Office for Nuclear Regulation

Civil Nuclear Reactor Build - Generic Design Assessment

Step 2 Assessment of the Mechanical Engineering of Hitachi-GE's UK Advanced Boiling Water Reactor

Assessment Report ONR-GDA-AR-14-011
Revision 0
28/08/2014

Published 08/2014

EXECUTIVE SUMMARY

This report presents the results of my assessment of the Mechanical Engineering of Hitachi-General Electric Nuclear Energy Ltd's (Hitachi-GE) UK Advanced Boiling Water Reactor (UK ABWR) undertaken as part of Step 2 of the Office for Nuclear Regulation's (ONR) Generic Design Assessment (GDA).

The GDA process calls for a step-wise assessment of the Requesting Party's (RP) safety submission with the assessment becoming increasingly detailed as the project progresses. Step 2 of GDA is an overview of the acceptability, in accordance with the regulatory regime of Great Britain, of the design fundamentals, including review of key nuclear safety, nuclear security and environmental safety claims with the aim of identifying any fundamental safety or security shortfalls that could prevent the proposed design from being licensed in Great Britain. Therefore during GDA Step 2 my work has focused on the assessment of the key claims in the area of Mechanical Engineering to judge whether they are complete and reasonable in light of my current understanding of the reactor technology.

For Mechanical Engineering safety claims are interpreted as being the ability of a Structure, System, or Component (SSC) to deliver its safety function during normal operations (including for shutdown), fault sequences, and accident conditions, with adequate consideration of the following characteristics:

- inherent safety – hazard avoidance, in preference to hazard control;
- fault tolerance – sensitivity to potential faults to be minimised;
- defence in depth – provision of adequate levels of protection; and
- safety function – structured fault analysis undertaken for both normal operation (including shutdown), and fault sequences.

The guidance I have used to judge the adequacy of the claims in the area of Mechanical Engineering has been primarily ONR's Safety Assessment Principles (SAPs), and ONR's Technical Assessment Guides (TAGs).

My GDA Step 2 assessment work has involved continuous engagement with the RP in the form of correspondence, technical exchange workshops and technical meetings. In addition, my assessment has significantly benefited from visits to:

- the RP's Rinkai works where reactor internal components and Fine Motion Control Rod Drive Units are manufactured and tested. The visit provided me with an overview of this manufacturing capability; and
- Ohma ABWR, which was approximately 35% built. The visit provided me with an appreciation of the construction site; in particular the Drywell floor construction module and the complexity of the pipework associated with the safety injection of the reactor's control rods.

Initially the RP was not planning a Step 2 submission for Mechanical Engineering. Through engagement I accepted an arrangement that identified, prioritised and programmed a phased issue of a limited number Basis of Safety Case documents. A total of ten Basis of Safety Case documents formed the basis of the RP's Step 2 submission for Mechanical Engineering. These were selected based on the Structures, Systems and Components importance to safety.

The RP's preliminary safety case as presented in these documents, can be summarised as providing:

- the strategy over how the Mechanical Engineering safety case will be structured
- the systems that underpin the safety requirements of the ABWR safety case.
- a system overview; setting out the role, function, basis of configuration and modes of operation;
- the design basis of the plant's safety claims; and
- the SSC's design rationale, including outline arguments and evidence to substantiate the safety claims; the design engineering safety functions, reliability and performance requirements; safety categorisation and classification; assigned codes and standards; qualification and examination, inspection, maintenance and testing requirements.

My assessment has identified the following areas of strength in relation to Mechanical Engineering.

- the RP has proposed and commenced the implementation of an auditable arrangement for developing its safety case for the GDA. For Step 2 the safety case adopted structure and nature of the claims are appropriate and broadly aligned to expectations;
- the RP has provided assurance that it has in place appropriate arrangements to define functional, reliability and performance claims;
- the RP's categorisation and classification arrangement is broadly aligned to expectations for Mechanical Engineering SSCs;
- the RP's Step 2 submission appropriately sets out codes and standards for the principal Mechanical Engineering equipment;
- the RP has provided an adequate level of assurance associated with its operational experience arrangements at an organisational level; and
- the RP has provided a level of assurance that examination, inspection, maintenance and testing is appropriately considered as part of its design process.

During my GDA Step 2 assessment of the UK ABWR safety case related to Mechanical Engineering I have identified the following topic areas that require follow-up. This is to ensure that the RP adequately addresses the following aspects:

- that the nuclear ventilation system designs are aligned to UK codes, standards and UK Relevant Good Practice (RGP);
- that SSCs' design, qualification, reliability, maintainability and associated operational experience justify that risks have been reduced So Far As Is Reasonably Practicable (SFAIRP);
- that the RP adopts a robust, auditable design process with arrangements that set out design principles, rules, and selection criteria for all SSCs;
- that SSCs' qualification are aligned with the 60 year design life claim or the building layout and access provisions are adequate to support replacement;
- that SSC isolation and configuration for examination, inspection, maintenance and testing are aligned with UK RGP and risks are reduced SFAIRP;
- that SSCs' reliabilities are aligned with the output of the UK ABWR deterministic and probabilistic safety analyses; and
- that the design processes adequately consider operational experience at an SSC level.

To facilitate alignment of the above topic areas of the UK ABWR design to my regulatory expectations it is my intention to create and issue four Regulatory Observations.

The Mechanical Engineering specialism has not engaged technical support contractors to undertake any technical reviews as part of this Step 2 assessment. However, as part of Step 2 I have considered and identified a number of potential topic areas for technical review as part of my Step 3 assessment. A prerequisite to undertaking these technical reviews will be the adequacy of the RP's Step 3 submission, in terms of sufficiency of documents and associated timescales.

In summary, from a Mechanical Engineering perspective I see no reason why the UK ABWR should not proceed to Step 3 of the GDA process. In addition, I have not identified any fundamental shortfalls at this stage that have the potential to prevent the issue of a Design Acceptance Confirmation (DAC).

FOREWORD

Mechanical Engineering

In carrying out this assessment, the term 'Mechanical Engineering' encompasses a Structure, System or Component (SSC) that generally contains dynamic elements and interfaces. This is to distinguish it from the discipline of Structural Integrity, which is concerned with SSCs which are static in nature, primarily focussing on confinement safety function pressure boundaries. Notwithstanding this definition, a number of static components will also be of interest to the Mechanical Engineering discipline, and subject to appropriate assessment.

Examples of SSCs that are considered to be of interest to Mechanical Engineering include:

- control rod drive mechanisms;
- pumps;
- valves, (check valves, motor operated valves, safety relief valves, and isolation valves);
- cranes;
- mechanical handling systems;
- nuclear ventilation systems used to augment nuclear containment barriers;
- heating ventilation and air conditioning; and
- diesel generators.

Examples of static SSCs that are considered to be of interest to Mechanical Engineering include:

- heat exchangers;
- gloveboxes, cabinets;
- stillages;
- seals; and
- strainers.

Structural Integrity aspects with reference to the confinement safety function pressure boundaries and vessel internals are not specifically considered or assessed under the Mechanical Engineering discipline. These aspects are the subject of assessment under the discipline of Structural Integrity and reported in the assessment report covering that topic. The ONR Mechanical Engineering and Structural Integrity working arrangement is set out in Ref. 24.

LIST OF ABBREVIATIONS

ABWR	Advanced Boiling Water Reactor
ALARP	As Low As Reasonably Practicable
ASME	American Society of Mechanical Engineering
BMS	Business Management System
BPVC	Boiler and Pressure Vessel Code
BSC	Basis of Safety Case
DAC	Design Acceptance Confirmation
EIM&T	Examination, Inspection, Maintenance, and Testing
FMCRDU	Fine Motion Control Rod Drive Unit
GDA	Generic Design Assessment
HEPA	High Efficiency Particulate Air
Hitachi-GE	Hitachi-General Electric Nuclear Energy Ltd
HVAC	Heating Ventilation and Air Conditioning
IAEA	International Atomic Energy Agency
JPO	(Regulators') Joint Programme Office
LOLER	Lifting Operations and Lifting Equipment Regulations
MDEP	Multinational Design Evaluation Programme
NEA	Nuclear Energy Agency
NRA	Nuclear Regulatory Authority
OECD	Organisation for Economic Co-operation and Development
ONR	Office for Nuclear Regulation
PCSR	Pre-construction Safety Report
PSR	Preliminary Safety Report
RGP	Relevant Good Practice
RO	Regulatory Observation
RP	Requesting Party
SAP(s)	Safety Assessment Principle(s)
SFAIRP	So Far As Is Reasonably Practicable

LIST OF ABBREVIATIONS

SME	Subject Matter Expert
SSC	Structure, System or Component
TAG(s)	Technical Assessment Guide(s)
TSC	Technical Support Contractor
UK	United Kingdom
USNRC	United States Nuclear Regulatory Commission
WENRA	Western European Nuclear Regulators’ Association

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1 INTRODUCTION

1.1 Background

1. The Office for Nuclear Regulation's (ONR) Generic Design Assessment (GDA) process calls for a step-wise assessment of the Requesting Party's (RP) safety submission with the assessments becoming increasingly detailed as the project progresses. Hitachi-General Electric Nuclear Energy Ltd (Hitachi-GE) is the RP for the GDA of the United Kingdom Advanced Boiling Water Reactor (UK ABWR).

