preference to the use of active dynamic systems and components.

622. Following identification of the above shortfalls, ONR issued RQ-ABWR-1217 [161]. In response, Hitachi-GE revised its ALARP assessment report for CRD upper component maintenance [159]. This document outlines an optioneering study conducted by Hitachi-GE and UK consultants. It identifies a number of measures to reduce risk during CRD upper component EIMT operations to ALARP. The document presents a number of risk mitigation options along with arguments for why Hitachi-GE has discounted some options. Hitachi-GE has stated that it does not intend to implement any design modifications to the CRD system or associated equipment within GDA. However, it does not limit or foreclose modification options by the licensee. I consider that further work to reduce the risk to ALARP is required during detailed design.

623. I have captured the above shortfall under GDA assessment finding AF_ABWR_ME006.

624. Hitachi-GE states in [159] that current practice for J-ABWR is that routine upper component EIMT will be carried out on five to six CRD units per outage. For the UK ABWR, Hitachi-GE proposes that the upper component will be qualified for a 60-year life and EIMT will only occur if a fault is detected. .

625. I challenged this strategy because the upper component contains two rubber 'O' rings which I consider would be difficult to qualify for a 60 year life under UK ABWR environmental conditions. In response to RQ-ABWR-1217, Hitachi-GE stated that the rubber 'O' rings do not contribute to any CRD safety function and therefore the EQ plan does not need to place maintenance requirements on the 'O' rings.

626. I consider that although the 'O' rings may not contribute directly to a safety function, if they do fail the upper component may need to be removed for 'O' ring replacement. In my judgement, it is during the removal of the upper component that a significant leakage of reactor coolant may still occur. Therefore, I consider that there are safety implications associated with not performing adequate EIMT. I have captured this under GDA assessment finding AF_ABWR_ME_002.

Hydraulic control unit support frame.

627. Following my assessment of Hitachi-GEs design justification report for the hydraulic control unit [162] I raised RQ-ABWR-0828 [163] requesting further design detail. The hydraulic control unit and associated components are required to rapidly shutdown the reactor.

628. I challenged Hitachi-GE's claim in [162] that the class 1 frame provides support to the class 1 hydraulic control despite it relying on class 3 foundation bolts to secure it to the civil structure. Hitachi-GE confirmed that it now intends to design and qualify the foundation bolts to meet safety Class 1 requirements.

629. I requested further clarification on the arrangements for fastenings and where this detail is to be set out in the safety case. Hitachi-GE explained how it intends to secure the frame to the civil structure and confirmed that this detail will be included with the next revision of its hydraulic control unit design justification report.

630. I requested details of the planned qualification testing to be conducted on the hydraulic control unit support frame to ensure the performance, reliability and safety of the frame throughout its 60-year service life. Hitachi-GE made outline proposals for how it intends to qualify the frame during detailed design. These proposals met my expectations for GDA.

4.20.7 Fine Motion Control Rod Drive system availability post power generation

631. Hitachi-GE's Basis of Safety Case (BoSC) [154] makes safety claims on the CRD system during the operational life of the plant only for 60 years of power generation. It does not consider any safety requirements for the CRD system during the post power generation phase.

632. ONR SAP SC.3 states that for each lifecycle stage, control of the hazard should be demonstrated by a valid safety case that takes into account the implications from previous stages and for future stages. To ensure that the future licensee considers the availability of the CRD system for the post power generation life cycle phases I have raised GDA mechanical engineering assumption (AS-ABWR-ME04)

4.20.8 Conclusions

633. I am satisfied that my assessment strategy provides me with an adequate sample to identify the SSCs that are important for safety.

634. I consider that Hitachi-GE has adopted a suitable design process that should enable it to satisfy the requirements of UK legislation and RGP.

635. Hitachi-GE has provided adequate evidence that its safety case has been prepared in accordance with its safety case development manual and GDA ALARP methodology. I am satisfied that Hitachi-GE has a process which enables it to consider normal operating and potential fault conditions including internal and external hazards, conventional safety and human factor influences that could affect safety.

636. Hitachi-GE has made significant progress in the development of its CRD system safety case within Step 4 of GDA. I consider the CRD system design to be adequate and to meet my expectations for GDA.

637. I recognise that further development of the CRD system design will occur during the detailed design phase. My expectation is that the responsibility to develop the CRD system safety case, to include adequate arrangements in respect of licence conditions, will rest with the future licensee. The findings and assumptions identified during my assessment, to be managed by the future licensee, are set out below.

4.20.9 Regulatory Findings and Assumptions

638. Assessment findings relating to the CRD system:

- **AF_ABWR_ME01:** Hitachi-GE has not provided sufficient evidence that equipment qualification plans meet UK expectation. In particular, the plans failed to demonstrate a suitable sample size and testing regime both at system and component level. Furthermore, Hitachi-GE had not identified test standards in its equipment qualification plans or applied UK test conditions and timescales commensurate with UK ABWR expected lifetime. To address these shortfalls, the licensee shall develop equipment qualification plans, which consider these issues.

- **AF_ABWR_ME04:** Hitachi-GE's design for UK ABWR eliminates routine EIMT for the FMCRD upper component. The licensee shall provide a suitable safety justification for this deviation from Japanese RGP during detailed design. The justification shall include evidence to support any decisions that the FMCRD upper component can be qualified to meet the 60-year design life.

- **AF_ABWR_ME06:** Hitachi-GE has not provided sufficient evidence during GDA to demonstrate that maintenance activities on the FMCRD upper component meet UK standards of double isolation. The licensee shall provide an adequate demonstration that EIMT activities on the FMCRD can be performed safely, ensuring that risks are reduced ALARP.

- **AF_ABWR_ME_07:** The J-ABWR design assumes a 40-year operational lifetime for SSC equipment qualification. In some cases, Hitachi-GE safety case claims that no further equipment qualification is required for UK ABWR despite it having a 60-year operational lifetime. The licensee shall identify and qualify those SSCs that they will not maintain or replace during the assumed lifetime of the plant.

639. Assumptions relating to the CRD system.

- **AS_ABWR_ME04:** During GDA, Hitachi-GE identified certain equipment that is required to perform activities associated with decommissioning of the plant at the end of its 60-year life. ONRs expectation is that any future licensee will provide suitable plant from the outset to avoid unnecessary modifications to plant in future prior to decommissioning. ONR assumes that the licensee shall identify and confirm the use of equipment for decommissioning shall consider design features to facilitate decommissioning and reduce future dose uptake by workers and where reasonably practicable include any necessary design features in the final design.

- **AS_ABWR_ME05:** ONR acknowledges that during GDA, Hitachi-GE has not sufficiently developed its design to enable the licensee to identify the appropriate EIMT requirements in line with UK expectations. ONR assumes that the licensee shall ensure that they consider EIMT requirements for all mechanical engineering items, which attract a safety classification. ONR also assumes that the licensee shall establish whether these requirements are directly driven by the safety case, are based on manufacturer recommendations or are based on plant operating experience (or appropriate combinations).

4.21 Systems, Structures and Components (SSC) assessed by ONRs Technical Support Contractor (TSC) at GDA Step 3.

4.21.1 Introduction

640. The range of mechanical engineering SSCs within the UK ABWR is extensive so it was not practicable, nor proportionate, for ONR to assess every SSC in detail during GDA to identify weaknesses in the overall safety case. The mechanical engineering strategy was to adopt a focused and targeted sampling approach to improve the overall efficiency of the assessment process. However, ONR assessed additional SSCs by using a Technical Support Contractor (TSC) to provide additional assessment resource.

