Office for Nuclear Regulation

ONR Assessment Report

Generic Design Assessment of the BWRX-300 – Step 2 Assessment Report – Mechanical Engineering

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Executive summary

In December 2024, the Office for Nuclear Regulation (ONR), together with the Environment Agency and Natural Resources Wales, began Step 2 of the Generic Design Assessment (GDA) of the BWRX-300 design on behalf of GE Vernova Hitachi Nuclear Energy International LLC, United Kingdom (UK) Branch, the Requesting Party (RP).

This report presents the outcomes of my mechanical engineering assessment of the BWRX-300 design as part of Step 2 of the ONR GDA. This assessment is based upon the information presented in the RP's safety, security, safeguards, and environment cases (SSSE), the associated revision 3 of the Design Reference Report and supporting documentation.

ONR's GDA process calls for an assessment of the RP's submissions, which increases in detail as the project progresses. The focus of my assessment in this step was to support ONR's decision on the fundamental adequacy of the BWRX-300 design and safety case, and the suitability of the methodologies, approaches, codes, standards and philosophies which form the building blocks for the design and generic safety, security and safeguards cases.

I targeted my assessment, in accordance with my assessment plan, at the areas that were fundamental to the acceptability of the design and methods for deployment in Great Britan (GB), benchmarking my regulatory judgements against the expectations of ONR's Safety Assessment Principles (SAPs), Technical Assessment Guides (TAGs) and other guidance which ONR regards as relevant good practice, such as International Atomic Energy Agency (IAEA) safety, security and safeguards standards. Where appropriate, I have also considered how I could use relevant learning and regulatory conclusions from the UK ABWR GDA to inform my assessment of the BWRX-300.

I have assessed thematic areas, targeting mechanical aspects that are novel and play a principal role in nuclear safety. Based upon my assessment, I have concluded that the RP has:

- Produced an SSSE which is logical and hierarchical aligning with relevant good practice. A future SSSE should consider improving the traceability between the safety analysis and engineering requirements;

- Adequately demonstrated how its limits and conditions development considers relevant good practice and is still developing;

- Adequately demonstrated that its defence in depth arrangements align with relevant good practice and implemented appropriate independence in mechanical engineering structures, systems and components in different defence in depth levels;

- Adequately demonstrated the isolation condenser system concept design qualification approach. A future SSSE should demonstrate adequate BWRX-300 isolation condenser system qualification;

- Adequately considered overpressure resilience in the BWRX-300 design. It has proposed use of a proven engineering technology in a novel application by including a bursting disc to relieve reactor pressure under accident conditions. A future SSSE should qualify the bursting disk and demonstrate that examination, inspection, maintenance and testing requirements are achievable and reduce risks as low as reasonably practicable;

- Adequately implemented design assurance arrangements which support valve selection including the reactor isolation valves. This includes redundancy and independence design rules. A future SSSE should consider and justify the reactor isolation valve mechanical diversity and demonstrate adequate qualification;

- Implemented some design improvements relative to previous BWR designs. A future SSSE should consider opportunities to reduce fuel route dropped load consequences, such that risks are reduced as low as reasonably practicable;

- Demonstrated it is working to reduce risks as low as reasonably practicable. It has identified appropriate relevant good practice, identified and mitigated risks and demonstrated an adequate optioneering approach. A future SSSE should continue to undertake analysis and produce evidence that risks have been reduced as low as reasonably practicable as the design matures;

- Established an appropriate methodology for its categorisation and classification principles' development. A future SSSE should include dropped load analysis to inform the nuclear lifting equipment categorisation and classification;

- Not yet identified appropriate qualification arrangements specific to safety class 2 components. However, a graded approach has been adopted which may result in proportionate equipment qualification being applied to safety class 2 structures, systems and components. A future SSSE should provide evidence that equipment qualification procedures provide a level of confidence commensurate with structures, systems and components safety classification;

- Implemented EIMT arrangements, which ensure adequate space is available to conduct planned activities. Its arrangements align with safe plant isolation relevant good practice; and,

- Implemented adequate ageing and degradation management arrangements which identify foreseeable structures, systems and components degradation mechanisms.

Overall, based on my assessment I have not identified any fundamental safety shortfalls that could prevent ONR permissioning the construction of a power station based on the generic BWRX-300 design; noting that any decision to permission a

BWRX-300 will require further assessment (in either a future Step 3 GDA or during site specific activities) of suitable and sufficient supporting evidence that can substantiate the claims and proposals made in the GDA Step 2 submissions.

List of abbreviations

ALARP	As low as Reasonably Practicable
ABWR	Advanced Boiler Water Reactor
BWR	Boiling Water Reactor
CNSC	Canadian Nuclear Safety Commission
DAC	Design Acceptance Confirmation
DiD	Defence in Depth
EIMT	Examination, Inspection, Maintenance and Testing
ESBWR	Economic Simplified Boiling Water Reactor
EQ	Equipment Qualification
GB	Great Britain
GDA	Generic Design Assessment
GVHA	GE Vernova Hitachi Nuclear Energy Americas LLC
HVAC	Heating Ventilation and Air Conditioning
IAEA	International Atomic Energy Agency
ICS	Isolation Condenser System
LCSO	Limits and Conditions of Safe Operations
NRC	Nuclear Regulatory Commission
ONR	Office for Nuclear Regulation
OPEX	Operating Experience
PCCS	Passive Containment Cooling System
PIE	Postulated Initiating Event(s)
PSR	Preliminary Safety Report
RGP	Relevant Good Practice
RIV	Reactor Isolation Valve
RP	Requesting Party
RQ	Regulatory Query
SC	Safety Classification
SSSE	Safety, Security, Safeguards and Environment Cases
SAP	Safety Assessment Principle(s)
SSCs	Structure, System and Components
TAG	Technical Assessment Guide(s) (ONR)
TIG	Technical Inspection Guide(s) (ONR)
TSC	Technical Support Contractor
UK	United Kingdom
US	United States of America
WENRA	Western European Nuclear Regulators' Association