2. During Step 1 of GDA, which is the preparatory part of the design assessment process, the RP established its project management and technical teams and made arrangements for the GDA of its UK ABWR design. Also, during Step 1 the RP prepared submissions to be evaluated by ONR and the Environment Agency during Step 2.

3. Step 2 of GDA is an overview of the acceptability, in accordance with the regulatory regime of Great Britain, of the design fundamentals, including review of key nuclear safety, nuclear security and environmental safety claims with the aim of identifying any fundamental safety shortfalls that could prevent the proposed design from being licensed in Great Britain.

4. This report presents the results of my assessment of the Mechanical Engineering of the RP's UK ABWR as presented in the UK ABWR Preliminary Safety Report (PSR) (Ref. 2 and 3) and its supporting documents (Refs. 4 -13 incl.).

1.2 Methodology

5. My assessment has been undertaken in accordance with the requirements of the ONR "How2" Business Management System (BMS) procedure PI/FWD (Ref. 14). The ONR Safety Assessment Principles (SAPs) (Ref. 15), together with supporting Technical Assessment Guides (TAGs) (Ref. 16) have been used as the basis for this assessment.

2 ASSESSMENT STRATEGY

6. This section presents my strategy for the GDA Step 2 assessment of the Mechanical Engineering of the UK ABWR (Ref. 32). It also includes the scope of the assessment and the standards, guidance and criteria that I have applied.

2.1 Scope of the Step 2 Mechanical Engineering Assessment

7. The objective of my GDA Step 2 Mechanical Engineering assessment for the UK ABWR was to review and judge whether the claims made by the RP relating to Mechanical Engineering that underpin the safety aspects of the UK ABWR are complete and reasonable in light of my current understanding of the reactor technology.

8. Acknowledging Step 2 is to target the design fundamentals, my Mechanical Engineering assessment has considered:

   - SSCs' reliability requirements;
   - SSCs' ability to facilitate Examination, Inspection, Maintenance and Testing (EIM&T) requirements; and
   - Nuclear safety aspects that may impact the building layout and spatial considerations.

9. For Mechanical Engineering the term “safety claim” is interpreted as being:

   - The ability of a SSC to deliver its safety function during normal operation, (including for shutdown), fault sequences and accidents with adequate consideration to the following characteristics:

     - inherent safety – hazard avoidance, in preference to hazard control;
     - fault tolerance – sensitivity to potential faults to be minimised;
     - defence in depth – provision of adequate levels of protection; and
     - safety function – structured fault analysis undertaken for both normal operation (including shutdown), and fault sequences.

10. During GDA Step 2 I have also evaluated whether the safety claims related to Mechanical Engineering are supported by technical documents sufficient to allow me to proceed with GDA work beyond Step 2.

11. Finally, during Step 2 I have undertaken the following preparatory work for my Step 3 assessment:

    - identify likely topic areas where Technical Support Contactors (TSCs) can provide assistance in undertaking technical reviews;
    - instigate documentation to support placement of a TSC contract;
    - obtain a level of assurance that the RP Step 3 submission for Mechanical Engineering is likely to be aligned with my expectations;
    - obtain an understanding of the RP phased Step 3 submission timeline for Mechanical Engineering; and
    - develop a high level project management assessment programme for Mechanical Engineering aspects.

2.2 Standards and Criteria

12. The aim of the GDA Step 2 assessment is to reach an independent and informed judgment on the adequacy of a nuclear safety, security and environmental case. For this purpose, within ONR, assessment is undertaken in line with the requirements of the How2 Business Management System (BMS) document PI/FWD (Ref. 14). Appendix 1 of Ref. 14 sets out the process of assessment within ONR; Appendix 2 explains the process associated with sampling of safety case documents.

13. In addition, the SAPs (Ref. 15) constitute the regulatory principles against which duty holders' safety cases are judged, and, therefore, they are the basis for ONR's nuclear safety assessment and have been used for GDA Step 2 assessment of the UK ABWR. The SAPs 2006 Edition (Revision 1 January 2008) were benchmarked against the International Atomic Energy Agency (IAEA) standards (as they existed in 2004). They are currently under review.

14. Furthermore, ONR are a member of the Western European Nuclear Regulators' Association (WENRA) (Ref. 18). WENRA have developed Reference Levels, which represent good practices for existing nuclear power plants, and Safety Objectives for new reactors.

15. The relevant SAPs, IAEA standards and WENRA reference levels are embodied and built upon in the Mechanical Engineering TAGs (Ref. 16). These guides provide the principal means for assessing the Mechanical Engineering aspects in practice.

16. It is worth noting, the nature of the Mechanical Engineering discipline generally drives the assessment down to component level. Assessment at this component level can be extremely wide ranging given the very large number of such items, with numerous interfaces, across various plant process systems and covering multiple disciplines. As a consequence, a wide range of SAPs and TAGs (Ref. 15 and 16) can be applicable to undertaking an effective assessment. Accordingly, the Mechanical Engineering approach to carrying out an effective assessment is to select the most appropriate SAPs and TAGs specific to the aspect to be assessed.

2.2.1 Safety Assessment Principles

17. My Mechanical Engineering assessment has been undertaken with the aid of a number of applicable SAPs. These are principles against which regulatory judgements are made and provide the fundamental guidance in scoping an assessment topic and in undertaking an effective assessment. This approach ensures the assessment provides a targeted, consistent and transparent consideration on the adequacy of the UK ABWR design proposal.

18. Those SAPs considered relevant to the Mechanical Engineering Step 2 assessment are listed in Table 1 of this document. Individual SAPs are also detailed within the assessment text of this document against the relevant section.

19. Generally SAPs capture the requirements of WENRA reference levels and the IAEA standards series requirements.

2.2.2 Technical Assessment Guides

20. The following technical assessment guides have been considered as part of this Step 2 assessment (Ref. 16):

- ONR Nuclear Safety Technical Assessment Guide; Design Safety Assurance; TAST/057 Issue 2; November 2010;
- ONR Nuclear Safety Technical Assessment Guide; Safety Systems; T/AST/003 Issue 6; July 2011;
- ONR Nuclear Safety Technical Assessment Guide; Guidance on the demonstration of ALARP (as low as Reasonably Practicable); NS-TAST-GD-005 Rev 6; September 2013;
- ONR Nuclear Safety Technical Assessment Guide; Nuclear Lifting Operations; T/AST/056 Issue 002; December 2011;
- ONR Nuclear Safety Technical Assessment Guide; Examination, Inspection, Maintenance, & Testing of Items Important to Safety; NS-TAST-GD-009 Rev 2; November 2012;
- ONR Nuclear Safety Technical Assessment Guide; Diversity, Redundancy, Segregation and Layout of Mechanical Plant; NS-TAST-GD-036 Rev 3; April 2014;
- ONR Nuclear Safety Technical Assessment Guide; Integrity of Metal Components and Structures; NS-TAST-GD-016; March 2013;
- ONR Nuclear Safety Technical Assessment Guide; Ventilation; NS-TAST-GD-022 Rev 2; April 2013; and
- ONR Nuclear Safety Technical Assessment Guide; Containment: Chemical Plants; NS-TAST-GD-021 Rev 2; March 2013.

2.2.3 National and International Standards and Guidance

21. The following national and international standards and guidance have also been considered as part of this Step 2 assessment:

- Relevant IAEA standards (Ref. 17):

    - IAEA – Safety Standards: Safety of Nuclear Power Plants: Design, Specific Safety Requirement; SSR-2/1; IAEA 2012;
    - IAEA - Safety Standards: Safety Assessment for Facilities and Activities General Safety Requirements Part 4; GSR Part 4; IAEA 2009;
    - IAEA - Safety Standards: Seismic Design and Qualification for Nuclear Power Plants Safety Guide; NS-G-1.6; IAEA 2003;
    - IAEA - Safety Standards: Design of the Reactor Coolant System and Associated Systems in Nuclear Power Plants Safety guide; NS-G-1.9; IAEA 2004;
    - IAEA - Safety Standards: Commissioning for Nuclear Power Plants Safety Guide; NS-G-2.9; IAEA 2003;
    - IAEA - Safety Standards: Design of Reactor Containment Systems for Nuclear Power Plants Safety Guide; NS-G-1.10; IAEA 2004;
    - IAEA - Safety Standards: A System for the Feedback of Experience from Events in Nuclear Installations Safety Guide; NS-G-2.11; IAEA 2006;
    - IAEA - Safety Standards: Aging Management for Nuclear Power Plants Safety Guide; NS-G-2.12; IAEA 2009;

- IAEA - Safety Standards: Design of Fuel Handling and Storage Facilities for Nuclear Power Plants Safety Guide; NS-G-1.4; IAEA 2003; and
- IAEA - Safety Standards: Maintenance, Surveillance and In-service Inspection in Nuclear Power Plants Safety Guide; NS-G-2.6; IAEA 2002.