4.21.2 Assessment strategy

641. During step 3 of GDA, ONR selected twenty-three of Hitachi-GEs mechanical engineering SSC BoSCs for assessment by its TSC. ONR raised a series of Regulatory Queries (RQs) and issued these to Hitachi-GE to capture the TSC's observations and findings. To ensure consistency and proportionality, ONR reviewed a sample of the TSC's observations and findings. ONR concurred with the TSCs observations and findings for this sample. The RQs for the five systems sampled by ONR are as follows;

- Primary Containment Isolation System - RQ-ABWR-0651 [164]
- Reactor coolant pressure boundary leak detection system - RQ-ABWR-0650 [165].
- Primary Containment Vessel gas control system RQ-ABWR-0656 [166]
- Reactor Building Cooling Water System RQ-ABWR-0615 [167].
- Standby Liquid Control system RQ-ABWR-0648 [168]

642. Hitachi-GE responded to each of the RQs and updated each of its BoSCs at Step 4 to address the queries raised. The responses to these RQ's and updated BoSC references are listed below:

Primary Containment Isolation System;

- RQ-ABWR-0651 response [169]
- Step 4 BoSC submission [170]

Reactor coolant pressure boundary leak detection system;

- RQ-ABWR-0650 response [171]
- Step 4 BoSC submission [172]

Primary Containment Vessel (PCV) gas control system;

- RQ-ABWR-0656 response [173]
- Step 4 BoSC submission [174]

Reactor Building Cooling Water System (RCW);

- RQ-ABWR-0615 response [175]
- Step 4 BoSC submission [176]

Standby Liquid Control system;

- RQ-ABWR-0648 [177]
- BoSC submission [178]

643. During step 4 GDA, I assessed Hitachi-GEs Step 4 BoSC submissions against the output of the TSCs Step 3 assessment and Hitachi-GEs response to the RQs.

644. Throughout step 4, ONR has engaged with Hitachi-GE through various level 4 engagements. During these engagements, ONR has challenged Hitachi-GE on the shortfalls identified by the TSC. I am now satisfied that Hitachi-GE's revised BoSC provides sufficient evidence to address the TSC's observations and findings.

4.21.3 Equipment description

645. The following paragraphs give a brief description of each of the SSCs sampled.

Primary containment isolation system

646. The primary containment isolation system isolates the systems that pass through the primary containment, thereby forming a barrier to contain any radioactive material and prevent release to the external environment. The isolation devices, pipework and containment vessel form the primary containment vessel boundary. The primary containment isolation system consists of the isolation devices (typically globe, gate, check and solenoid valves) and the controls required for the isolation of all pipework penetrating the reactor containment.

Reactor coolant pressure boundary leak detection system

647. The reactor coolant pressure boundary leak detection system detects any leakage from the reactor coolant systems. If permissible leakage limits are exceeded an alarm is initiated and the system is isolated automatically to achieve a safe condition. The main structures, systems and components (SSC) considered were the piping and valves associated with a number of measuring devices as follows:

- Temperature measurement instruments
- Pressure measurement instruments
- Flow meters
- Radioactivity meters

Primary containment vessel gas control system

648. The primary containment vessel gas system controls the atmosphere (pressure and composition) inside the primary containment vessel. This system ensures that adverse conditions do not threaten vessel integrity under normal and fault conditions. A series of valves open and close in response to signals from sensors to achieve the safety function. Some of the key system safety functions are listed below:

- Inerting – The system establishes an inert nitrogen atmosphere during plant start-up to reduce the oxygen concentration to no greater than 4.0% by volume. This may occur for example following refuelling or shutdown
- Pressure Control – During power operation, a slightly positive pressure differential is maintained inside the primary containment vessel to prevent ingress of air into the inert containment vessel atmosphere;
- De-Inerting – During shutdown or refuelling outages, the system replaces the primary containment vessel nitrogen atmosphere with air to allow safe access for personnel
- Nitrogen supply - The system supplies nitrogen gas to charge the safety relief valve accumulators;
- Pressurisation of the primary containment vessel for tests - The system supplies nitrogen gas to support the initial structural integrity test and the periodic integrated leak rate test.

Reactor building cooling water system

649. The reactor building cooling water system supplies cooling water to safety equipment within the reactor building. These include the Safety Class 1 systems for core cooling and long-term heat removal. The system recirculates cooling water through each SSC it serves via a loop containing a heat exchanger and circulation pump.

Standby liquid control system

650. The standby liquid control system automatically shuts down the reactor and maintains sub-criticality if there is a loss of the control rod insertion function. It achieves this by injecting a neutron absorbing solution (sodium pentaborate) into the core to provide sufficient negative reactivity. This shuts down the reactor in a safe manner from full power operation to cold shutdown. The system consists of tanks, pumps, piping and valves.

4.21.4 Technical Support Contractor (TSC) assessment of Hitachi-GEs Step 3 BoSCs.

651. ONR's TSC assessed the BoSCs submitted at GDA Step 3 against ONR SAP's and RGP. The TSC's findings identified shortfalls in the evidence presented by Hitachi-GE as detailed in the RQs listed above. The TSC identified a number of minor SSC specific shortfalls. In particular, the TSC identified common shortfalls across all BoSCs. These shortfalls represented potential issues with Hitachi-GEs underpinning design process. I considered the following overarching themes to be significant so these formed the basis for my assessment sample;

- Further justification for identification of system failure mechanisms;
- Insufficient claims and arguments to demonstrate that the design will meet its allocated safety functions;
- System reliability requirements to be set out at Step 4;
- Insufficient evidence presented to demonstrate that SSC designs will comply with UK expectations;

4.21.5 ONR assessment of Hitachi-GEs Step 4 BoSC.

652. At GDA Step 4 Hitachi-GE implemented an improved design process [18]. I consider the improved process to align with ONR SAP EDR.1 which states that due account should be taken for the need for SSCs to be designed to be inherently safe using a formal analysis. As a result Hitachi-GE's Step 4 BoSCs have addressed the following areas;

- The safety functions of each SSC have been derived using a structured analysis and can be linked back to the UK ABWR fault schedules. This aligns with ONR expectations as set out in SAP EKP.4, which states that the safety function(s) to be delivered within the facility should be identified using a structured analysis;

- Hitachi-GE has assigned its SSCs with safety categorisations which align with ONR SAP ECS.2 which states SSCs that have to deliver safety functions should be identified and classified on the basis of those functions and their significance to safety;

- Hitachi-GE is proposing a probability of failure on demand reliability of 10^{-4} for Class 1 components and 10^{-3} for Class 2 [143]. I consider this to align with the ONR TAG on categorisation and classification [179] which sets out target reliability figures for Class 1 and 2 SSCs;

- Hitachi-GE's Step 4 safety case, and in particular their list of applicable legislation and standards [142], identifies the general codes and standards applicable to the safe design of the SSC's discussed in this section. ONR SAP ECS.3 states SSCs important to safety should be designed to appropriate codes and standards. I consider Hitachi-GEs Step 4 safety case to satisfy this expectation.

4.21.6 Conclusions

653. I am satisfied that my assessment strategy for the SSCs sampled in this section has enabled me to make appropriate judgements on the adequacy of Hitachi-GE's BoSCs. My assessment has satisfied me that Hitachi-GE's Step 4 BoSCs provide sufficient evidence to satisfy the TSC's key observations and findings.

654. I consider that Hitachi-GE has now developed a suitable design process that has addressed shortfalls, previously identified by the TSC. I am satisfied that Hitachi-GE now has an adequate process to consider normal operating and potential fault conditions.

655. I recognise that further development of the system designs will occur during the detailed design phase. My expectation is that the responsibility to develop the system safety cases, to include adequate arrangements in respect of licence conditions, will rest with any future licensee.

4.21.7 Regulatory Findings and Shortfalls

656. None

4.22 Residual Heat Removal System

4.22.1 Introduction

657. The residual heat removal system (RHRS) comprises three redundant systems. Each system includes the necessary pipework, pumps, valves and heat exchangers required to deliver its operational and safety functions.

658. The safety roles of the system are to remove decay heat from the reactor during normal shutdown operation and to provide cooling under fault conditions. The residual heat removal system also has auxiliary functions including cooling the spent fuel storage pool.