Contents

1. Introduction

1. This report presents the outcome of my mechanical engineering assessment of the BWRX-300 design as part of Step 2 of the Office for Nuclear Regulation (ONR) Generic Design Assessment (GDA). My assessment is based upon the information presented in the Safety, security, safeguards and environment cases (SSSE) head document [1], specifically chapters (refs. [2], [3], [4], [5], [6], [7], [8], [9], [10], [11]) the associated revision of the design reference report (ref. [12]) and supporting documentation.

2. Assessment was undertaken in accordance with the requirements of ONR Management System and follows ONR's guidance on the mechanics of assessment, NS-TAST-GD-096 (ref. [13]) and ONR's risk informed, targeted engagements (RITE) guidance (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.

3. This is a major report as per ONR's guidance on the production of reports, NS-PER-GD-108 (ref. [17]).

1.1. Background

4. The ONR's GDA process (ref. [18]) calls for an assessment of the Requesting Party's (RP) submissions with the assessments increasing in detail as the project progresses. This GDA will be finishing at Step 2 of the GDA process. For the purposes of the GDA, GE Vernova Hitachi Nuclear Energy International LLC, United Kingdom (UK) Branch, is the RP. GE Vernova Hitachi Nuclear Energy Americas LLC (GVHA) is a provider of advanced reactors and nuclear services and is the designer of the BWRX-300. GVHA is headquartered in Wilmington, North Carolina, United States of America (US).

5. In Step 1, and for the majority of Step 2, the RP was known as GE-Hitachi Nuclear Energy International LLC, UK Branch, and GVHA as GE-Hitachi Nuclear Energy Americas LLC. The entities formally changed names in October 2025 and July 2025 respectively. The majority of the submissions provided by the RP during GDA were produced prior to the name change, and thus the reference titles in Section 6 of this report reflects this.

6. In the UK, the RP has been supported by its supply chain partner, Amentum, who has assisted the RP in the development of the UK-specific chapters of the SSSE, and other technical documents for the GDA.

7. In January 2024 ONR, together with the Environment Agency and Natural Resources Wales began Step 1 of this two-Step GDA for the generic BWRX-300 design.

8. Step 1 is the preparatory part of the design assessment process and is associated with initiation of the project and preparation for technical assessment in Step 2. Step 1 completed in December 2024. Step 2 is the first substantive technical assessment step and began in December 2024 and will complete in December 2025.

9. The RP has stated that currently it has no plans to undertake Step 3 of GDA and obtain a Design Acceptance Confirmation (DAC). It anticipates that any further assessment by the UK regulators of the BWRX-300 design will be on site-specific basis and with a future licensee.

10. The focus of ONR's assessment in Step 2 was:

- The fundamental adequacy of the design and safety, security, and safeguards cases; and,
- The suitability of the methodologies, approaches, codes, standards, and philosophies which form the building blocks for the design and cases.

11. The objective is to undertake an assessment of the design against regulatory expectations to identify any fundamental safety, security or safeguards shortfalls that could prevent ONR permissioning the construction of a power station based on the design.

12. Prior to the start of Step 2 I prepared a detailed Assessment Plan for mechanical engineering [19]. This has formed the basis of my assessment and was also shared with the RP to maximise openness and transparency.

13. This report is one of a series of assessments which support ONR's overall judgements at the end of Step 2 which are recorded in the Step 2 Summary Report (ref. [20]) and published on the regulators' website.

1.2. Scope

14. The assessment documented in this report is based upon the SSSE for the BWRX-300 (refs. [1], [2], [3], [4], [21], [5], [6], [22], [23], [7], [24], [8], [25], [26], [27], [28], [9], [29], [30], [31], [32], [33], [34], [35], [36], [37], [10], [38], [39], [40], [41], [42], [43], [44], [45], [46], [47], [11], [48]).

15. The RP's GDA scope has been agreed between the regulators and the RP during Step 1. This is documented in an overall Scope of Generic Design Assessment report (ref. [49]). This is further supported by its design reference report (ref. [12]) and the Master Document Submission List (ref. [50]). The GDA scope report documents the submissions which were provided in each topic area during Step 2 and provides a brief overview of the physical and functional scope of the nuclear power plant that is proposed for consideration in the GDA. The design reference report provides a list of the structures, systems and components (SSCs) which are included in the scope of the GDA, and their relevant GDA reference design documents.

16. The RP has stated it does not have any current plans to undertake GDA beyond Step 2. This has defined the boundaries of the GDA and therefore of my own assessment.

17. The GDA scope includes the Power Block (comprising the Reactor Building, Turbine Building, Control Building, Radwaste Building, Service Building, Reactor Auxiliary Structures) and protected areas as well as the balance of plant. It includes all modes of operation.

18. The regulatory conclusions from GDA apply to everything that is within the GDA scope. However, ONR does not assess everything within it or all matters to the same level of detail. This applies equally to my own assessment, and I have followed ONR's guidance on the mechanics of assessment, NS-TAST-GD-096 (ref. [13]) and ONR's guidance on Risk Informed, Targeted Engagements (ref. [14]).