■ WENRA references (Ref. 18):

- Reactor Safety Reference Levels (January 2008);
- Safety Objectives for New Power Reactors (December 2009) and Statement on Safety Objectives for New Nuclear Power Plants (November 2010);
- Working Group on Waste and Decommissioning (WGWD); Waste and Spent Fuel Storage Safety Reference Levels (February 2011);
- Working Group on Waste and Decommissioning (WGWD); Decommissioning Safety Reference Levels (November 2011);
- Waste and Spent Fuel Storage Safety Reference Levels (February 2011);
- Decommissioning Safety Reference Levels (March 2012); and
- Statement on Safety Objectives for New Nuclear Power Plants (March 2013) and Safety of New NPP Designs (March 2013).

2.3 Use of Technical Support Contractors

22. The Mechanical Engineering specialism has not engaged technical support contractors to undertake any technical reviews as part of this Step 2 assessment. However, as part of Step 2 I have considered and identified a number of potential topic areas for technical review as part of my Step 3 assessment. A prerequisite to undertaking these technical reviews will be the adequacy of the RP's Step 3 submission, in terms of sufficiency of documents and associated timescales.

2.4 Integration with Other Assessment Topics

23. Early in GDA I recognised that during the project there would be a need to consult with other assessors (including Environment Agency's assessors) as part of the Mechanical Engineering assessment process. Similarly, other assessors will seek input from my assessment of the Mechanical Engineering for the UK ABWR. I consider these interactions to be important to ensure the prevention of assessment gaps and duplications, and, therefore key to the success of the project. Thus, from the start of the project, I made every effort to identify the many potential interactions between the Mechanical Engineering and other technical areas, with the understanding that this position would evolve throughout the UK ABWR GDA.

24. It should also be noted that the interactions between Mechanical Engineering and some technical areas may need to be formalised since certain aspects of the assessment constitute formal inputs to the Mechanical Engineering assessment, and vice versa. At this stage however interactions have been on an informal basis covering the following:

■ Deterministic Fault Studies and Probabilistic Safety Analysis have provided input to safety functional, reliability and availability requirements of the Mechanical Engineering SSCs; and

- other disciplines have provided input to the design of SSCs from a perspective of reliability, availability, maintainability and access provision, to minimise operator dose uptake, manage radioactive waste and to demonstrating SSCs' risks have been reduced So Far As Is Reasonably Practicable (SFAIRP).

25. These interactions are expected to continue throughout GDA and are considered important to ensure consistency across the various technical assessment areas.

3 REQUESTING PARTY'S SAFETY CASE

26. This section presents a summary of the RP's preliminary safety case in the area of Mechanical Engineering. It also identifies the documents submitted by the RP which have formed the basis of my assessment of the UK ABWR during GDA Step 2.

3.1 Summary of the RP's Preliminary Safety Case

27. As the RP was not initially planning a Step 2 submission for Mechanical Engineering, it implemented an arrangement (Ref. 1) that identified, prioritised and programmed a phased issue of a limited number of Basis of Safety Case (BSC) documents. The documents were selected on the SSCs importance to safety and covered a range of safety functions associated with:

- reactivity control;
- heat transfer and removal; and
- confinement of radioactive substances

28. The RP's Pre-Construction Safety Report (PCSR) has not formed part of its Step 2 Submission. The RP plans to submit its PCSR for assessment as part of its Step 3 submission.

29. As part of its Step 2 submission the RP has issued a PSR on Mechanical Engineering (Ref. 2 and 3). This sets out the RP's strategy for its GDA safety case from a Mechanical Engineering perspective. The PSR, which is a summary Level-1 document, identifies 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, in the form of:

- Safety Functional claims;
- Reliability claims; and
- Performance claims.

30. Beneath the PSR are a suite of Level-2 BSC documents. It is the BSC documents, which are broadly system based that are of a principal interest to Mechanical Engineering.

31. Each issued BSC document provides:

- a system overview; setting out the role, function, basis of configuration and modes of operation;
- the design basis of the plant's safety claims; and
- the SSC's design rationale, including arguments and evidence to substantiate the safety claims; design engineering safety functions, reliability and performance requirements; safety categorisation and classification; assigned codes and standards; qualification and EIM&T requirements.

32. The RP's Step 2 submission for Mechanical Engineering did not cover the equipment important to safety for the whole plant, instead the RP has provided a sample of BSC documents for the following UK ABWR systems:

- Standby gas treatment system (Ref. 4);
- Condensate & feedwater system (Ref. 5);

- Heating ventilation and air conditioning system (Ref. 6);
- Emergency power supply (Ref. 7);
- Off-gas system (Ref. 8);
- Nuclear boiler system (Ref. 9);
- Reactor building cooling water systems (Ref. 10);
- Reactor recirculation system (Ref. 11);
- Control rod drive system (Ref. 12); and
- Residual heat removal system (Ref. 13).

33. The RP's plan is to submit its complete suite of system BSC documents for the UK ABWR as part of its Step 3 submission.

3.2 Basis of Assessment: RP's Documentation

34. The RP's documents that have formed the basis for my GDA Step 2 assessment of the safety claims related to the Mechanical Engineering for the UK ABWR are:

- UK ABWR PSR chapter on Mechanical Engineering "UK ABWR GDA (Generic Design Assessment) preliminary safety report on Mechanical Engineering" (Ref. 2 and 3).

  This document sets out the RP's strategy for the safety case from the Mechanical Engineering perspective.
- UK ABWR Basis of Safety Case document "*UK ABWR GDA (Generic Design Assessment) Basis of Safety Cases on Standby Gas Treatment System*" (Ref. 4);
- UK ABWR Basis of Safety Case document "*UK ABWR GDA (Generic Design Assessment) Condensate & Feedwater System Basis of Safety Case"* (Ref. 5);
- UK ABWR Basis of Safety Case document "*UK ABWR GDA (Generic Design Assessment) Basis of Safety Cases on Heating Ventilation and Air Conditioning System"* (Ref. 6);
- UK ABWR Basis of Safety Case document "*UK ABWR GDA (Generic Design Assessment) Emergency Power Supply Systems Basis of Safety Case"* (Ref. 7);
- UK ABWR Basis of Safety Case document "*UK ABWR GDA (Generic Design Assessment) Off-Gas System Basis of Safety Case"* (Ref. 8);
- UK ABWR Basis of Safety Case document "*UK ABWR GDA (Generic Design Assessment) Nuclear Boiler System Basis of Safety Case"* (Ref. 9);
- UK ABWR Basis of Safety Case document "*UK ABWR GDA (Generic Design Assessment) Reactor Building Cooling System Basis of Safety Case"* (Ref. 10);
- UK ABWR Basis of Safety Case document "*UK ABWR GDA (Generic Design Assessment) Reactor Recirculation System Basis of Safety Case"* (Ref. 11);
- UK ABWR Basis of Safety Case document "*UK ABWR GDA (Generic Design Assessment) Control Rod Drive System Basis of Safety Case e"* (Ref. 12); and
- UK ABWR Basis of Safety Case document "*UK ABWR GDA (Generic Design Assessment) Residual Heat Removal System Basis of Safety Case"* (Ref. 13).
- UK ABWR GDA document tracking sheet (Ref. 20); and
- ONR GDA Regulatory Queries tracking database (Ref. 20).

35. The BSCs are a Level-2 suite of documents. They are the subject of frequent updates as the safety case for the UK ABWR develops. They aim to set out the detailed safety case of a system or group of systems and provide a link to the related Level-1 PCSR summary chapters. The PCSR chapters are a summary of the safety case as presented in the BSC suite of documents. The BSC suite of documents will be developed further as part of GDA Step 4 to summarise the detailed level 3 design documentation.

36. In May 2014 the RP submitted to ONR for information an advance copy of the UK ABWR PCSR (Ref. 21). Although I have not reviewed this report in detail as part of my GDA Step 2 formal assessment, seeing it has been of value and will aid the planning and preparation of my GDA Step 3 assessment.

4 MECHANCIAL ENGINEERING ASSESSMENT

4.1 Background

37. My assessment has been carried out in accordance with ONR "How2" BMS document PI/FWD, "Purpose and Scope of Permissioning" (Ref. 14).

38. My GDA Step 2 Mechanical Engineering assessment has followed the strategy described in Section 2 of this report.

39. My Step 2 assessment work has involved frequent engagement with the RP's Mechanical Engineering Subject Matter Experts (SMEs), i.e. I have undertaken two planned technical meetings in Japan, four progress / clarification video conference meetings and one Step 2 assessment position meeting via video conference. I have also visited:

- the RP's Rinkai works where reactor internal components and Fine Motion Control Rod Drive Units (FMCRDUs) are manufactured and tested. The visit provided me with an overview of this manufacturing capability. In addition, it provided clarification as to my understanding of how the component functions; and
- Ohma ABWR which was approximately 35% built. The visit provided me with an appreciation of the construction site; in particular the Drywell floor construction module and the complexity of the pipework associated with the safety injection of the reactor's control rods. The system pipework is located within the plant's Drywell area that is underneath the main reactor pressure vessel, where I judged space to be limited to undertake planned EIM&T.