4.22.2 Overview of Hitachi-GE Safety Case for Residual Heat Removal System

659. Hitachi-GE produced a Basis of Safety Case document for the RHRS [122]. The Pre-Construction Safety Report (PCSR) summarises the case presented in the Basis of safety case documents. The basis of safety case for the residual heat removal system is supported by several topic reports that present detailed analysis and evidence on specific topics.

4.22.3 Assessment Strategy for Residual Heat Removal System

660. At Step 4, I assessed the residual heat removal heat exchanger. I targeted my assessment on this component because of its significance to safety. I used a Technical Support Contractor (TSC) to review a sample of the Hitachi-GE submissions for the residual heat removal heat exchanger. The objective of this was to review the adequacy of the evidence contained within the BoSC submissions and topic report for the residual heat removal heat exchanger, from a mechanical engineering perspective. This built upon the work already carried out by the TSC at step 3, including determining whether, during step 4, Hitachi-GE had adequately addressed the findings and observations identified at step 3.

661. The main documents considered by the TSC were the BoSC for the residual heat removal system [122], the topic report for the residual heat removal heat exchanger [180] and the response to RQ-ABWR-0617 [181]. Hitachi-GE has presented the evidence supporting the safety case in references to these documents. The TSC reviewed a sample of the supporting references to assess the adequacy of this evidence at step 4. The TSC listed their findings in a report [84] and I oversaw the TSC's work through regular technical and progress meetings and by reviewing their final report. I am content that the review completed by the TSC meets my expectations.

4.22.4 Residual Heat Removal System Equipment Description

662. The RHRS provides a means of removing heat from the reactor pressure vessel, spent fuel storage pool and the suppression pool. There are three independent RHRSs, which provide redundancy. Each of these systems has a pair of dedicated residual heat removal heat exchangers. The Residual Heat Removal heat exchangers transfer heat from the tube (hot) side to the shell (cold) side in all operating conditions. The shell side of the heat exchanger is fed by the Reactor Building Cooling Water system.

4.22.5 Residual Heat Removal System Safety Functions

663. The residual heat removal heat exchanger fulfils a role in supporting safety functions relating to heat transfer for the residual heat removal system and to maintain the integrity of the pressure boundary. These are identified in the Topic Report [18] In consideration of ONR SAP EKP.4 – Safety functions, I am satisfied that Hitachi-GE has identified the safety function(s) to be delivered using a structured analysis.

664. The residual heat removal heat exchanger supports both Category A (function that plays a principal role in nuclear safety) and Category B (function that make a

significant contribution to nuclear safety) safety functions. The residual heat removal heat exchanger is therefore designed to meet Class 1 (forms a principal means of fulfilling a category A function) requirements. Hitachi-GE's categorisation and classification is based on its process for categorisation of safety functions and classification of SSCs [73]. I sampled the categorisation of the safety functions and classification of the residual heat removal heat exchanger and I am satisfied that it is appropriate and is in line with SAPs ECS.1 – Safety categorisation and ECS.2 – Safety classification of structures, systems and components.

665. The TSC sampled Hitachi-GE's evidence supporting its safety claims and arguments at step 4. It concluded that this was acceptable and sufficient for step 4, although further evidence is required during detailed design. Based on my review of the TSC's report, I am satisfied that this can be developed during detailed design.

4.22.6 Assessment of Residual Heat Removal System Design Process

666. Hitachi-GE has developed the design of the residual heat removal heat exchanger from the J ABWR design that has a proven operating record. Hitachi-GE has considered alternative designs (e.g. plate type) but it has selected a development of the shell and tube design. Hitachi-GE has increased the heat removal capacity for UK ABWR by providing a second unit mounted on the shell of the original unit to minimise layout impact. The UK ABWR tube plate is welded to the shell and header to reduce maintenance compared with the flanged joint used on J ABWR that requires periodic replacement of gaskets.

4.22.7 Residual Heat Removal System Codes and Standards

667. ONR SAP ECS.3 – Codes and standards, states that SSCs that are important to safety should be designed, manufactured, constructed, installed, quality assured, maintained, tested and inspected to the appropriate codes and standards. For the residual heat removal heat exchanger, Hitachi-GE has proposed to apply the ASME III code. I am satisfied that this code is appropriate and is commensurate with the safety classification of residual heat removal heat exchanger.

668. Hitachi-GE has applied a different ASME code class to the tube side compared with the shell side of the heat exchanger. I queried this with Hitachi-GE [182]. In response, [183] Hitachi-GE explained the process it has followed to select the ASME code class for the residual heat removal heat exchanger. This is based on the approach defined in US Nuclear Regulatory Commission Guide 1.26 [184]. UK understanding of this approach is summarised in the ONR Technical Advice Guide on categorisation of safety functions and classification of structures, systems and components [37]. Based on the response from Hitachi-GE [183] and ONR guidance [37], I am content with Hitachi-GE's approach for safety classification of the residual heat removal heat exchanger.

4.22.8 Redundancy, Diversity and Segregation for the Residual Heat Removal System

669. ONR SAP EDR.2 – Redundancy, diversity and segregation, requires redundancy, diversity and segregation to be incorporated as appropriate within the designs of structures, systems and components. Hitachi-GE's Topic Report on Mechanical SSC Architecture [121] sets out how it has incorporated these aspects into the design of the residual heat removal system.

670. **Redundancy** is achieved by the provision of three RHRSs. Each system is capable of fulfilling the safety function of residual heat removal following reactor shutdown. To achieve this, Hitachi-GE has increased the heat removal capacity of the residual heat removal system compared with the J-ABWR design [122].

671. **Segregation** is achieved through physical separation of the three RHRSs.

672. **Diversity** is provided by the flooder system of the specific safety facility. This provides a diverse means of long-term heat removal for 'frequent' design basis faults. I sampled the flooder system of the specific safety facility as part of my assessment of the UK ABWR severe accident mechanical systems. I have presented these findings in an earlier section of this report.

673. Based on the information provided by Hitachi-GE in references [122] and [121], I am satisfied that the provision for redundancy, diversity and segregation for the residual heat removal system is in line with ONR SAP EDR.2 – Redundancy, diversity and segregation.

4.22.9 Qualification of the Residual Heat Removal System

674. ONR SAP EQU.1 – Qualification procedures, requires qualification procedures to be applied to confirm that structures, systems or components will perform their allocated safety functions in all normal operational, fault and accident conditions for the duration of their operational lifecycle. Hitachi-GE has developed an equipment qualification (EQ) plan to ensure that the functions specified in the design specification are adequately met. The TSC reviewed the EQ plan and concluded that it presented adequate arguments for step 4, but that the evidence can only be assessed when the specified qualification activities are completed. Based on my review of the TSC's report [84], I am satisfied that this can be completed following detailed design.

4.22.10 Examination, Inspection, Maintenance and Testing of the Residual Heat Removal System

675. At step 3, ONR raised RQ-ABWR-0647 [185] to query the provisions for EIMT for the residual heat removal heat exchanger. The basis of safety case [122] and topic report [180] outline the EIMT provisions for the residual heat removal heat exchanger.

676. Hitachi-GE proposes visual inspection at regular intervals and replacement of gaskets every five years. Hitachi-GE does not propose internal examination of the residual heat removal heat exchanger during operation. ONR queried this at a technical workshop [186]. Hitachi-GE explained that this approach is in line with ASME XI requirements and supports UK OPEX from operation of Pressurised Water Reactors (PWR). Should inspection be deemed necessary all welds can be accessed via confined space working. Hitachi-GE indicated that replacement, repair and tube plugging are all viable and can be carried out under confined space working conditions. I have noted that the heat exchanger unit will have detectors for contaminated water. However as the unit is only used during shutdown when contamination levels are low I do not consider that this will have a significant effect on EIMT. My expectation is that provision for EIMT will be developed during detailed design to address ONR SAPs EMT.1 – Identification of requirements and EMT.2 - Frequency.