19. As appropriate for Step 2 of the GDA, information has not been submitted for all aspects within the GDA Scope during Step 2. The following aspects of the SSSE are considered beyond GDA Step 2 and are therefore out of scope of this assessment:

- Load schedule production setting out key parameters of all lifting and handling operations;
- A demonstration that the analysis methods/computer programmes are adequate to assess the SSCs and have been adequately validated;
- Design code assessments for the major SSCs; and,
- Design substantiation of mechanical SSCs.

20. I have assessed mechanical aspects that are novel and safety significant. This is recorded in my assessment plan (ref. [19])

2. Assessment standards and interfaces

21. The primary goal of the GDA Step 2 assessment is to reach an independent and informed judgment on the adequacy of the RP's SSSE for the reactor technology being assessed.

22. ONR has a range of internal guidance to enable Inspectors to undertake a proportionate and consistent assessment of such cases. This section identifies the standards which have been considered in this assessment. This section also identifies the key interfaces with other technical topic areas.

2.1. Standards

23. The ONR Safety Assessment Principles (SAPs) (ref. [15]) constitute the regulatory principles against which the RP's case is judged. Consequently, the SAPs are the basis for ONR's assessment and have therefore been used for the Step 2 assessment of the BWRX-300.

24. The International Atomic Energy Agency (IAEA) safety standards (ref. [51]) and nuclear security series (ref. [52]) are a cornerstone of the global nuclear safety and security regime. They provide a framework of fundamental principles, requirements, and guidance. They are applicable, as relevant, throughout the entire lifetime of facilities and activities.

25. Furthermore, ONR is a member of the Western European Nuclear Regulators Association (WENRA). WENRA has developed Reference Levels (ref. [53]), which represent good practices for existing nuclear power plants, and Safety Objectives for new reactors (ref. [54]).

26. The relevant SAPs, IAEA standards and WENRA reference levels are embodied and expanded on in the TAGs (ref. [16]). The TAGs provide the principal means for assessing the mechanical engineering aspects in practice.

27. The key guidance is identified below and referenced where appropriate within Section 4 of this report. Relevant good practice, where applicable, has also been cited within the body of this report.

2.1.1. Safety assessment principles (SAPs)

28. The key SAPs applied within my assessment are:

- FP.6 (Prevention of accidents). This is relevant to claims that risks have been reduced as low as reasonably practicable (ALARP);
- SC.4 (Safety case characteristics). This is relevant to claims that the BWRX-300 design has applied conservative design principles;
- EKP.5 (Safety measures). This is relevant where claims are made regarding conservative design principles;
- ECS.3 (Codes and standards). This is relevant to the RP's claims that relevant good practice has informed the BWRX-300 design;
- EQU.1 (Qualification procedures). This is relevant to the RP's claims that it has a robust approach to verification and validation;
- EMT.1 (Identification of requirements). This is relevant to the RP's examination, inspection, maintenance and testing (EIMT) claims;

- EAD.1 (Safe working life) and EAD.2 (Lifetime margins). These are relevant to ageing and degradation claims; and,
- ELO.1 (Access). This is relevant to through life access and egress claims

29. A list of the SAPs used in this assessment is recorded in Appendix 1.

2.1.2. Technical assessment guides (TAGs)

30. The following TAGs have been used as part of this assessment:

- NS-TAST-GD-005 - Regulating duties to reduce risks to ALARP (ref. [55])
- NS-TAST-GD-006 - Design Basis Analysis (re. [56])
- NS-TAST-GD-009 - Examination, Inspection, Maintenance and Testing of Items Important to Safety (ref. [57])
- NS-TAST-GD-022 - Ventilation (ref. [58])
- NS-TAST-GD-036 - Redundancy, Diversity, Segregation and Layout of Structures, Systems and Components (ref. [59])
- NS-TAST-GD-056 - Nuclear Lifting Operations (ref. [60])
- NS-TAST-GD-057 - Design Safety Assurance (ref. [61])
- NS-TAST-GD-067 - Pressure System Safety (ref. [62])
- NS-TAST-GD-094 - Categorisation of Safety Functions and Classification of Structures, Systems and Components (ref. [63])
- NS-TAST-GD-096 - Guidance on the Mechanics of Assessment (ref. [13])
- NS-TAST-GD-109 - Ageing and Degradation Management (ref. [64])

2.1.3. National and international standards and guidance

31. The following standards and guidance have been used as part of this assessment:

- IAEA SSR-2/1 - Safety of Nuclear Power Plants: Design (ref. [65])
- IAEA SSG-56 - Design of Reactor Coolant System and Associated Systems for Nuclear Power Plants (ref. [66])

- IAEA SSG-61 - Format and Content of the Safety Analysis Report for Nuclear Power Plants (ref. [67])
- IAEA SSG-67 - Seismic Design for Nuclear Installations (ref. [68])
- IAEA SSG-69 - Equipment Qualification for Nuclear Installations (ref. [69])
- IAEA SSG-70 - Operational Limits and Conditions and Operating Procedures for Nuclear Power Plants (ref. [70])
- IAEA NS-G-1.13 - Radiation Protection Aspects of Design for Nuclear Power Plants (ref. [71])
- IAEA SRS-46 - Assessment of Defence in Depth for Nuclear Power Plants (ref. [72])
- HSE HSG 253 - The safe isolation of plant and equipment [73]

2.2. Integration with other assessment topics

32. To deliver the assessment scope described above I have worked closely with other topics to inform my assessment. Similarly, other assessors sought input from my assessment. These interactions are key to the success of GDA to prevent or mitigate any gaps, duplications, or inconsistencies in ONR's assessment.