40. During my GDA Step 2 assessment, I have raised 22 Regulatory Queries (RQ) (Ref. 20). These have been discussed at the technical meetings and the responses assessed to inform my position as set out within this report.

41. Details of my GDA Step 2 assessment of the UK ABWR preliminary safety case in the area of Mechanical Engineering including the areas of strength that I have identified, as well as the items that require follow-up and the conclusions reached, are set out in the following sub-sections.

42. To facilitate my Step 2 assessment findings and to align the UK ABWR design to my regulatory expectations it is my intention to create and issue four regulatory observations.

43. My Step 2 assessment strategy for Mechanical Engineering was set out in advance (Ref. 32). It set out the scope, standards and criteria that I planned to apply to my assessment. I prepared it in December 2013 and in line with ONR regulatory policy of openness and transparency I shared it with the RP.

44. Mechanical Engineering SSCs typically deliver the principal safety functions of:

- reactivity control;
- heat transfer and removal; and
- confinement of radioactive substances.

45. The principal aims of Step 2 are to:

- identify fundamental issues;
- identify the key claims, confirming they are complete and reasonable; and
- identify the availability of supporting arguments and evidence for assessment during Step 3 and Step 4 of the GDA process.

46. My assessment has targeted the RP's design philosophies, the robustness of its design process arrangement and aspects that I consider may impact the building footprint and spatial considerations. I consider the RP's extant design process arrangements should be able to provide the necessary assurance and confidence to demonstrate an SSC:

- design has been adequately optioneered, taking account of operational experience and Relevant Good Practice (RGP);
- is suitable for the purpose for which it is to be used;
- risks have been reduced SFAIRP, a requirement of UK legislation (Health & Safety at Work etc. Act 1974); and
- requiring EIM&T or replacement has adequate space and routing provision.

4.2 Claim Structure Overview

47. The documents (Ref.3 to 13 incl.) adequately set out how the top level and the general plant level claims are cascaded to the specific Mechanical Engineering SSCs, in the form of:

- Safety Functional claims;
- Performance claims; and
- Reliability claims.

48. Each BSC document adopts a standard structure which provides:

- a system overview; setting out the role, function, basis of configuration and modes of operation;
- the design basis of the plant's safety claims; and
- the SSC's design rationale, including arguments and evidence to substantiate the safety claims; the design engineering safety functions, reliability and performance requirements; safety categorisation and classification; assigned codes and standards; qualification and EIM&T requirements.

49. These submissions are in line with my expectations and set out an acceptable safety case methodology and structure. My Step 2 assessment has been limited to the extent of the RP's issued submission for Mechanical Engineering (References 4-13). I acknowledge the documents are at an early stage of development, and that some of the claims require more detail and refinement. They will be subject to controlled updates as the RP develops its UK ABWR safety case and takes account of ONR regulatory expectations.

50. In conclusion I judge the RP's safety case claim structure and nature of the claims as set out for Step 2 are broadly aligned with my expectations.

4.3 Safety Functional, Reliability and Performance Claims

51. It is my regulatory expectation that engineered SSCs should be designed and substantiated to deliver their required safety functions with adequate reliability, according to the magnitude and frequency of the radiological hazard, to provide confidence in the robustness of the overall design.

52. In addition, it is my expectation that the RP demonstrates the proposed engineering is of a level commensurate to the reliability figures defined by the UK ABWR deterministic and probabilistic safety analyses.

53. I consider the safety assessment principles series ”EDR” Design for reliability (Ref. 15) to be relevant to this topic area.

54. The starting point to designing a SSC to meet its design intent is to undertake a safety analysis, the output of which defines the principal safety case claims. Subsequently, the analysis allows the safety functional, reliability and performance claims to be defined and placed on specific SSCs.

55. In line with my expectation each assessed BSC document contains a section that appropriately sets out:

- the safety functional claims;
- the reliability claims; and
- the performance claims for the system.

56. The safety analyses define the safety functions placed on the high level claims that on evaluation place lower level performance, reliability and availability claims on specific SSCs. In line with expectations the documents examined appropriately set out these aspects taking account of normal operations as well as fault sequences.

57. Through examination of the submission I have identified that limited reliability figures have been provided to date. I consider this to be a shortfall in expectations and a topic to be followed up during my Step 3 assessment.

58. Through undertaking technical discussions and assessment of RQ responses (Ref. 20), the RP has stated its UK ABWR deterministic and probabilistic safety analyses are ongoing, but are planned to be completed within the GDA Step 3 timeframe. Once completed the RP plans to use the analyses to define the SSCs reliability requirements, which will assign the appropriate codes, standards and quality assurance to the SSCs. This strategy provides assurance that the SSCs can deliver their design intent.

59. In conclusion at Step 2 the RP has provided assurance that it has in place adequate arrangements to define auditable functional, reliability and performance claims and a suitable claims structure to appropriately set them out. Due to the absence of the availability of the safety analyses and to ensure my expectations are met I plan to follow up this topic area as part of my Step 3 planned assessment (Ref. 31).

4.4 Categorisation and Classification

60. The “ECS” series of the ONR SAPs (Ref. 15) for nuclear facilities expect:

- 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; and
- SSCs that have to deliver safety functions should be identified and classified on the basis of those functions and their significance with regards to safety.

61. The RP sets out its categorisation and classification methodology within its main summary Level-1 documents (Ref. 28). This is judged to be broadly in line with my expectations. Examination of the Mechanical Engineering PSR and the BSCs has

provided further assurance of the RP methodology being applied appropriately at an SSC level.

62. One assessment finding that requires follow up is the RP's response to Regulatory Query (RQ-ABWR-0005)(Ref. 20) that states the RP's approach to safety categorisation and classification may assign a third line of protection as a classification "3" while supporting a categorisation "A" safety function.

63. I consider this is not aligned with my regulatory expectations from the aspect of assignment of appropriate codes and standards to category "A" safety functions. During Step 3 (Ref. 31) I plan to follow up this aspect to understand the rationale behind SSCs being assigned with a category "A" safety function and as classification "3" and the assignment of appropriate codes and standards.

64. In conclusion at Step 2 the RP's arrangement is broadly aligned with my expectations. However, follow up of the category "A", classification "3" design approach is planned as part of my Step 3 assessment (Ref.31).

4.5 Codes and Standards

65. The "ECS" series of the ONR SAPs (Ref. 15) for nuclear facilities expects SSCs assigned as important to safety to be designed, manufactured, constructed, installed, commissioned, quality assured, maintained, tested and inspected to the appropriate standards.

66. In line with my expectation each assessed BSC document contains a section that appropriately sets out the assigned codes and standards for the principal Mechanical Engineering SSCs.

67. In principle, the RP claims Mechanical Engineering equipment will be designed in accordance with ISO, British Standards and European standards. However, some ABWR specific equipment such as the control rod drive units and the reactor internal pumps have no dedicated standards. Thus, the RP designs and manufactures them to the manufacturer's standards. The RP claims these to be of equivalence to relevant nuclear standards for the assigned classification. In addition, the RP claims the nuclear vent systems are designed in accordance with UK RGP.

68. I consider these observations to be of interest and as part of my assessment selected to undertake further examination of the claims. In addition, due to their importance to safety I also selected to further examine the assigned codes and standards of the main steam isolation valves.

4.5.1 Main Steam Isolation Valves, Fine Motion Control Rod Drive Units and Reactor Internal Pumps

69. The RP has stated through responses to regulatory queries RQ-ABWR-0008 and 14 (Ref. 20) and at a technical meeting (Ref. 22) that there are no official codes and standards applicable for the internal aspects of the reactor internal pumps and the FMCRDUs. I consider this to be a shortfall in my regulatory expectations. Pursuing this aspect and in response to questions the RP has stated its FMCRDUs are designed and manufactured in accordance with United States Nuclear Regulatory Commission (USNRC) 10CFR50 Appendix B guidance. This guidance is concerned with the requirement to include an appropriate quality assurance programme for the

design, fabrication, construction and testing of SSCs of a nuclear power plant. The adequacy of such arrangements will be followed up as part of my Step 3 assessment.

70. At a technical meeting (Ref.22) the RP stated the external pressure boundary aspects of the Fine Motion Control Rod Drive Units (FMCRDUs) and the reactor internal pumps are designed and manufactured in accordance with the requirements of the American Society of Mechanical Engineering (ASME) Boiler and Pressure Vessel Code (BPVC) Section III. Through informal consultation with the ONR Structural Integrity Inspector, I consider this to be in line with regulatory expectations for Class 1 and 2 SSCs.

71. I have also targeted the design codes, standards and quality assurance assigned to the main steam isolation valves and the pressure relief valves, both of which are assigned with a safety classification 1. The RP stated that it has an internal specification that places specific quality control requirements on the design and manufacturing aspects of the valves. I consider this to be in line with my expectations and plan to consider the need to undertake an assessment of the specification as part of my Step 3 assessment.