677. For replacement of gaskets, Hitachi-GE provided a walkthrough [186] of its optioneering study to provide safe access for this task. This is summarised in the topic report [180]. I do not consider this optioneering conclusive. However, I am content that Hitachi-GE is proposing a permanent access platform for GDA and I satisfied that the detailed design could be developed outside of GDA such that no options are foreclosed at this stage. My expectation is that this solution will be developed during detailed design in line with ONR SAP ELO.1 – Access. I recognise that further considerations for plant layout will occur during detailed design. My expectation is that responsibility for making and implementing adequate EIMT arrangements in respect of licence conditions will rest with the licensee. I have raised a regulatory assumption to capture this expectation, which Hitachi-GE should communicate to any future licensee.

678. Hitachi-GE claims that it has provided sufficient margin to enable plugging of 1% of tubes without compromising the required heat exchanger performance. Replacement of heat exchanger tube bundles is not possible on the UK ABWR design (in contrast with the J-ABWR). Complete heat exchanger module replacement would be required should the limit of tube failures be reached. Hitachi-GE has outlined a plan for module replacement that the TSC reviewed. I consider this is acceptable for step 4 GDA and that this can be developed by the licensee during detailed design.

4.22.11 Conclusions

679. I am satisfied with the categorisation of the safety functions for the residual heat removal system and classification of the residual heat removal heat exchanger. I am also satisfied that Hitachi-GE has provided adequate evidence to demonstrate that the residual heat removal heat exchanger will deliver its safety functions. I am satisfied additional evidence will be available during detailed design. I am content that the codes and standards selected for the residual heat removal heat exchanger are appropriate.

680. I am satisfied that Hitachi-GE has considered access for EIMT in its safety case submission. I have identified an assumption that detailed design adequately considers access EIMT activities.

4.22.12 Regulatory Assumptions

681. I have identified one assumption which is applicable to the residual heat removal system:

- **AS-ABWR-ME05 -** ONR acknowledges that during GDA, Hitachi-GE has not sufficiently developed its design to enable the licensee to identify the appropriate EIMT requirements in line with UK expectations. ONR assumes that the licensee shall ensure that they consider EIMT requirements for all mechanical engineering items, which attract a safety classification. ONR also assumes that the licensee shall establish whether these requirements are directly driven by the safety case, are based on manufacturer recommendations or are based on plant operating experience (or appropriate combinations).

4.23 Off Gas System

682. The off gas system is designed to maintain a vacuum in the main condenser by extracting non-condensable gases. The system also provides abatement of radioactive material prior to atmospheric discharge and recombines the radiolytic hydrogen and oxygen generated in the reactor. The off gas system reduces the radiological release from the UK ABWR during normal operation of the reactor.

4.23.1 Overview of Hitachi-GE Safety Case for the Off Gas System

683. PCSR chapter 18 [187] presents the safety case for the off gas system, this summarises the case presented in the Basis of Safety Case (BoSC) for the off gas system [188]. The Topic Report on ALARP assessment for the off gas system [[189] presents the arguments and summary of evidence that the off gas system reduces risks ALARP. The technical supporting document [190] provides the detailed evidence to support the demonstration of ALARP for the off gas system.

4.23.2 Assessment Strategy for the Off Gas System

684. ONR raised RO-ABWR-0073 [191] to state its expectations with respect to Hitachi-GE producing a robust demonstration that the design of the UK ABWR off gas system reduces risks SFAIRP. References [188], [189] and [190] present Hitachi-GE's response to this RO. I have sampled Hitachi-GE's response to this RO from a

mechanical engineering perspective. I considered that the hydrogen generated by radiolysis during reactor operation could foreseeably be a significant hazard. I therefore targeted my assessment on the hydrogen management functions of the off gas system during normal operation and following an off gas system fault.

685. A Technical Support Contractor (TSC) assessed the off gas system during step 3 GDA. Hitachi-GE presented its responses to the TSC's comments at step 3 in the response to RQ-ABWR-0614 [192]. This RQ required Hitachi-GE to:

- familiarise itself with the report findings and observations,
- confirm the report factual accuracy;
- prepare to discuss the findings, observations and expectations in detail as part of the planned mechanical Engineering technical workshops; and
- in advance of the planned technical workshops, develop and advise its strategy to address the findings, observations and expectations.

686. Throughout step 4 ONR has engaged with Hitachi-GE through various level 4 engagements and I am satisfied that Hitachi-GE has adequately discussed the TSC's findings during these engagements. Through my assessment during step 4, I am satisfied that Hitachi-GE's final versions of the BoSC now provide sufficient evidence to satisfy the TSC's initial observations and findings.

4.23.3 Equipment Description for the Off Gas System

687. During normal power operation, the off gas system extracts non-condensable gases from the main condenser using steam driven air ejectors. The extracted gases pass through a pre-heater and into the hydrogen recombiner that Hitachi-GE has designed to recombine hydrogen and oxygen to prevent an explosive atmosphere. The extracted gases then pass through a condenser and a cooler into a charcoal absorber. This holds up the radioactive gases to minimise the release of radioactivity to the environment. The extracted gases pass through a HEPA filter before discharge to atmosphere.

688. Hydrogen analysers detect high hydrogen concentration should the hydrogen recombiners fail. This automatically closes the valves on the condenser air extraction, valves and the steam driven air ejector driving steam isolation valve.

689. During reactor start-up, the off gas system creates the required vacuum in the main condenser. This is achieved through a mechanical vacuum pump and by the start-up air ejectors. The start-up air ejectors are also used when the reactor is shut down to purge radioactive contamination from the main condensers. This must be done before opening the main condensers during a refuelling outage. The off gas system includes the required instrumentation and controls.

4.23.4 Defence in Depth for the Off Gas System

690. ONR SAP EKP.3 – Defence in depth recommends that nuclear facilities are designed and operated so that defence in depth against potentially significant faults or failures is achieved by the provision of multiple independent barriers to fault progression.

691. To reduce the risk from hydrogen explosion, the off gas system incorporates measures [188] to:

- Recombine hydrogen and oxygen generated during normal operation using redundant parallel hydrogen recombiners.
- Detect an increase in hydrogen concentration caused by off gas recombiner failure and automatically shut the off gas system isolation valves.
- Prevent hydrogen combustion by eliminating ignition sources.

692. In the event that a hydrogen explosion does occur, the off gas system is designed to withstand the pressure increase and thereby prevent radioactive release. I am therefore satisfied that the design of the off gas system offers defence in depth and therefore is in line with ONR SAP EKP.3 – Defence in depth.

4.23.5 Safety Functions and Equipment Classification for the Off Gas System

693. The off gas system has the following safety functions which are listed in the BoSC [188]:

- Minimisation of doses to the public during normal conditions
- Minimisation of doses to workers during normal conditions
- Hydrogen management during normal conditions
- Off-gas extraction during normal conditions
- Minimisation of doses to public from off gas system fault conditions
- Minimisation of doses to worker during off gas system fault conditions
- Hydrogen management during off gas system fault conditions

694. In consideration of ONR SAP EKP.4 – Safety function, I am satisfied that Hitachi-GE has identified the safety function(s) to be delivered using a structured analysis.

695. I sampled the categorisation of functions delivering hydrogen management under normal operation and during off gas system fault conditions. Hitachi-GE has identified the safety function of delivering hydrogen management during normal operation as Safety Category C (function that contributes to nuclear safety). The main components delivering this function are designed to meet Safety Class 3 requirements:

- Start-up Steam Jet Air Ejector
- Steam driven Air Ejector
- Steam driven Air Ejector Condenser
- Off Gas Preheater
- Off Gas Recombiner

696. Delivering hydrogen management during off gas system fault conditions is a Category B safety function (function that makes a significant contribution to nuclear safety). The components delivering this function are designed to Class 2 (makes a significant contribution to fulfilling a category A safety function) requirements [188]:

- Condenser Air Extraction Valves
- Steam driven Air Ejector Driving Steam Isolation Valve
- Off Gas Hydrogen Analyser at Off Gas Cooler Condenser outlet

697. The ONR TAG on categorisation of safety functions and classification of SSCs [37] provides guidance on prevention versus protection. This states that in line with the approach to defence in depth, the focus should be on preventing a fault occurring and thereby limiting the demand placed on protection systems. The integrity of the systems delivering preventative safety functions should not therefore be automatically lowered simply because a safety system delivering a protective function exists. ONR challenged Hitachi-GE on this issue and raised RQ-ABWR-1415 [193] to query:

- The safety classification of off gas system normal operation SSCs, and
- Justification for cases where off gas system protective SSCs carry a higher safety classification than preventative SSCs.