33. The key interactions with other topic areas were:

- Civil engineering - regarding the RP's consideration of modularisation construction methods;
- Internal hazards - regarding the RP's approach to mechanical handling and drop loads;
- Fault studies - regarding the RP's approach to safety function categorisation and SSC classification;
- Nuclear site health and safety - regarding the RP's application of principles of prevention at the early design stage;
- Structural integrity – regarding the RP's approach to ageing and degradation management and valve selection;
- Human factors - regarding the RP's approach to heating, ventilation and air conditioning (HVAC) main control room habitability;
- Electrical engineering - regarding the RP's approach to engineering traceability and the BRWX-300 electrical generating frequency;

- Control and instrumentation - regarding the RP’s approach to engineering traceability, diversity in reactor isolation valves (RIVs) and HVAC claims; and,

- Radiation protection - regarding the RP’s use of cobalt-based materials and maintenance activities.

2.3. Use of technical support contractors

34. During Step 2 I have not engaged Technical Support Contractors (TSCs) to support my assessment.

3. Requesting party’s submission

35. The RP submitted the SSSE at the start of Step 2 in four volumes that integrate environmental protection, safety, security, and safeguards. This was accompanied by a head document (ref. [1]), which presents the integrated GDA environmental, safety, security, and safeguards case for the BWRX-300 design.

36. All four volumes were subsequently consolidated to incorporate any commitments and clarifications identified in regulatory engagements, regulatory queries, and regulatory observations, and were resubmitted in July 2025. This consolidated revision is the basis of the regulatory judgements reached in Step 2.

37. This section presents a summary of the RP’s SSSE for mechanical engineering. It also identifies the documents submitted by the RP which have formed the basis of my Step 2 assessment of the BWRX-300 design.

3.1. Summary of the BWRX-300 design

38. The BWRX-300 is a single unit, direct-cycle, natural circulation, boiling water reactor with a power of ~870 MW (thermal) and a generating capacity of ~ 300 MW (electrical) and is designed to have an operational life of 60 years. The RP claims the design is at an advanced concept stage of development and is being further developed during the GDA in parallel with the RP’s SSSE.

39. The BWRX-300 is the tenth generation of the boiling water reactor (BWR) designed by GVHA and its predecessor organisations. The BWRX-300 design builds upon technology and methodologies used in its earlier designs, including the ABWR, Simplified Boiling Water Reactor (SBWR) and the Economic Simplified Boiling Water Reactor (ESBWR). The ABWR has been licensed, constructed and is currently in operation in Japan, and a UK version of the design was assessed in a previous GDA with a view to potential deployment at the Wylfa Newydd site. Neither the SBWR or ESBWR have been built or operated.

40. The BWRX-300 reactor core houses 240 fuel assemblies and 57 control rods inside a steel reactor pressure vessel (RPV). It uses fuel assemblies (GNF2) that are already currently widely used globally (ref. [21]).

41. The reactor is equipped with supporting systems for normal operations and a range of safety measures are present in the design to provide cooling, control criticality and contain radioactivity under fault conditions. The BWRX-300 uses natural circulation and passive cooling rather than active components, reflecting the RP's design philosophy.

42. There are novel aspects I consider relevant to my mechanical engineering assessment. These include:

- compact layout
- double reactor isolation valves
- no reactor circuit pressure relief valves
- passive safety systems

3.2. BWRX-300 case approach and structure

43. The RP has submitted information on its strategy and intentions regarding the development of the SSSE (refs. [74], [75], [76], [77]). This was submitted to ONR during Step 1.

44. The RP has submitted a SSSE for the BWRX-300 that claims to demonstrate that the standard BWRX-300 can be constructed, operated, and decommissioned on a generic site in Great Britan (GB) such that a future licensee will be able to fulfil its legal duties for activities to be safe, secure and will protect people and the environment. The SSSE comprises a Preliminary Safety Report (PSR) which also includes information on its approach to safeguards and security, a Security Assessment, and a Preliminary Environment Report (PER), and their supporting documents.

45. The format and structure of the PSR largely aligns with the IAEA guidance for safety cases, SSG-61 (ref. [67]), supplemented to include UK specific chapters such as Structural Integrity and Chemistry. The RP has also provided a chapter on ALARP, which is applicable to all safety chapters. The RP has stated that the design and analysis referenced in the PSR is consistent with the March 2024 Preliminary Safety Analysis Report has been submitted to the US Nuclear Regulatory Commission (NRC). The Security Assessment and PER are for the same March 2024 design but have more limited links to any US or Canadian submissions.