72. In conclusion at Step 2 the RP's submission appropriately sets out codes and standards used for design of the UK ABWR. The RP has provided adequate assurance for codes and standards associated with the sampled SSCs' pressure boundary aspects. However, further assessment is planned during Step 3 (Ref. 31) to facilitate assurance that those internal procedures used for the design and manufacture of parts of SSCs are adequate given the classification and categorisation of the SSC.

4.5.2 Nuclear Ventilation Systems

73. Containment and ventilation systems should confine the nuclear matter within the facility and prevent its leakage and escape to the environment in normal operation and fault conditions, except in accordance with authorised discharge conditions, or as part of a planned transfer to another facility.

74. The term 'containment' encompasses a wide range of structures and plant items, from the massive buildings surrounding power reactors, to glove boxes and individual packages and containers. Containments often have associated systems, such as cooling systems and sprays, which are considered to be part of the containment system.

75. I consider the safety assessment principles series "ECV" Containment and ventilation and "ECS.3" Standards (Ref. 15) to be relevant to this topic

76. The RP has stated that it is to adopt UK RGP NVF/DG001 (Ref. 25) nuclear industry guidance for its ventilation SSCs. This is aligned with my expectations, however, I also consider ISO 17873:2004 and ISO 26802:2010 (Ref. 26 and 27) to be applicable.

77. Through discussion of the HVAC, off-gas and standby gas systems the RP has provided some limited assurance that it understands the impact of adopting its stated guidance of NVF/DG001 to the UK ABWR design. Discussion on undertaking filter EIM&T also identified the RP has limited understanding of other applicable UK legislation and its impact on the proposed design. For example: Confined Spaces Regulations 1997; Manual Handling Operations Regulations 1992, Lifting Operations and Lifting Equipment Regulations 1998 (LOLER).

78. Through the systems overview discussion I noted the following observations that are examples of aspects that are not aligned with my expectations:

- systems do not incorporate safe change filter housings or circular type filters;
- systems do not incorporate primary and secondary filter banks;
- a number of extract systems are reliant on bag type filtration in preference to High Efficiency Particulate Air (HEPA) filtration;
- the standby gas system fan is located upstream of the system filter bank;
- the standby gas system incorporates an unfiltered bypass process line;
- the active workshops and waste treatment facilities are not defined with dedicated local filtered extract systems. In addition, it is not the RP normal practice to incorporate such systems in these plant areas; and
- the temperature design basis of systems (+33 to -9.3 ºC) is not aligned with external hazard expectations.

79. The RP has stated that it is currently undertaking a review and an impact assessment to understand the implications of aligning its design to NVF/DG001. I was encouraged when the RP stated it is seeking the support of a UK consultancy that is familiar and knowledgeable of working to the NVF/DG001 guidance to aid its review.

80. I have informally shared my assessment observations with the ONR Conventional Safety, Radiation Protection, Civil and External Hazard Inspectors, and the Environment Agency, for consideration within their assessment.

81. In conclusion at Step 2 the RP's claim is reasonable but is not complete and further assessment is required. The RP has not provided sufficient assurance that it adequately understands the design requirements to incorporate UK RGP codes and standards to its nuclear ventilation SSCs. I acknowledge the RP is undertaking a review to understand the impact of aligning its design to UK RGP, which I see as a positive approach. However, due to the significance of the assessment finding, it is my intention to raise a Regulatory Observation. Furthermore this will form an important topic to follow up during my Step 3 assessment (Ref. 31).

4.6 Design Process

4.6.1 Robustness

82. It is my regulatory expectation that the RP has a robust, consistent and auditable design process that sets out design principles, rules and selection criteria for all the ABWR SSCs important to safety. I consider the SAP series "EDR" Design for reliability (Ref. 15) to be pertinent to this topic.

83. Through technical discussion (Ref. 22) the RP has provided an adequate level of assurance that its adopted design process incorporates an embedded valve selection arrangement (Ref. 30). I consider this to be aligned with my expectations.

84. However, my assessment has identified that the RP design process arrangements do not set out any design principles, rules, considerations and selection criteria for the following sampled components:

- flexible and temporary hoses; or
- heat exchangers.

85. Through discussion (Ref. 22) the RP has explained it achieves a robust and consistent approach across the various plant systems by applying its design review process to a developed concept. I consider this approach has the potential for the various system teams to optioneer different solutions for the same application. I consider this falls short of my regulatory expectations. I acknowledge the RP's proposal to capture the above aspects within its proposed design documentation (RQ-ABWR-0006)(Ref. 20)

for the UK ABWR. However, I consider the flexible and temporary hoses plus heat exchangers are only examples of where I judge the RP's design process is not fully aligned with my regulatory expectations. I consider the RP's design process arrangement should set out design principles, rules, and selection criteria for all equipment important to safety in advance of optioneering a concept.

86. In conclusion at Step 2 the RP's design process is not fully aligned with my expectations. The RP has provided some limited assurance of its adequacy and robustness. However, it is my intention to raise a Regulatory Observation that the RP should have a robust, consistent and auditable design process across the various systems and for all the plants SSCs. In addition, this topic area will form an important aspect to progress during my Step 3 assessment (Ref. 31).

4.6.2 UK ABWR Design Life

87. Engineered SSCs need be designed to deliver their required safety functions with adequate reliability (SAP ERL.1) (Ref. 15). It is the SSC qualification processes (SAP EQU.1) (Ref. 15) and assignment of commensurate EIM&T regimes (SAP EMT.1) (Ref. 15) that provide assurance of achieving the reliability requirements. However, if a SSC is expected to be replaced during the life of the plant, adequate engineered provision should be provided within the plant's design (SAP ELO.1) (Ref. 15).

88. The ABWR reference design has a 40 year design life. However, the UK submission states the facility has a 60 year design life. To gain a level of assurance, I pursued the RP's equipment qualification arrangements for the Mechanical Engineering SSCs with a 60 year design life requirement.

89. Through assessment and technical discussion of the RP's Step 2 BSC submissions (4-13 incl.) the RP has provided limited assurance that it has reviewed or considered the need to undertake additional qualification of Mechanical Engineering SSCs important to safety to support its 60 year design life claim.

90. The exceptions to the above are the main steam isolation valves and the pressure relief valves; the RP has stated that it intends to undertake additional valve cycling qualification to substantiate the valves for the revised claim. I was encouraged by the RP's strategy for these specific SSCs.

91. In addition, limited assurance has been provided by the RP that it has considered the building layout and ingress/egress provision as being adequate to replace any large SSCs that are to retain a 40 year design life. An example of this is the ability to replace the condensate and feedwater system heaters.

92. I shared with the RP that it is my expectation that SSCs are optioneered and risks are reduced SFAIRP, with adequate consideration to the volume of radioactive waste generated and operator dose uptake. I stated that an increase in a SSC design life claim from 40 to 60 years may lead to additional qualification being required to be undertaken. In addition, it is my regulatory expectation that the building layout for a SSC that retains a 40 year design has a fully engineered route to support its replacement.

93. In conclusion at Step 2 the RP's claim may be reasonable but further assessment is required. The RP has provided limited assurance that it has reviewed or considered undertaking additional SSC qualification to support the revised claim for all SSC's. In addition, the RP has not adequately demonstrated the building layout incorporates sufficient ingress/egress provision for SSCs that will require replacement during the plants operational phase. Due to the potential significance of these two assessment

findings it is my intention to raise a Regulatory Observation. In addition, this topic area will form an important aspect to follow up during my Step 3 assessment (Ref. 31).

4.6.3 Operational Experience

94. Adequate consideration to operational experience is an important aspect of ensuring the safety of a nuclear power plant. It is particularly important to Mechanical Engineering as it is the SSC's performance, reliability, engineering and quality control that provide the assurance and confidence to deliver the safety functions of the facility. In addition, it supports the evolution and definition of the next generation of the design. I consider the UK legislation – the Health and Safety at Work etc Act 1974 and the requirement to reduce risks SFAIRP plus the following safety assessment principles (Ref. 15) to be pertinent to this topic area:

- safety assessment principles series "EAD" ageing and degradation;
- safety assessment principle ECM.1, which states "Before operating any facility or process that may affect safety it should be subject to commissioning tests to demonstrate that, as built, the design intent claimed in the safety case has been achieved"; and
- safety assessment principle EMT.7, which states "In-service functional testing of SSCs important to safety should prove the complete system and the safety-related function of each component".

95. My assessment has noted the RP submission presents a mature design that has in excess of 17 years operational experience. Through responses to regulatory queries (Ref.20) and technical engagement (Ref. 22), the RP has provided an adequate level of assurance that it has arrangements for collating and considering operational experience at an organisational level. However, discussions on specific SSCs have not provided the same level of assurance (Ref. 22 & 23).

96. In conclusion at Step 2 I am content that the RP has arrangements for considering operational experience at an organisational level. However, to facilitate assurance at a SSC level I plan to pursue the topic further as part of my Step 3 planned assessment (Ref. 31) and plan to target:

- my line of enquiry to specific SSC design changes instigated directly from operational experience; and
- any areas of potential improvements highlighted in the ONR commissioned operational experience reports.