698. In response to RQ-ABWR-1415, Hitachi-GE provided [194], which justified why it was not reasonably practicable to provide a higher safety category for the structures, systems and components (SSC) that prevent a hydrogen explosion.

699. Guidance in ONR TAG 094 [37] recognises that in some cases, it is not practicable for the normal operation system to carry a high safety class and is it is appropriate for this to be reduced in favour of increasing the class of a protective safety system. From a mechanical engineering perspective, I consider that the categorisation of safety functions and classification of the off gas system SSCs is appropriate and aligns with ONR SAPs ECS.1 – Safety categorisation and ECS.2 – Safety classification of structures, systems and components.

4.23.6 Assessment of Off Gas System Pipework

700. During off gas system operation, the off gas stream is diluted using steam from the steam driven air ejector. Redundant parallel recombiners convert hydrogen and oxygen into water vapour, and ignition sources are eliminated. Additionally, hydrogen detection is installed where necessary in the off gas system [189]. To prevent a hydrogen explosion leading to a release of radioactivity in the event of failure of the normal operation functions, Hitachi-GE claims [188] that the off gas system piping from the main condenser to the off gas charcoal absorber outlet is designed to withstand the pressure increase caused by a hydrogen explosion . These components are classified by Hitachi-GE as class 3.

701. Appendix G of the technical supporting document on the off gas system ALARP report [190] presents the evidence available at GDA to support the claims made on the structural integrity of the off gas system. In my opinion, further evidence is required to substantiate the claim that a straight length of pipe in itself is able to withstand the maximum peak pressure from a detonation. In addition, the safety case does not present analysis of the additional implications from a contained explosive event. In particular:

- No analysis has been carried out against convergent or divergent pipework design features which could enhance the peak pressure resulting in the estimated maximum pressure in a straight length of pipe being exceeded;
- The path of the shockwave and its impact on other SSCs within the system has not been adequately analysed.

702. My assumption is that the licensee shall ensure that their detailed design includes analysis of convergent or divergent pipework design features that could enhance the peak pressure during such an event. I also that the path of the shockwave generated by a hydrogen detonation in the off gas system, and its impact on other SSCs will be analysed.

703. Hitachi GE has considered whether they can make any improvements to reduce risks ALARP [189]. It proposes to design the off gas system pipework to ASME Section III, rather than ASME Section VIII, which would usually be applied to a class 3 SSC. It claims that this will achieve a reliability of 10^{-4} to 10^{-5}/yr., which is a higher level of reliability than I would normally expect for a class 3 SSC. Hitachi-GE has recognised that this will require additional testing and inspection of these SSCs but has not identified these requirements in the safety case at step 4. I have captured this in an assumption (AS-ABWR-ME08).

4.23.7 Off Gas System Conclusions.

704. Based on my mechanical engineering assessment, I do not consider that Hitachi-GE has provided adequate evidence for the claim that off gas system pipework can withstand a hydrogen explosion. My assumption is that the licensee will consider this during detailed design.

705. I am satisfied that Hitachi-GE has considered EIMT of the off gas system pipework in its safety case submission. I assume that this will be developed by the licensee during detailed design.

4.23.8 Regulatory Assumptions – Off Gas System

706. I have identified two assumptions which are applicable to the off gas system:

- **AS-ABWR-ME05 -** ONR acknowledges that during GDA, Hitachi-GE has not sufficiently developed its design to enable the licensee to identify the appropriate EIMT requirements in line with UK expectations. ONR assumes that the licensee shall ensure that they consider EIMT requirements for all mechanical engineering items, which attract a safety classification. ONR also assumes that the licensee shall establish whether these requirements are directly driven by the safety case, are based on manufacturer recommendations or are based on plant operating experience (or appropriate combinations).

- **AS-ABWR-ME08-** during GDA, Hitachi-GE did not provide sufficient evidence to demonstrate that the off gas system pipework is qualified to withstand a hydrogen detonation event. ONR assumes that the licensee shall ensure that their detailed design includes analysis of convergent or divergent pipework design features, which could enhance the peak pressure during such an event. ONR also assumes that the path of the shockwave generated by a hydrogen detonation in the off gas system, and its impact on other SSCs will be analysed.

4.24 Use of Stellite™ within Mechanical Equipment

4.24.1 Introduction

707. The transport of cobalt atoms into fluid systems through either wear, maintenance dressing of sealing surfaces, or corrosion, is a known problem in nuclear power plants. This can lead to high worker dose rates through the activation of cobalt due to neutron flux within the primary circuit. However, Stellite™, a cobalt chromium alloy, has good mechanical material properties; for example, impact, wear and galling resistance. This leads to its use in systems structures or components that are subject to an onerous mechanical duty. The majority of the Stellite™ present in the current UK ABWR concept design is in various components of the fine motion control rod drives with the remainder used in valve seats. Many cobalt-free alloys have inferior mechanical properties to Stellite™.

4.24.2 Assessment of Use of Stellite™ within Mechanical Equipment

708. Reducing the inventory of potentially harmful substances to a minimum assists in achieving an inherently safe design. In my judgement, this can be achieved by avoiding or reducing the use of Stellite™ compared to some of the earlier BWR designs, so far as is reasonably practicable.

709. Hitachi-GE has gone some way to reduce the use of Stellite™ in the J-ABWR and, hence, the UK ABWR design. For example, nickel-based alloys and iron-based alloys have been successfully qualified for use as a replacement material for Stellite™ in some components thus eliminating their contribution to the cobalt source term. Reference [195] presents a case to justify that, while some Stellite™ remains in the UK ABWR design, its use is minimised SFAIRP.

710. ONR, through RQ-ABWR-1339 [196] sought further assurance that Hitachi-GE has limited the use of Stellite™ in the UK ABWR design SFAIRP. The response to RQ-ABWR-1339 [197] states that identification of replacement materials may be possible in the future. This requires extensive testing and qualification but I consider this

sufficient for this stage of the UK ABWR design process. However, given the significance of this potential contributor to operator dose I would expect the future licensee to provide a further ALARP justification during detailed design for the use of cobalt containing alloys down to component level. From a mechanical engineering perspective, I consider that this ALARP justification should focus on the mechanical properties of the end component. It should demonstrate that in replacing the material of a component there is no detrimental effect to the safety and reliability of the system structure or component that cannot be managed by other means.

4.24.3 Conclusion

711. Hitachi-GE has already gone some way to reduce the use of Stellite™ in the J-ABWR, and hence the UK ABWR. In addition, in response to RQ-ABWR-1339 [196] its states that identification of replacement materials may be possible in the future, subject to testing and qualification. I consider this sufficient for a concept design. However, I consider further ALARP justification is required for the use of cobalt containing alloys down to component level by any future licensee during detailed design. I have captured this as a GDA assumption (AS-ABWR-ME10)...

4.24.4 Regulatory Assumptions

712. Assumption relating to the use of high-cobalt containing materials.

> **AS-ABWR-ME10 -** Hitachi-GE has indicated that it will eliminate high-cobalt containing alloys from UK ABWR where reasonably practicable. However, Hitachi-GE has not indicated whether this could have a detrimental effect to the safety and reliability of individual components. ONR assumes that the licensee shall ensure that they consider these effects during detailed design and provide suitable evidence to demonstrate that there is no detriment to the safety and reliability of the SSC.