3.3. Summary of the requesting party's case for mechanical engineering

46. The aspects covered by the BWRX-300 SSSE in mechanical engineering can be broadly grouped under the following headings summarised as follows:

- PSR Chapter 3: Safety Objectives and Design Rules for Structures, Systems, and Components (ref. [4]). This contains the RP's top-level objective, that "The BWRX-300 is capable of being constructed, operated, and decommissioned in accordance with the standards of environmental, safety, security and safeguard protection required in the UK";
- PSR Chapter 5: Reactor Coolant System and Associated Systems (ref. [5]), describes the relevant Reactor Coolant systems and the associated system groups;
- PSR Chapter 6: Engineered Safety Features (ref. [6]) describes the BWRX-300 engineered safety features;
- PSR Chapter 9A: Auxiliary Systems (ref. [7]) introduces the BWRX-300 auxiliary systems including the fuel handling, HVAC and overhead lifting equipment;
- PSR Chapter 10: Steam and Power Conversion Systems (ref. [8]) introduces the BWRX-300 steam and power conversion systems including the turbine equipment;
- PSR Chapter 15: Safety Analysis (ref. [9]) presents the RP's safety analysis approach. It evaluates the defence lines (DL) that are an integral part of the safety strategy, founded on the Defence in Depth (DiD) concept; and,
- PSR Chapter 16: Operational Limits and Conditions (ref. [10]) describes how the safe operating envelope is evaluated and implemented through operational limits and conditions.

3.3.1. ALARP

47. The RP presents its high-level ALARP summary in PSR Chapter 27 (ref. [11]). This contains its holistic high-level approach to reducing risks ALARP.

48. The engineering chapters contain specific ALARP arguments. This supports the holistic ALARP argument by providing the claims and arguments made within each chapter and references the appropriate evidence. The RP states that the evidence demonstrates that:

- Relevant good practice (RGP) has demonstrably been followed;
- Operating experience (OPEX) has been taken into account within the design process; and,
- All reasonably practicable options to reduce risk have been incorporated within the design.

3.4. Basis of assessment: requesting party's documentation

49. The principal documents that have formed the basis of my mechanical engineering assessment of the SSSE are chapters 3, 5, 6, 9A and 10 (refs. [21], [5], [6], [7], [8]). These are supported by system design description documents and detailed tier 3 documentation. These are referenced as appropriate throughout the report.

3.5. Design maturity

50. My assessment is based on revision 3 of the design reference report (ref. [12]). The design reference report presents the baseline design for GDA Step 2, outlining the physical system descriptions and requirements that form the design at that point in time.

51. The reactor building and the turbine building, along with most of the significant SSCs are housed within the 'power block.' The power block also includes the radwaste building, the control building and a plant services building. For security, this also includes the protected area boundary and the protected area access building.

52. The GDA Scope Report (ref. [49]) describes the RP's design process that extends from baseline (BL) 0 (where functional requirements are defined) up to BL3 (where the design is ready for construction).

53. In the March 2024 design reference, SSCs in the power block are stated to be at BL1. BL1 is defined as:

- system interfaces established
- (included) in an integrated 3D model
- instrumentation and control aspects have been modelled
- deterministic and probabilistic analysis has been undertaken
- system descriptions developed for the primary systems

54. The balance of plant remains at BL0 for which only plant requirements have been established, and SSC design remains at a high concept level.

55. I consider the BWRX-300 design to be in the early stages of maturity. I recognise that further detailed considerations along with qualification evidence is expected during detailed design.

4. ONR assessment

4.1. Assessment strategy

56. The objective of my GDA Step 2 assessment was to reach an independent regulatory judgement on the fundamental aspects of the BWRX-300 design, relevant to mechanical engineering as described in sections 1 and 3 of this report. My assessment strategy is set out in this section and defines how I have chosen which matters to target for assessment. My assessment is consistent with the delivery strategy for the GVHA BWRX-300 GDA (ref. [78]).

57. GVHA is currently engaging with regulators internationally, including the US NRC and the Canadian Nuclear Safety Commission in Canada (CNSC). It is proposing a standard BWRX-300 design for global deployment with minimal design variations from country to country. My assessment takes cognisance of work undertaken by overseas regulators where appropriate.

58. Whilst there is no operating BWR plant in the UK, ONR has previously performed a four-step GDA on the Hitachi-GE UK ABWR (ref. [79]). I have taken learning from this previous activity, targeting my assessment on those aspects of the BWRX-300 which are novel or specific to this design. I have not looked to reassess inherent aspects of BWR technology which were considered in significant detail for the UK ABWR and judged to be acceptable.

59. My assessment plan (ref. [19]) details the assessment strategy.

4.2. Assessment scope

60. My assessment scope and the areas I have chosen to target for my assessment are set out in this section.

61. My assessment scope is consistent with the GDA scope agreed between the regulators and the RP during Step 1 and detailed in Section 1.2 of this report. I have targeted my assessment within this scope.

62. In line with the objectives for Step 2, I have undertaken a broad review of the highest level, fundamental claims and supporting arguments related to mechanical engineering. To support this, I have sampled a targeted set of the claims or arguments as set out below. Where applicable, I have also sampled the evidence available to support any claims and arguments.

63. To fulfil the aims for the Step 2 assessment of the BWRX-300, I have assessed the following thematic areas, targeting areas of novelty and mechanical aspects with an important role in nuclear safety:

- SSSE adequacy
- design parameters
- defence in depth
- passive safety
- design safety assurance
- modularisation
- containment
- redundancy, diversity, segregation
- ALARP and relevant good practice
- categorisation of safety functions and classification of mechanical structures, systems, and components
- equipment qualification
- examination, inspection, maintenance, and testing

4.3. Assessment

4.3.1. Safety case

64. To assess the adequacy of the RP's SSSE I sought to determine if it:

- Has appropriate scope and structure; and
- Is sufficiently complete to enable a meaningful assessment.