4.7 EIM&T

4.7.1 Consideration & Frequency

97. Engineered structures, systems and components should be designed to deliver their required safety functions with adequate reliability, according to the magnitude and frequency of the radiological hazard, to provide confidence in the robustness of the overall design.

98. It is also an expectation that SSCs important to safety are assigned with a commensurate EIM&T regime. This is to provide the assurance that a SSC retains its design intent throughout the operational phase of the plant, including plant shutdowns. I consider the safety assessment principles series "EDR" Design for reliability and "EMT" Maintenance, inspection, and testing (Ref. 15) to be pertinent to this topic.

99. In line with my expectation, the assessment of the RP's BSC documents provide an adequate level of assurance that the SSCs are designed to take account of examination, inspection, maintenance and undertaking surveillance activities. I consider this to be important in securing the delivery of the SSC's reliability demands. In addition, it is the Mechanical Engineering SSC's reliability in supporting duty operations that limits the reliance on the plant's safety systems.

100. However, through discussion of the listed systems (as set out in Section 3.1), the RP stated that several SSCs contain components that are frequently replaced. In response to questions on this aspect the RP stated the frequency of replacement is based on normal Japanese practice. I consider the RP stated replacement frequencies are more frequent than currently observed within the UK; examples of such components include:

- diesel generator pistons on a 2-8 year cycle;
- valves (throttling type) replaced on a 3 year cycle;
- feedwater pump bearings on a 2 year cycle;
- standby gas treatment system heating coil and electric space heater on a 2 year cycle; and
- sluice gate seals on a 3 year cycle.

101. Due to the replacement frequency attached to a number of the SSCs I judge at this stage of my assessment that there may be potential issues with respect to the SSCs' design, reliability and qualification aspects.

102. In conclusion at Step 2 I consider the RP takes account of EMI&T requirements within its design process. However, my engineering judgement considers a number of SSCs are replaced more frequently than expected. To facilitate my regulatory judgement I plan to pursue the topic as part of my planned Step 3 assessment (Ref. 31) by:

- targeting the arguments and evidence to substantiate the adopted strategy;
- considering if the strategy reduces risks SFAIRP as a number of components will be consigned as radioactive waste, and replacement activities will incur operator dose uptake;
- considering the impact of human factors activities on the reliability claim; and
- targeting the components' design intent, reliability and qualification aspects.

4.7.2 Isolations and Configurations

103. It is my expectation that all EIM&T isolations and configurations are in accordance with RGP HSG253 (Ref 29) and risks are reduced SFAIRP. I consider UK legislation – the Health and Safety at Work etc Act 1974 and the requirement to reduced risks SFAIRP and safety assessment principles series “EDR” Design for reliability and ELO. 1 (Ref. 15) to be pertinent to this topic area:

104. My assessment has highlighted an example of planned EIM&T with the reactor internal pump plug that relies on a single isolation seal for confinement. The single seal is the only design measure preventing active fluid that is under a significant hydraulic pressure within the reactor pressure vessel from leaking onto operators undertaking EIM&T activities in the Drywell area directly underneath the reactor pressure vessel. I consider this assessment observation to be a shortfall in my regulatory expectations to reduce risks SFAIRP.

105. In conclusion at Step 2 the RP's arrangements do not meet all aspects of UK RGP. It is my intention to raise a Regulatory Observation and the topic area will form an important aspect to follow up during my Step 3 assessment (Ref. 31).

4.8 PCSR

106. The RP has provided a progress update and timeline for its Step 3 submission for Mechanical Engineering. The RP submission will include a PCSR and all the applicable Mechanical Engineering BSC documents.

107. Through discussion at technical meetings, the RP has provided an adequate level of assurance that its Mechanical Engineering BSCs are being generated to include the applicable arguments to support the SSC claims. I also noted the Mechanical Engineering BSCs are programmed to be issued in a phased manner. I need to take this into account when planning my Step 3 assessment, in particular programming the requirement for technical support contractor resource.

108. With reference to Step 4 and the ONR assessment of evidence, I have shared with the RP that I acknowledge:

- a significant level of evidence for Mechanical Engineering SSCs is provided from undertaking factory acceptance tests and site commissioning tests; and
- this evidence is not expected to be available during the GDA timeframe.

109. I have shared with the RP that in the absence of the availability of actual Mechanical Engineering design substantiation evidence, I would target and undertake an assessment of the RP arrangements to produce and underpin this evidence.

4.9 Out of Scope Items

110. The list below sets out the systems that were excluded from the scope of my Mechanical Engineering Step 2 GDA. This was accepted and agreed in advance of initiating my Step 2 assessment, noting the documents are expected to form part of the RP's Step 3 submission:

- reactivity control:
    - Standby liquid control system.
- heat transfer and removal:
    - Emergency core cooling systems; and
    - Fuel storage systems.
- confinement of radioactive substances:
    - Seals and leak detection systems;
    - Overpressure protection system;
    - Severe accident management systems;
    - Steam and power conversion system; and
    - SSCs handling systems.

111. The RP has also stated during technical discussion that the battery limits of the GDA for the reactor building cooling water system are the service supply water pump and the output of the heat exchanger. The RP has stated the downstream design aspects are to form part of site specific aspects.

112. It should be noted that the above omissions do not invalidate the conclusions from my GDA Step 2 assessment. During my GDA Step 3 assessment I plan to consider the detailed follow-up aspects as set out in this report; and the above listed systems. My GDA Step 3 Assessment Plan (Ref. 31) will set out my assessment process in detail.

4.10 Interactions with Other Regulators

113. As part of my Step 2 assessment I have worked with the Environment Agency under the ONR memorandum of understanding arrangement as an integral part of the assessment process. However, at this stage I have not identified any specific areas of Mechanical Engineering interest where formal detailed liaison has been considered necessary.

114. As part of my Step 3 assessment process I shall continue to consider Mechanical Engineering specific topic areas that would benefit from undertaking detailed liaison with other regulators.

4.11 Overseas Regulatory interface

115. In accordance with its strategy, ONR collaborates with overseas regulators, both bilaterally and multinationally.

116. At this stage of my assessment I have not identified any specific areas of Mechanical Engineering where detailed liaison has been considered necessary.

117. As part of my Step 3 assessment process I shall continue to consider Mechanical Engineering specific topic areas that would benefit from undertaking bilateral collaboration, with particular consideration to engage with the Japanese Nuclear Regulation Authority (NRA).

4.12 Multilateral collaboration

118. ONR collaborates through the work of the International Atomic Energy Agency (IAEA) and the Organisation for Economic Co-operation and Development (OECD) Nuclear Energy Agency (OECD-NEA). ONR also represents the UK in the Multinational Design Evaluation Programme (MDEP) - a multinational initiative taken by national safety authorities to develop innovative approaches to leverage the resources and knowledge of the national regulatory authorities tasked with the review of new reactor power plant designs. This helps to promote consistent nuclear safety assessment standards amongst different countries.

119. At this stage of my assessment I have not identified any specific areas of Mechanical Engineering where detailed liaison has been considered necessary.

120. As part of my Step 3 assessment process I shall continue to consider Mechanical Engineering specific topic areas that would benefit from undertaking multilateral collaboration.

5 CONCLUSIONS AND RECOMMENDATIONS

5.1 Conclusions

121. The RP has provided a PSR and a limited number of BSCs for the UK ABWR for assessment by ONR during Step 2 of GDA. The PSR and BSCs together with their supporting references provide a level of assurance that the claims can be adequately developed and supported by arguments and evidence in the area of Mechanical Engineering for the GDA.

122. During Step 2 of GDA I have conducted an assessment of the parts of PSR, BSCs and their references that are relevant to the area of Mechanical Engineering against the expectations of the SAPs and TAGs. From the UK ABWR assessment completed so far I conclude the following:

- the RP has proposed and commenced the implementation of an auditable arrangement for developing its safety case for the GDA. For Step 2 the safety case adopted structure and nature of the claims are appropriate and broadly aligned to expectations;
- the RP has provided assurance that it has in place appropriate arrangements to define auditable functional, reliability and performance claims;
- At Step 2 the RP’s reliability claims are not complete, full assessment has not been possible due to their absence and further assessment will be undertaken in Step 3;
- the RP’s categorisation and classification arrangement is broadly aligned to expectations for Mechanical Engineering SSCs;
- the RP’s Step 2 submission appropriately sets out codes and standards for the principal Mechanical Engineering equipment. Although the RP has provided appropriate assurance to codes and standards associated to SSCs pressure boundary aspects. The RP has not provided an appropriate level of assurance that its internal procedures used for the design and manufacture of parts of SSCs are adequate given the classification and categorisation of the SSC. In addition, the RP has not provided an appropriate level of assurance that it adequately understands the design requirements to incorporating UK RGP codes and standards to its nuclear ventilation SSCs;
- the RP’s design process is not fully aligned with my expectations from having a consistent equipment selection process. Although, the RP’s plant 60 year design life claim may be reasonable the RP will require to provide the appropriate level of substantiation;
- the RP has provided an adequate level of assurance associated with its operational experience arrangements at an organisational level. Although the RP will require to provide the appropriate level of assurance at an SSC level; and
- the RP has provided a level of assurance that EIM&T is appropriately considered as part of its design process. Although the RP will require to provide the appropriate level of assurance for component replacement frequencies and isolations and configurations.