4.25 Traversing In-core Probe (TIP) explosive shear valve

4.25.1 Introduction

713. The Traversing In-core Probe (TIP) system provides a means of calibrating the local power range monitors and measurement of the axial neutron flux distributions in the reactor core. The probe is located at the end of a cable wrapped with helical wire inserted into and withdrawn from the core by a drive mechanism. The TIP system penetrates the primary containment pressure boundary and when not inserted into the core a Safety Class 1 ball valve isolates the pressure boundary to confine radioactive material in the core and prevent its dispersion to the environment. If the probe becomes snagged in the core, the cable penetrating the containment pressure boundary must be cut to release it, and the probe, into the core thus allowing the ball valve to provide its isolation function. Hitachi-GE intends to use an explosively actuated shear valve as a means of cutting the cable.

714. The explosively actuated shear valve contains Zirconium Potassium Perchlorate (ZPP) as its propellant and is located approximately 1 metre away from the primary containment vessel. Hitachi-GE claims the temperature at this location to be up to 40°C in normal operating conditions and up to 100°C in abnormal conditions. Hitachi-GE has claimed the reliable service life of the valve as being five years.

715. The shear valve is assigned safety function categorisation C and safety classification 3.

4.25.2 Assessment of Traversing In-core Probe (TIP) explosive shear valve

716. During my GDA assessment, I chose to assess the TIP shear valve to ascertain whether Hitachi-GE had given sufficient consideration within GDA to the hazards presented by Class 3 SSCs as was to Class 1 and 2 SSCs and their potential to affect other Class 1 and 2 SSCs.

717. I consider ONR SAPs of particular relevance to the use of a shear valve to be;

- ONR SAP EHA.13 (external and internal hazards): the use, storage and generation of hazardous materials should be minimised, controlled and located, taking due account of potential faults. Principle EKP.1 (Inherent safety) is relevant here and should lead to designs that seek to eliminate the hazard or use less hazardous substitutes.
- SAP Para 167 is also relevant and states; appropriately designed interfaces should be provided between (or within) SSCs of different classes to ensure that any failure in a lower class item will not propagate to an item of a higher class. Equipment providing the function to prevent the propagation of failures should be assigned to the higher class.

718. I consider the potential safety risks associated with this valve to be;

- If not adequately qualified for its intended use the explosive charge may not reliably actuate the Class 3 shear valve. This would leave the TIP inserted into the core preventing the Class 1 ball valve from isolating the primary containment;
- Spurious actuation of the valve would result in non-availability of the TIP system to calibrate the local power range monitors and measure the axial neutron flux distributions in the core;
- Spurious actuation of the valve from a sensitised explosive presents a risk to workers particularly during EIMT operations when the explosive actuator will be physically tested or handled.

719. The focus of my assessment was to seek evidence that the valve type (explosively actuated) reduces risk SFAIRP, the valve is adequately qualified, will reliably function and a suitable EIMT regime exists.

720. Through RQ-ABWR-0591 [198], I sought assurance that;

- Optioneering: Adequate optioneering had been carried out prior to selecting the valve for use in the UK ABWR application;
- Qualification: The ZPP propellant had been adequately qualified for its intended application;
- EIMT: A suitable EIMT regime had been developed that meets with UK RGP for explosives handling and processing.

721. Hitachi-GEs RQ-ABWR-0591 response [199] provided the following detail;

- Optioneering: In [199], attachment B presents the optioneering result for the TIP shear valve. This identifies alternative design options to the explosive shear valve that eliminate the explosive hazards; however these are discounted on the basis that a change to the baseline design would be required. Furthermore, Hitachi-GE has not justified its design basis requirement for a fast acting explosively driven shear valve over other options that can reliably provide the same function by less hazardous means. I do not consider Hitachi-GEs optioneering result to have sufficiently justified that the explosive shear valve reduces risk SFAIRP. However, from subsequent discussions with Hitachi-GE I am satisfied that they have considered the options in sufficient detail. This detail is sufficient to inform a future licensee who should satisfy themselves that this type of valve reduces risks ALARP. Therefore, during a workshop in 2016, ONR indicated that it was satisfied with the proposed conceptual design [113].

- Qualification: Material characterisation data for the propellant (ZPP) was requested in [198]. In [199] Hitachi-GE stated, “the powder used for this valve is a development product by our supplier, it is not based on any public standards”. Any material, particularly an explosive, must be well characterised prior to service use to understand how it reacts to a range of credible insults, any limitations and any cliff edge effects. Explosive compounds can deteriorate with age and this deterioration is accelerated by higher temperature. In my judgement, Hitachi-GE has not presented sufficient evidence to demonstrate that the explosive has been adequately qualified to reliably deliver its safety function at sustained temperatures above ambient over the claimed five year service life.

- EIMT: Detail of the safety precautions to be taken whilst operators carry out EIMT on the valve was requested in [198]. In [199], Hitachi-GE claims that an antistatic regime would be implemented. This meets with my expectations for preventing an electrostatic discharge insult to the explosive that could result in a spurious actuation whilst the valve is being handled. Further detail of the antistatic regime is to be provided during detailed design if an explosive shear valve is selected.

4.25.3 Conclusion

722. For a generic, concept design I consider Hitachi-GEs response in [2] to present high level claims that partially address the queries raised in RQ-ABWR-0591. However, Hitachi-GE has not presented adequate justification through its optioneering to justify that a fast acting explosively driven shear valve is the most suitable design option to reduce risks SFAIRP.

723. If Hitachi-GE concludes that an explosively driven valve is the ALARP design solution then it must be adequately qualified to demonstrate reliability at high temperature over the claimed five-year service life. Furthermore, during detailed design, the antistatic regime arrangements should be provided. I am satisfied that the shortfalls identified above can be addressed by the future licensee during detailed design and I have captured this as a GDA assumption (AS-ABWR-ME-09.

4.25.4 Regulatory Assumptions

724. Assumption relating to the Traversing In-core Probe (TIP) explosive shear valve.

- **AS-ABWR-ME09:** Hitachi-GE did not provide adequate evidence during GDA to demonstrate that a fast acting explosively driven valve (squib valve) reduces risks, so far as is reasonably practicable, when compared to other options. Furthermore, Hitachi-GE did not adequately demonstrate that the explosive charge had been suitably qualified to underpin the safety and reliability claims.

ONR assumes that the licensee shall ensure that they provide suitable optioneering and ALARP justification for the design, qualification to underpin the safety and reliability claims and evidence of an adequate EIMT regime.

4.26 Operational experience

4.26.1 Introduction

725. During step 3 GDA, ONR raised cross cutting RO-ABWR-0045 [200] on Operational Experience (OPEX). ONR raised this RO requesting that Hitachi-GE provide further evidence that the UK ABWR design has taken in to account relevant operational experience from around the world.

4.26.2 Assessment of Operational Experience

726. Specific mechanical engineering queries on OPEX were captured by a number of RQs as follows:

- RQ-ABWR-0253 Operational Experience Safety Relief Valves
- RQ-ABWR-0254 Operational Experience UK ABWR Control Rod Design
- RQ-ABWR-0255 Operational Experience Fine Motion Control Rod Drive Units
- RQ-ABWR-0262 Operational Experience Main steam Isolation valves
- RQ-ABWR-0837 Operational Experience Sentencing
- RQ-ABWR-0838 Transfer of Operational Experience
- RQ-ABWR-0843 Turbine Driven Auxiliary Feed Water Pump Sentencing

727. Hitachi-GE provided responses to each of these RQs and I am now content that its responses adequately satisfy ONR queries.

728. In response to RO-ABWR-0045 Hitachi-GE produced an operational experience report [98]. The OPEX report provides background information in to Hitachi-GEs review in to operational experience review in which several international organisations detailing operational experience were identified. Hitachi-GE also demonstrated that information from international organisations is being adequately collected and stored on an internal database. I am content that the Hitachi-GE's internal operational experience database adequately sentences incidents and indicates how lessons learned are applied to the UK ABWR design. I consider that Hitachi-GE's response adequately addresses RO-ABWR-0045 and therefore a closure letter was issued [201]

4.26.3 Conclusions for Operational Experience

729. I am satisfied that Hitachi-GE has adequate processes to capture operation experience and apply lessons learned.

4.26.4 Regulatory Findings and Assumptions

730. None

5 REGULATORY OBSERVATIONS

731. Regulatory Observations (ROs) are raised when ONR identifies a potential regulatory shortfall, which requires action and further justification by requesting parties for it to be resolved. Each RO can have several associated actions.