65. I sought to judge the safety functional requirements traceability (golden thread) through the SSSE. This included the:

- safety function identification
- safety categorisation
- SSC performance requirements

66. I assessed the RP's introduction to its SSSE (ref. [2]) and noted that the RP's case uses a hierarchical structure to support accessibility. I am satisfied that the RP's case is logical and accessible. This is because the

hierarchical structure provides a claims, arguments, and evidence trail. I consider that it aligns with IAEA SSG-61 (ref. [67]). This satisfies ONR SAP SC.2 (Safety case process outputs) expectations.

67. I assessed the RP's approach to safety function categorisation and SSC classification. Its methodology is presented in SSSE chapter 3 (ref. [4]) and has been written with the intention of aligning with ONR expectations. I discuss this later in this report (see section 4.3.11). It uses fault analysis (ref. [80]) to identify primary safety functions to deliver the required defence lines with the appropriate categorisation and classification applied. This satisfies ONR SAPs SC.2 (Safety case process outputs) and SC.4 (Safety case characteristics) expectations.

68. I sought assurance that the RP's safety case methodology requires an auditable and transparent link between the safety analysis and the engineering requirements. I sampled the Isolation Condenser System (ICS) condensate return valves and the fine motion control rod drive mechanisms. The RP's information (ref. [81]) presented links through:

- fault identification
- system requirements
- classification and categorisation methodology
- component requirements

69. The RP presented these links across multiple safety case documents. I consider that the traceability from the safety analysis to engineering requirements is complex. The RP has not developed an engineering schedule (or utilised a similar methodology), something that may allow the RP to simplify traceability throughout the SSSE (ref. [82]). ONR SAP paragraph 86 expects that there is a clear trail from safety claims through arguments to evidence. I consider that the clarity within the RP's submission could be improved. However, I am content that there are no fundamental shortfalls preventing development of the associated SSSE evidence to support any future permissioning activities.

4.3.2. Design parameters

70. I sought to judge whether the RP has identified and substantiated key generic design parameters, including limits and conditions of safe operations (LCSO).

71. I assessed chapter 16 (ref. [10]). This presents evidence to support the RP's claim that the design has adequate LCSO. I noted that the RP identifies:

- Appropriate LCSO RGP (IAEA SSG-70 ref. [83]);

- How it will establish minimum requirements for the safe operation of the facility;
- How it will conduct testing to mitigate and meet safety standards;
- That its design rules provide important input; and,
- That LCSOs are being identified as the design progresses.

72. I recognise that the BWRX-300 design and SSSE is still under development, so I am satisfied that the RP has not identified all limits and conditions at this stage. I consider the RP has adequately demonstrated how its LCSO development approach considers RGP and that it is still developing its LCSO. For Step 2, I judge that this satisfies ONR SAP SC.2 (Safety case process outputs) expectations.

73. I targeted the design parameters application for the passive containment cooling system (PCCS) due to its novel nature. The PCCS provides containment cooling during the limiting design basis accident for the containment pressure and temperature. These are large break loss of coolant accidents from main steam or feedwater pipe breaks. The PCCS uses the equipment pool as its heat sink. I sampled the primary containment system design description (ref. [84]) and identified the following design parameter examples:

- Each PCCS train shall provide the capability to passively reject a minimum of 50% of the required heat load during off-normal conditions to maintain acceptable containment pressure and temperature limits;
- The minimum total surface area of two PCCS trains shall be 160 m^2;
- The maximum PCCS equipment pool temperature shall be 43.3°C during through all operating modes. ; and,
- PCCS pipe supports shall not reduce the heat transfer area greater than 5%.

74. In response to RQ-01801 (ref. [85]), the RP provided clarification on how the PCCS design parameters support containment cooling safety functions. The RP has undertaken analysis (refs. [33], [31]) that demonstrates the operational limits for containment shell temperature of 165.6°C and containment pressure of 463.5 kPa will not be surpassed during the following faults:

- Feedwater pipe break (peak pressure 407 kPa, peak temperature 134°C);
- Main steam pipe break (peak pressure 423 kPa, peak temperature 134°C);

- Small steam pipe break (peak pressure 191 kPa, peak temperature 125°C); and
- Small liquid pipe break (peak pressure 191 kPa, peak temperature 110°C).

75. I am content that the RP has begun to establish appropriate design parameters for the PCCS system enabling it to deliver its safety functions. I consider the expectations of ONR SAPs EKP.4 (Safety function) and EHT.1 (Design) to be met. However, the PCCS design is still developing, and further analysis is required. The RP recognises this within its response to RQ-01801 (ref. [85]). A future SSSE should provide evidence of suitable engineering analysis and qualification for the BWRX-300 PCCS design, commensurate with its safety classification. I am content that there are no fundamental shortfalls preventing development of the BWRX-300 design and associated SSSE to support future permissioning activities.

4.3.3. Defence in depth

76. To assess the RP's DiD approach, I sampled its safety case objectives and design rules for SSCs (ref. [4]). I noted that the RP claims it applies DiD principles against postulated initiating events (PIE) by providing practicable independent measures. The RP presents independent levels of DiD as per IAEA SSR-2/1 (ref. [65]). I consider that the RP's DiD framework aligns with ONR SAP EKP.3 (Defence in depth) expectations. However, the RP has only applied this to reactor faults. ONR's fault studies assessor has considered DiD further within their report (ref. [86]).