123. Several topic areas require follow-up to ensure that the RP adequately addresses:

- that the nuclear ventilation system designs are aligned to UK codes, standards and UK RGP;
- that SSCs’ design, qualification, reliability, maintainability and associated operational experience justify that risks have been reduced SFAIRP;
- that the RP adopts a robust, auditable design process with arrangements that set out design principles, rules, and selection criteria for all SSCs;

- that SSCs' qualification are aligned with the 60 year design life claim or the building layout and access provisions are adequate to support a replacement;
- that SSC isolation and configuration for EIM&T are aligned with UK RGP and risks are reduced SFAIRP;
- that SSCs' reliabilities are aligned with the output of the UK ABWR deterministic and probabilistic safety analyses; and
- that the design processes adequately consider operational experience at an SSC level.

124. To facilitate alignment of the above topic areas of the UK ABWR design to my regulatory expectations it is my intention to generate and issue four regulatory observations.

125. The Mechanical Engineering specialism has not engaged technical support contractors to undertake any technical reviews as part of this Step 2 assessment. However, as part of Step 2 I have considered and identified a number of potential topic areas for technical review as part of my Step 3 assessment. A prerequisite to undertaking these technical reviews will be the adequacy of the RP's Step 3 submission, in terms of sufficiency of documents and associated timescales.

126. The Step 2 technical engagement has provided an adequate level of assurance that the RP Step 3 submission will be aligned to my regulatory expectations.

127. Through my Mechanical Engineering assessment I see no reason why the UK ABWR should not proceed to Step 3 of the GDA process. In addition, I have not identified any fundamental shortfalls at this stage that have the potential to prevent the issue of a Design Acceptance Confirmation (DAC).

5.2 Recommendations

128. I make the following recommendations:

- **Recommendation 1**: The UK ABWR should proceed to Step 3 of the GDA process from a Mechanical Engineering perspective;

- **Recommendation 2**: All the items identified in Step 2 as requiring follow-up should be included in ONR's GDA Step 3 assessment plan for the UK ABWR Mechanical Engineering; and

- **Recommendation 3**: All the Step 2 out-of-scope items identified in sub-section 4.9 of this report should be considered as part of ONR's GDA Step 3 assessment plan for the UK ABWR Mechanical Engineering.

6 REFERENCES

1 UK ABWR GDA (Generic Design Assessment) Mechanical Engineering Strategy and Design Process SE-GD-0064 Rev 0; Dated March 2014; TRIM 2014/140793

2 UK ABWR GDA (Generic Design Assessment) Preliminary Safety Report on Mechanical Engineering XE-GD-00149 Rev A; Dated March 2014; TRIM 2014/113639

3 UK ABWR GDA (Generic Design Assessment) Preliminary Safety Report on Mechanical Engineering XE-GD-00149 Rev B; Dated March 2014; TRIM 2014/112254

4 UK ABWR GDA (Generic Design Assessment) Basis of Safety Cases on Standby Gas Treatment System SE-GD-0043 Rev 0; Dated March 2014; TRIM 2014/121121

5 UK ABWR GDA (Generic Design Assessment) Condensate & Feedwater System Basis of Safety Case SBE-GD-0011 Rev 0; Dated February 2014; TRIM 2014/55811

6 UK ABWR GDA (Generic Design Assessment) Basis of Safety Cases on Heating Ventilation and Air Conditioning System HPE-GD-H006 Rev 0; Dated March 2014; TRIM 2014/121074

7 UK ABWR GDA (Generic Design Assessment) Emergency Power Supply Systems Basis of Safety Case VDE-GD-0001 Rev 0; Dated March 2014; TRIM 2014/121068

8 UK ABWR GDA (Generic Design Assessment) Off-Gas System Basis of Safety Case GE-GD-0009 Rev 0; Dated February 2014; TRIM 2014/59744

9 UK ABWR GDA (Generic Design Assessment) Nuclear Boiler System Basis of Safety Case SE-GD-0041 Rev 0; Dated February 2014; TRIM 2014/59769

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11 UK ABWR GDA (Generic Design Assessment) Reactor Recirculation System Basis of Safety Case SE-GD-0037 Rev 0; Dated December 2013; TRIM 2014/1004

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2. ONR Nuclear Safety Technical Assessment Guide; Safety Systems; T/AST/003 Issue 6; July 2011;
3. ONR Nuclear Safety Technical Assessment Guide; Guidance on the demonstration of ALARP (as low as Reasonably Practicable); NS-TAST-GD-005 Rev 6; September 2013;
4. ONR Nuclear Safety Technical Assessment Guide; Nuclear Lifting Operations; T/AST/056 Issue 002; December 2011;
5. ONR Nuclear Safety Technical Assessment Guide; Examination, Inspection, Maintenance, & Testing of Items Important to Safety; NS-TAST-GD-009 Rev 2; November 2012;
6. ONR Nuclear Safety Technical Assessment Guide; Diversity, Redundancy, Segregation and Layout of Mechanical Plant; NS-TAST-GD-036 Rev 3; April 2014;
7. ONR Nuclear Safety Technical Assessment Guide; Ventilation; NS-TAST-GD-022 Rev 2; April 2013;
8. ONR Nuclear Safety Technical Assessment Guide; Containment: Chemical Plants; NS-TAST-GD-021 Rev 2; March 2013; and
9. ONR Nuclear Safety Technical Assessment Guide; Integrity of Metal Components and Structures; NS-TAST-GD-016 Rev 4; March 2013.

www.onr.gov.uk/operational/tech_asst_guides/index.htm

17 IAEA Standards and Guidance:

1. Safety of Nuclear Power Plants: Design, Specific Safety Requirements; SSR-2/1; (2012);
2. Safety Assessment for Facilities and Activities General Safety Requirements Part 4; GSR Part 4; (2009);
3. Seismic Design and Qualification for Nuclear Power Plants Safety Guide; NS-G-1.6; (2003);
4. Design of the Reactor Coolant System and Associated Systems in Nuclear Power Plants Safety guide; NS-G-1.9; (2004);
5. Commissioning for Nuclear Power Plants Safety Guide; NS-G-2.9; (2003);
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8. Aging Management for Nuclear Power Plants Safety Guide; NS-G-2.12; (2009);
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29 Health and Safety Executive; The safe isolation of plant and equipment; HSG253; ISBN 978 0 7176 6171 8; Dated 2006

30 UK ABWR GDA (Generic Design Assessment) Translation of Valve Selection and Configuration Guide (Response to RQ-ABWR-0017) Rev 0; SE-GD-0050; TRIM 2014/85323

31 ONR Generic Design Assessment of HGNE’s Advanced Boiling Water Reactor (ABWR) - Step 3 Assessment Plan for Mechanical Engineering; ONR-GDA-AP-14-008 Revision 0; July 2014. TRIM Ref 2014/231236

32 ONR Generic Design Assessment of HGNE’s Advanced Boiling Water Reactor (ABWR) - Step 2 Assessment Plan for Mechanical Engineering; ONR-GDA-AP-13-013 Revision 0; December 2013. TRIM Ref 2013/416024

33 UK ABWR- SE-GD-0021 Rev 1 – Mechanical Engineering Safety Case Approach; Dated November 2013. TRIM 2013/441274

Table 1

Relevant safety assessment principles considered during the assessment of Mechanical Engineering

SAP No and Title	Description	Interpretation	Comment
EAD	Ageing and degradation		
EAD.1 Safe working life	The safe working life of structures, systems and components that are important to safety should be evaluated and defined at the design stage	TAGs (Ref. 16):	See Sections 4.3.1 and 4.5
EAD.2 Lifetime margins	Adequate margins should exist throughout the life of a facility to allow for the effects of materials ageing and degradation processes on structures, systems and components that are important to safety	TAG. 056 (Ref. 16)	See Section 4.5
EAD.3 Periodic measurement of material properties	Where material properties could change with time and affect safety, provision should be made for periodic measurement of the properties	TAG. 056 (Ref. 16)	See Section 4.5
EAD.4 Periodic measurement of parameters	Where parameters relevant to the design of plant could change with time and affect safety, provision should be made for their periodic measurement	TAG. 056 (Ref. 16)	See Section 4.5