732. A summary of ROs related to my mechanical engineering assessment of Hitachi-GE's UK ABWR can be found in Annex 3

733. Information on the basis, evidence and closure of ROs has been covered in the main body of this assessment

6 COMPARISON WITH STANDARDS, GUIDANCE AND RELEVANT GOOD PRACTICE

734. Comparison with standards, guidance and relevant good practice has been included as part of the main body of this assessment see section 4.4

7 ASSESSMENT FINDINGS

735. During my assessment, I identified matters for the licensee to take forward in its site-specific safety submissions. These matters do not undermine the generic safety submission and are primarily concerned with the provision of site-specific safety case evidence. This evidence will usually become available as the project progresses through the detailed design, construction and commissioning stages. I have captured these items as assessment findings and they are detailed in Annex 5.

736. I have recorded an assessment finding if one or more of the following apply:

- site specific information is required to resolve this matter;
- resolving 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.

737. Assessment Findings are residual matters that must be addressed by the Licensee and the progress of this will be monitored by ONR.

8 REGULATORY ASSUMPTIONS

738. ONR recognises that designers cannot establish certain design and operating information until detail design commences. Hence, my judgment of the generic design assumes that the future licensee does not overlook such information during detail design.

739. I have identified specific assumptions used to inform my judgment and indicated that Hitachi-GE should capture these assumptions and make provision to communicate these to the licensee.

9 CONCLUSIONS

740. This report presents the findings of my Step 4 mechanical engineering assessment of Hitachi-GE's generic design for the UK ABWR.

741. Hitachi-GE has presented adequate evidence during step 4 to support the claims and arguments it presented for mechanical systems during step 3. I am satisfied that the safety case was presented appropriately and I am satisfied that Hitachi-GE has provided sufficient opportunity for me to discuss and challenge evidence where necessary, with Hitachi-GE's relevant subject matter experts.

742. I am satisfied with the claims, arguments and evidence laid down within the PCSR and supporting documentation for mechanical engineering. I consider that from a mechanical engineering perspective, the Hitachi-GE UK ABWR design is suitable for construction in the UK subject to future permissions and permits beings secured .

743. I have identified several assessment findings (annex 5) that the licensee shall consider and take forward in its site-specific safety submissions. These matters do not undermine the generic safety submission and require licensee input/decision.

744. I have identified several assessment assumptions that Hitachi-GE shall record and communicate to the licensee to inform the detailed design.

9.1 Key Findings from the Step 4 Assessment

745. I consider that from a mechanical engineering viewpoint, the UK ABWR design is suitable for construction in the UK.

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[127] Hitachi-GE, “HGNE COMMERCIAL - UK ABWR - GA91-9201-0002-00042 - VDE-GD-0001 - Rev 3 - Basis of Safety Cases on Emergency Power Supply System - 30 June 2017”.

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[139] Hitachi-GE, “HGNE COMMERCIAL - UK ABWR - GA91-9201-00~PE-GD-H063 - Rev 0 - HVAC Calculation Report (Control Building) - 29 March 2017;”.

[140] Hitachi-GE, “HGNE COMMERCIAL - UK ABWR - GA91-9201-00-065 - Rev 0 - HVAC

Calculation Report (Heat Exchanger Building) - 29 March 2017;".

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[166] ONR, “RQ-ABWR-0656 - PCV Gas Control System - Russell Makinson - 14 October 2015”.

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[174] Hitachi-GE, “HGNE COMMERCIAL - UK ABWR - GA91-9201-00~-0196 - Rev 2 - Basis of Safety Cases on PCV Gas Control Systems - 04 July 2017”.

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[196] Hitachi-GE, “RQ-ABWR-1339 - Stellite Reduction in UK ABWR - Jeff Glover - 27 February 2017 TRIM: 2017/80407”.

[197] Hitachi-GE, “HGNE COMMERCIAL - UK ABWR - GA91-9201-0003-02107 - 1E-GD-5088 - Rev 0 - Stellite Reduction in UK ABWR (Response to RQ-ABWR-1339) - 03 May 2017 TRIM 2017/173686”.

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[199] Hitachi-GE, “HGNE COMMERCIAL - UK ABWR - GA91-9201-0003-01005 - KIE-GD-0002 - Rev 0 - Control and Instrument System - Explosively actuated shear valve design

disclosure, qualification and EIM&T (Response to RQ-ABWR-0591) - 29 October 2015".

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Annex 1

Safety Assessment Principles

SAP No	SAP Title	Description
FP series	Fundamental principles	FP.1 to FP.8
SC series	Safety cases	SC.1 to SC.8
EKP series	Key principles	EKP.1 to EKP.5
ECS series	Safety classification and standards	ECS.1 to ECS.5
EQU series	Equipment qualification	EQU.1
EDR series	Design for reliability	EDR.1 to EDR.4
EMT series	Maintenance, inspection and testing	EMT.1 to EMT.8
EAD series	Aging and degradation	EAD.1 to EAD.5
ELO series	Layout	ELO.1 to ELO.4
EHA series	External and internal hazards	EHA.1 to EHA.17
EPS series	Pressure systems	EPS.1 to EPS.5
ESS series	Safety systems	ESS.1 to ESS.27

EES series	Essential services	EES.1 to EES.9
ECV series	Containment and ventilation	ECV.1 to ECV.10
EHT series	Heat transport systems	EHT.1 to EHT.5
AM series	Accident management and emergency preparedness	AM.1

Annex 2

Technical Assessment Guide

TAG Ref	TAG Title
NS-TAST-GD-003 Revision 7	Safety Systems
NS-TAST-GD-005 Revision 8	Guidance on the Demonstration of ALARP (As Low As Reasonably Practicable)
NS-TAST-GD-056 Revision 3	Nuclear Lifting Operations
NS-TAST-GD-057 Revision 3	Design Safety assurance
NS-TAST-GD-084 Revision 10	Guidance on Production of Reports
NS-TAST-GD-085 Revision 6	Peer Review for Legal and Technical Assurance
NS-TAST-GD-094 Revision 0	Functions and Classification of Structures, Systems and Components

Annex 3

Regulatory Observations

RI / RO Ref	RI / RO Title	Date Closed	Report Section Reference
RO-ABWR-0015	Mechanical engineering SSC’s qualification and layout provision	31/01/2017	
RO-ABWR-0016	Mechanical Engineering Design Process	16/02/2017	4.1, 4.5, 4.17,4.18
RO-ABWR-0017	Nuclear Ventilation Codes and Standards	03/03/2017	4.17,4.18,4.21
RO-ABWR-0018	Examination, Inspection Maintenance and Testing (EIMT) Isolations and Configurations	24/11/2016	4.2, 4.17,4.18
RO-ABWR-0045	Mechanical Engineering –Operational Experience (OPEX)	31/03/2017	4.5.7,4.8.6,4.20.5,4.26.1,4.26.2
RO-ABWR-0047	Mechanical Engineering – Wet Lifting Beams – Material of Construction	22/12/2016	4.5
RO-ABWR-0049	Mechanical Engineering – Dropped Load Counter Measures	02/02/2016	4.5
RO-ABWR-0050	Mechanical Engineering - Crane Control Measures	22/12/2016	4.5
RO-ABWR-0051	Mechanical Engineering – SSCs Qualification	10/04/2017	4.17, 4.18, 4.22, 4.24
RO-ABWR-0052	Mechanical Engineering - Design Process - SSCs' Detailed Design	18/04/2016	4.1
RO-ABWR-0075	Robust demonstration that the design of the UK ABWR HVAC system has been adequately conceived and reduces risks SFAIRP	03/07/2017	4.17