77. I assessed the DiD application to the ICS (ref. [6]) due to its safety significance and novelty. I consider this a novel emergency core cooling approach as there is no mechanically driven flow. The ICS is a defence line 3 system, performing a category A safety function to provide heat removal capabilities during design basis conditions. I identified that the RP has applied DiD to the design through:

- Prevention of abnormal operation through conservative design and RGP application. This is because the RP has considered bounding conditions in its analysis;
- ICS independence from the BWRX-300 duty core heat removal systems;
- ICS component classification, redundancy, diversity and segregation, (defence line 1):
  - three independent class 1 ICS trains
  - independent steam supply lines
  - independent condensate return lines

    - diverse and redundant condensate return valves

- Anticipatory activation of one ICS train to control key plant parameters for anticipated operational occurrences (defence line 2); and,
- Activation of all ICS trains to support design basis accident functions within the first 72 hours (defence line 3).

78. I judge, from a mechanical engineering perspective, that the DiD applied to the design of the ICS system is adequate. This is because the DiD principles have been recognised in the RP's design rules (ref. [4]), and these are being applied through provision of independent and redundant mechanical engineering SSCs at different DiD levels. The RP's approach aligns with ONR SAP EKP.3 (Defence in depth) and IAEA SRS-46 (ref. [72]). I am content that there are no fundamental shortfalls preventing development of the generic BWRX-300 design and associated SSSE evidence to support future permissioning activities.

4.3.4. Passive safety

79. I assessed the adequacy of the RP's approach to passive safety. I sought assurance that the engineering analysis supports the claims made on the passive safety systems. I targeted the following due to their novel safety case claims and relevance to nuclear safety:

- Isolation condenser system. This provides a passive emergency cooling function which I consider a novel approach within GB; and,
- Design response to loss of coolant. Pressure relief valves are not utilised within the BWRX-300 reactor circuit design. I consider this to be a novel approach within GB.

4.3.4.1. Isolation condenser system

80. I targeted the ICS concept design qualification and in particular, demonstration of its passive heat removal capability.

81. I assessed chapter 6 (ref. [6]) and the ICS system design description (ref. [87]). I was unable to identify ICS design parameters qualification evidence. The RP provided additional clarification (ref. [85]) stating it had selected design parameters based on the previously qualified ESBWR. The qualification undertaken included full scale prototype testing and modelling. This qualification concluded that a 33.75 MW (thermal) heat removal capacity per isolation condenser train provided sufficient ESBWR design cooling capacity. While I have not assessed the ESBWR testing or analysis adequacy, the RP provided the design records (for information only) which underpin the ESBWR qualification. I noted that the RP conducted the analysis using higher pool temperatures than those required for the BWRX-300. The underpinning prototype testing used a 100°C cooling pool

temperature with 289°C saturated steam. The BWRX-300 design will have 43°C maximum pool temperature.

82. Following a reactor scram on high reactor pressure at 7.787 MPaG (294°C) the ICS initiates at 3 progressively higher set points. These range between 8.289 MPaG (298°C) and 8.719 MPaG (302°C). These initial set points are higher than the ESBWR test conditions. However, the BWRX-300 ICS has a higher temperature difference (due to cooling pool temperature) which serves to increase its heat removal capacity. I noted that the deterministic analysis using the above values, presented within chapters 15.5 (ref. [33]) and 15.3 (ref. [31]), shows a 2.5 MPa pressure drop from the initial conditions in the first 5 minutes of ICS initiation. This shows that one ICS has capacity to maintain sufficient margin to the reactor pressure vessel design pressure of 10.34 MPaG (ref. [87]).

83. I also sought assurance that the RP has adequately considered the ICS cooling pool capacity. Within its response to RQ-01801 (ref. [85]), I noted that calculations were undertaken showing that that a single isolation condenser pool can meet the required 72-hour endurance period. These calculations are claimed to use conservative input assumptions. For example, pools are at minimum water level on ICS initiation and no condensate occurs on pool structures with the assumption that all steam is lost. The analysis shows that, at the end of the 72-hour period, with one ICS cooling train in operation, 50.6% of the smaller cooling pool (A) volume remains available. Following a 7-day extended deployment of the ICS in the larger pools (B&C) 7.3% volume remains available.

84. I judge that the RP has provided sufficient evidence to demonstrate adequate ICS heat removal qualification for the purposes of GDA step 2 assessment. It has also provided adequate evidence to demonstrate the cooling pool capacity is sufficient to meet the ICS demands for its 72-hour and 7-day safety case performance requirement. The RP has provided this assurance though the application of testing and other engineering analysis captured within its design records, highlighted in its response to RQ-01801 (ref. [85]). In my opinion, this meets the expectations of ONR SAPs EQU.1 (Qualification procedures), EHT.2 (Coolant inventory and flow) and ECS.5 (Use of experience, test or analysis).

85. I note that the engineering analysis was undertaken for the ESBWR design. The records also identify the need for further ICS design qualification. The ICS has developed since the ESBWR design. A future SSSE should provide evidence that the ICS is qualified specifically for the BWRX-300 design and is supported by procedures that provide a level of confidence commensurate with its safety classification (ONR SAP EQU.1 (Qualification procedures)). It should also consider current RGP for undertaking such analysis. The RP should justify the calculations and engineering analysis with reference to the uncertainties in the data and in the methods employed (ONR SAP EHT.2 (Coolant inventory and flow)). However, I am content that there are no

fundamental shortfalls preventing development of the generic BWRX-300 design and associated SSSE evidence to support any future permissioning activities.

4.3.4.2. Design response to overpressure/loss of coolant

86. The BWRX-300 does not include safety relief valves in the reactor circuit design which I consider to be a novel approach. Therefore, I assessed the BWRX-300 design response to overpressure/loss of coolant scenarios.