EAD.5 Obsolescence	A process for reviewing the obsolescence of structures, systems and components important to safety should be in place	TAG. 056 (Ref. 16)	See Section 4.5
ECS	Safety classification and standards		
ECS.1 Safety categorisation	The safety functions to be delivered within the facility, both during normal operation and in the event of a fault or accident, should be categorised based on their significance with regard to safety	TAG (Ref. 16): T/AST/003; T/AST/009; T/AST/021 T/AST/056; and T/AST/057	See Sections 4.3.1 and 4.9
ECS.2 Safety classification of structures, systems and components	Structures, systems and components that have to deliver safety functions should be identified and classified on the basis of those functions and their significance with regard to safety	TAG (Ref. 16): T/AST/003; T/AST/009; T/AST/021 T/AST/056; and T/AST/057	See Sections 4.3.1 and 4.9
ECS.3 Standards	Structures, systems and components that are important to safety should be designed, manufactured, constructed, installed, commissioned, quality assured, maintained, tested and inspected to the appropriate standards	TAGs (Ref. 16): T/AST/003; NS-TAST-GD-009; NS-TAST-GD-021 NS-TAST-GD-022 T/AST/056; and T/AST/057	See Sections 4.3.1 and 4.3.2

ERL	Reliability claims		
ERL.1 Form of claims	The reliability claimed for any structure, system or component important to safety should take into account its novelty, the experience relevant to its proposed environment, and the uncertainties in operating and fault conditions, physical data and design methods	TAGS. (Ref. 16): T/AST/003 NS-TAST-GD-009; NS-TAST-GD-021 NS-TAST-GD036; AND T/AST/057	See Section 4.4
EDR	Design for reliability		
EDR.1 Failure to safety	Due account should be taken of the need for structures, systems and components important to safety 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 where appropriate	TAGS (Ref. 16): NS-TAST-GD-005; NS-TAST-GD-009; NS-TAST-GD-021; and T/AST/057	See Sections 4.6; 4.7; 4.8 and 4.9
EDR.2 Redundancy, diversity and segregation	Redundancy, diversity and segregation should be incorporated as appropriate within the designs of structures, systems and components important to safety	TAGS (Ref. 16): NS-TAST-GD-005; NS-TAST-GD-009; NS-TAST-GD-021; and T/AST/057	See Sections 4.6; 4.7 and 4.8

EDR.3 Common cause failure	Common cause failure (CCF) should be explicitly addressed where a structure, system or component important to safety employs redundant or diverse components, measurements or actions to provide high reliability	TAGS (Ref. 16): NS-TAST-GD-005; NS-TAST-GD-009; NS-TAST-GD-021; and T/AST/057	See Sections 4.6; 4.7 and 4.8
EDR.4 Single failure criterion	During any normally permissible state of plant availability no single random failure, assumed to occur anywhere within the systems provided to secure a safety function, should prevent the performance of that safety function	TAGS (Ref. 16): NS-TAST-GD-005 NS-TAST-GD-009; and T/AST/057	See Sections 4.6; 4.7 and 4.8
EMT	Maintenance, inspection and testing		
EMT.1 Identification of requirements	Safety requirements for in-service testing, inspection and other maintenance procedures and frequencies should be identified in the safety case	TAGS (Ref. 16): NS-TAST-GD-005; NS-TAST-GD-009; NS-TAST-GD-021; and T/AST/057	See Sections 4.4 and 4.7
EMT.2 Frequency	Structures, systems and components important to safety should receive regular and systematic examination, inspection, maintenance and testing	TAGS (Ref. 16): NS-TAST-GD-005; NS-TAST-GD-009; NS-TAST-GD-021; and T/AST/057	See Section 4.7

EMT.3 Type-testing	Structures, systems and components important to safety should be type tested before they are installed to conditions equal to, at least, the most severe expected in all modes of normal operational service	TAGS (Ref. 16): NS-TAST-GD-005; NS-TAST-GD-009; NS-TAST-GD-021; and T/AST/057	See Section 4.7
EMT.4 Validity of equipment qualification	The validity of equipment qualification for structures, systems and components important to safety should not be unacceptably degraded by any modification or by the carrying out of any maintenance, inspection or testing activity	TAGS (Ref. 16): NS-TAST-GD-005; NS-TAST-GD-009; NS-TAST-GD-021; and T/AST/057	See Section 4.7
EMT.5 Procedures	Commissioning and in-service inspection and test procedures should be adopted that ensure initial and continuing quality and reliability	TAGS (Ref. 16): NS-TAST-GD-005; NS-TAST-GD-009; NS-TAST-GD-021; and T/AST/057	See Section 4.7
EMT.6 Reliability claims	Provision should be made for testing, maintaining, monitoring and inspecting structures, systems and components important to safety in service or at intervals throughout plant life commensurate with the reliability required of each item	TAGS (Ref. 16): NS-TAST-GD-005; NS-TAST-GD-009; NS-TAST-GD-021; and T/AST/057	See Section 4.7

EMT.7 Functional testing	In-service functional testing of systems, structures and components important to safety should prove the complete system and the safety-related function of each component	TAGS (Ref. 16): NS-TAST-GD-005; NS-TAST-GD-009; NS-TAST-GD-021; and T/AST/057	See Sections 4.5 and 4.7
EMT.8 Effect of internal/external events	Structures, systems and components important to safety should be inspected and/or re-validated after any internal or external event that might have challenged their design basis	TAGS (Ref. 16): NS-TAST-GD-005; NS-TAST-GD-009; NS-TAST-GD-021; and T/AST/057	See Section 4.7
ELO	Layout		
ELO.1 Access	The design and layout should facilitate access for necessary activities and minimise adverse interactions during such activities	TAGS (Ref. 16): NS-TAST-GD-005 NS-TAST-GD-009; and T/AST/057	See Sections 4.4 and 4.6
EMC	Highest reliability components and structures		
EMC.13 Materials	Materials employed in manufacture and installation should be shown to be suitable for the purpose of enabling an adequate design to be manufactured, operated, examined and maintained throughout the life of the facility	TAGS (Ref. 16): NS-TAST-GD-005; NS-TAST-GD-009; NS-TAST-GD-016; NS-TAST-GD-021; and T/AST/057	See Section 4.13

ECV	Containment and Ventilation		
ECV.1 Prevention of leakage	Radioactive substances should be contained and the generation of radioactive waste through the spread of contamination by leakage should be prevented	TAGS (Ref. 16): NS-TAST-GD-022; and T/AST/057	See Section 4.3.2
ECV.2 Minimisation of releases	Nuclear containment and associated systems should be designed to minimise radioactive releases to the environment in normal operation, fault and accident conditions	TAGS (Ref. 16): NS-TAST-GD-022; and T/AST/057	See Section 4.3.2
ECV.3 Means of confinement	The primary means of confining radioactive substance should be by the provision of passive sealed containment systems and intrinsic safety features, in preference to the use of active dynamic systems and components	TAGS (Ref. 16): NS-TAST-GD-022; and T/AST/057	See Section 4.3.2

ECV.4 of containment barriers	Where the radiological challenge dictates, waste storage vessels, process vessels, piping, ducting and drains (including those that may serve as routes for escape or leakage from containment) and other plant items that act as containment for nuclear matter, should be provided with further containment barrier(s) that have sufficient capacity to deal safely with the leakage resulting from any design basis fault	TAGS (Ref. 16): NS-TAST-GD-022; and T/AST/057	See Section 4.3.2
ECV.5 Minimisation of personnel access	The need for access by personnel to the containment should be minimised	TAGS (Ref. 16): NS-TAST-GD-022; and T/AST/057	See Section 4.3.2
ECV.6 Monitoring devices	Suitable monitoring devices with alarms and provisions for sampling should be provided to detect and assess changes in the stored radioactive substances or changes in the radioactivity of the materials within the containment	TAGS (Ref. 16): NS-TAST-GD-022; and T/AST/057	See Section 4.3.2
ECV.7 Leakage monitoring	Appropriate sampling and monitoring systems and other provisions should be provided outside the containment to detect, locate, quantify and	TAGS (Ref. 16): NS-TAST-GD-022; and T/AST/057	

	monitor leakages of nuclear matter from the containment boundaries under normal and accident conditions		
ECV.8 Minimising of provisions	Where provisions are required for the import or export of nuclear matter into or from the facility containments, the number of such provisions should be minimised	TAGS (Ref. 16): NS-TAST-GD-022; and T/AST/057	
ECV.9 Standards	The design should ensure that controls on fissile content, radiation levels, the overall containment and ventilation standards are suitable and sufficient at all times	TAGS (Ref. 16): NS-TAST-GD-022; and T/AST/057	See Section 4.3.2
ECV.10 Safety standards	The safety functions of the ventilation system should be clearly identified and the safety philosophy of the system in normal and fault conditions should be defined in terms of the relative priorities given to the functions associated with the system	TAGS (Ref. 16): NS-TAST-GD-022; and T/AST/057	See Section 4.3.2
EQU	Equipment qualification		
EQU.1 Qualification Procedures	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	TAGS (Ref. 16): T/AST/003; NS-TAST-GD-009; NS-TAST-GD-022; and T/AST/057	See Section 4.4

	their operational lives		
ECM	Commissioning		
ECM.1 Commission testing	Before operating any facility or process that may affect safety it should be subject to commissioning tests to demonstrate that, as built, the design intent claimed in the safety case has been achieved	TAGS (Ref. 16): T/AST/003; NS-TAST-GD-009; NS-TAST-GD-022; and T/AST/057	See Section 4.5