Annex 4

GDA Issues

None

Annex 5

Assessment Findings

Assessment Finding Number	Assessment Finding	Report Section Reference
AF_ABWR_ME_001	Hitachi-GE has not provided sufficient evidence that equipment qualification plans meet UK expectation. In particular, the plans failed to demonstrate a suitable sample size and testing regime both at system and component level. Furthermore, Hitachi-GE had not identified test standards in its equipment qualification plans or applied UK test conditions and timescales commensurate with UK ABWR expected lifetime. To address these shortfalls, the licensee shall develop equipment qualification plans, which consider these issues.	4.16, 4.17, 4.20
AF_ABWR_ME_002	It is established relevant good practice in UK reactor safety cases to demonstrate that at least two diverse structures, systems or components (SSCs) are provided to deliver the key nuclear safety functions required after a frequent design basis fault, so far as is reasonably practicable. Hitachi-GE has made adequate arguments that, in some cases, it would be grossly disproportionate to provide fully diverse and independent design provision. In these cases, Hitachi-GE's intention is for licensees to consider other types of diversity, such as diverse manufacturing practices, and enhanced EIMT regimes during detail design. The licensee shall ensure that they consider these alternative methods of achieving diversity during detailed design and implement them wherever reasonably practicable.	4.10

AF_ABWR_ME_003	Hitachi-GE proposes to use high integrity cranes designed for nuclear use for lifting tasks associated with rotating incoming new fuel from horizontal to vertical. ONR considers that these lifts are complex and if incorrectly implemented could foreseeably lead to crane damage, crane collisions or dropped loads any of which could present a risk to nuclear safety. The licensee shall demonstrate, during detail design, that they have considered alternative methods for rotating fuel that do not involve using nuclear use cranes for complex lifting tasks. The licensee shall implement alternative methods wherever reasonably practicable.	4.6
AF_ABWR_ME_004	Hitachi-GE's design for UK ABWR eliminates routine EIMT for the FMCRD upper component. The licensee shall provide a suitable safety justification for this deviation from Japanese RGP during detailed design. The justification shall include evidence to support any decisions that the FMCRD upper component can be qualified to meet the 60-year design life.	4.20
AF_ABWR_ME_005	Hitachi-GE in its ALARP study identified that the design of the reactor internal pump upper plug does not meet UK RGP since it relies on a single isolation. The licensee shall develop the design of the reactor internal pump upper plug, meeting the identified requirements of UK relevant good practice. The concept design shall be qualified with evidence provided to substantiate any design changes made.	4.19
AF_ABWR_ME_006	Hitachi-GE has not provided sufficient evidence during GDA to demonstrate that maintenance activities on the FMCRD upper component meet UK standards of double isolation.	4.20

	The licensee shall provide an adequate demonstration that EIMT activities on the FMCRD can be performed safely, ensuring that risks are reduced ALARP.	
AF_ABWR_ME_007	The J-ABWR design assumes a 40-year operational lifetime for SSC equipment qualification. In some cases, Hitachi-GE safety case claims that no further equipment qualification is required for UK ABWR despite it having a 60-year operational lifetime. The licensee shall identify and qualify those SSCs that they will not maintain or replace during the assumed lifetime of the plant.	4.16, 4.17, 4.20

Annex 6

Minor Shortfalls

None

Annex 7
Regulatory Assumptions

Assumption ID	Regulatory Assumptions to be taken forward by the Licensee	Report Section Reference
AS-ABWR-ME01	ONR considers that piping and instrumentation diagrams presented during GDA are at a preliminary stage requiring further development. ONR assumes that the licensee shall develop these diagrams for all systems so that they are, suitable to facilitate transfer of piping and instrumentation details from the responsible designer.	4.13
AS-ABWR-ME02	ONR acknowledges that during GDA, Hitachi-GE had not sufficiently developed its design to enable the licensee to identify the appropriate design qualifications details. ONR assumes that the licensee shall establish detailed design substantiation, factory acceptance test information, and site acceptance test information for individual mechanical items and their associated systems, which are important to safety. ONR also assumes that the licensee shall generate appropriate evidence that equipment qualification is adequately specified for all mechanical items important to safety.	4.1, 4.9, 4.13, 4.19
AS-ABWR-ME03	ONR acknowledges that during GDA, Hitachi-GE had not sufficiently developed its design to enable the licensee to identify the appropriate plant limits and conditions. ONR assumes that the licensee shall identify plant limits and conditions, from the safety case, covering all mechanical engineering equipment important to safety. ONR also assumes that the licensee shall generate sufficient safety case information to satisfy the requirements of LC 23 (Operating Rules), and specifically they shall establish an interface for transferring this information from the responsible designer.	4.8, 4.14
AS-ABWR-ME04	During GDA, Hitachi-GE identified certain equipment that is required to perform activities associated with decommissioning of the plant at the end of its 60-year life. ONRs expectation is that a future licensee will ensure that	4.1, 4.5, 4.16, 4.17, 4.20

	initial design and manufacture of this identified equipment is suitable for these decommissioning tasks avoiding foreseeable modifications in the future.. ONR assumes that the licensee shall identify and confirm the use of equipment for decommissioning, The licensee shall consider design features to facilitate use of the equipment for the identified decommissioning tasks to reduce future dose uptake by workers where reasonably practicable by including any necessary design features in the final design.	
AS-ABWR-ME05	ONR acknowledges that during GDA, Hitachi-GE has not sufficiently developed its design to enable the licensee to identify the appropriate EIMT requirements in line with UK expectations. ONR assumes that the licensee shall ensure that they consider EIMT requirements for all mechanical engineering items that attract a safety classification. ONR also assumes that the licensee shall establish whether these requirements are directly driven by the safety case, are based on manufacturer recommendations or are based on plant operating experience (or appropriate combinations).	4.2, 4.4, 4.5, 4.7, 4.8, 4.9, 4.15, 4.16, 4.17, 4.19, 4.20, 4.22, 4.23
AS-ABWR-ME06	Hitachi-GE's change to the J-ABWR reference design (to facilitate removal of all lead materials where reasonably practicable) has the potential to alter the geometry and size of through wall penetrations. ONR assumes that the licensee shall ensure that these changes are made in accordance with Hitachi-GE's penetration design guidelines to minimise the impact on the reference design parameters.	4.18
AS-ABWR-ME07	During GDA Hitachi-GE did not describe the effective use, management and storage of Bio fuel for emergency diesel generators. ONR assumes that the licensee shall ensure that any diesel combustion plant used for the UK ABWR is designed to take into account the regulation amendment in respect of fuels, (Motor Fuel (Composition and Content) Regulations 1999), in terms of meeting their safety functional requirements.	4.16
AS-ABWR-ME08	During GDA, Hitachi-GE did not provide sufficient evidence to demonstrate that the off gas system pipework is qualified to withstand a hydrogen detonation event.	4.23

	ONR assumes that the licensee shall ensure that their detailed design includes analysis of convergent or divergent pipework design features that could enhance the peak pressure during such an event. ONR also assumes that the path of the shockwave generated by a hydrogen detonation in the off gas system, and its impact on other SSCs will be analysed.	
AS-ABWR-ME09	Limited evidence was provided during GDA to demonstrate that a fast acting explosively driven valve (squib valve) reduces risks, so far as is reasonably practicable, when compared to other options. Furthermore, Hitachi-GE did not adequately demonstrate that the explosive charge had been suitably qualified to underpin the safety and reliability claims. ONR assumes that the licensee shall ensure that they provide suitable optioneering to support the ALARP justification for the design, qualification to underpin the safety and reliability claims and evidence of an adequate EIMT regime.	4.25
AS-ABWR-ME10	Hitachi-GE has indicated that they will eliminate high-cobalt containing alloys from UK ABWR where reasonably practicable. However, Hitachi-GE has not indicated whether this could have a detrimental effect to the safety and reliability of individual components. ONR assumes that the licensee shall ensure that they consider these effects during detailed design and provide suitable evidence to demonstrate that there is no detriment to the safety and reliability of the SSC.	4.24