87. I assessed chapter 6 (ref. [6]) and chapter 15.5 (ref. [33]) and identified that overpressure design considerations support 'core cooling' and 'control reactivity' fundamental safety functions. These relate to the reactor pressure vessel and containment. I identified that that overpressure protection is provided by:

- The large RPV volume (this reduces the rate of reactor pressurisation following RPV isolation) and design margins;
- Rapid reactor shutdown by inserting control rods;
- The ICS, which delivers depressurisation and emergency core cooling functions for the RPV/reactor coolant pressure boundary during design basis events and design extension conditions;
- The containment inerting system which provides overpressure protection within containment using a bursting disc for design extension conditions;
- The PCCS which reduces containment temperature and pressure in loss of coolant accidents; and,
- The ultimate pressure regulation (UPR) ICS subsystem which reduces RPV pressure during design extension conditions only.

88. I targeted the UPR system as this performs a mechanical means of pressure relief. The UPR performs a defence line 4b function and is a safety class 3 system. It is located on the steam side of the ICS. It contains a bursting disc that relieves pressure above the ICS initiation pressures and vents into the containment volume, reducing the pressure in the RPV. The RP has stated (ref. [88]) a bursting disc arrangement has been chosen (over a safety relief valve) due to its reliability and to avoid reseating faults during design extension conditions. I am content that there are no regulatory requirements as identified within table 2 of ONRs pressure system safety TAG (NS-TAST-GD-067 (ref. [62]) for the RP to include a pressure relief valve within the reactor circuit. I also judge from a mechanical engineering perspective that the RP has appropriately considered overpressure within the design and has applied proven engineering practices in the use of a bursting disc. This aligns with my expectations of ONR SAP EKP.2 (Fault tolerance).

89. I recognise that bursting discs are single-use items and typically cannot be routinely tested. To demonstrate the continuous delivery of nuclear safety functions for the bursting disc, a future SSSE should explain how qualification and maintenance activities will be undertaken. This is because ONR's licence condition (LC) 27 requires licensees to ensure that a plant is not operated, inspected, maintained or tested unless suitable and sufficient safety mechanisms, devices and circuits are properly connected and in good working order (ref. [89]). I consider LC27 expectations would be applicable to the proposed UPR bursting disk. I am content that there are no fundamental shortfalls preventing development of the generic BWRX-300 design and associated SSSE evidence to support any future permissioning activities.

4.3.5. Redundancy, diversity, and segregation

90. I sought to judge whether the RP has adequate redundancy, diversity and segregation arrangements that it has applied during its design process. I targeted the RIVs due to their novelty (two ball valve arrangements within one valve body) and importance to nuclear safety.

91. I assessed the RP's safety strategy (refs. [74], [90]). I noted that its design approach to redundancy, diversity and segregation considers:

- Common cause failure (CCF) within probabilistic safety analysis;
- Application of single failure criterion within its design rules;
- Redundant means of addressing PIEs; and
- Design principles for independence concepts.

92. This aligns with my expectations against ONR SAPs EDR.2 (Redundancy, diversity and segregation), EDR.3 (Common cause failure), and EDR.4 (Single failure criterion). For Step 2, I am content that there are no fundamental shortfalls preventing further development of the generic BWRX-300 design and associated SSSE evidence to support any future permissioning activities.

4.3.5.1. Redundancy, diversity, and segregation of reactor isolation valves

93. I targeted the application of redundancy, diversity, and segregation to the RIVs due to their novel nature and importance to nuclear safety. I sampled chapter 5 (ref. [5]) and the RIV purchase specification (ref. [91]). I identified the following:

- RIVs contain two valves within a single body;
- Each valve can independently provide isolation;
- Each valve has a hydraulic actuator with a spring-loaded return;

- An independent hydraulic system provides motive force for one of the RIV actuators in each valve pair; and,
- Actuation signals are diverse via instrumentation and control systems. Independent and segregated systems operate each valve.

94. I am content that this aligns with the expectations of ONR SAP EDR.2 (Redundancy, diversity and segregation). However, I note that failure of the valve body may challenge the redundancy and independency claims made on the RIVs. ONR's structural integrity assessment (ref. [92]) considers the material integrity of the RIVs further. I consider that the RP has not provided adequate justification for the mechanical diversity applied. I recognise that as the RIVs are a novel design, there may be a limited supply chain. So, the risk of common cause failures may increase. A future SSSE should provide justification relating to RIV mechanical diversity, specifically on how common cause failure risks have been reduced ALARP. However, I am content that there are no fundamental shortfalls preventing further development of the generic BWRX-300 design and associated SSSE evidence to support any future permissioning activities.

4.3.6. Design safety assurance

95. I sought confidence that the RP has applied appropriate design assurance arrangements to ensure safety principles are implemented. So, I sampled its:

- design process
- valve selection arrangements
- pipe design arrangements

4.3.6.1. Design process

96. I sampled the RP's Design Plan (ref. [93]), which describes its integrated design engineering process. The RP's design process is phased and includes hold points at significant stages. There are multi-disciplinary design reviews undertaken at each hold point to verify that objectives and criteria have been met, prior to progressing past the hold point. This aligns with my expectation against ONR SAP EMC.4 (Procedural control) and ONR's design safety assurance TAG (NS-TAST-GD-057 ref. [61]). I am content that there are no fundamental shortfalls preventing further development of the generic BWRX-300 design and associated SSSE evidence to support any future permissioning activities.

4.3.6.2. Valve selection arrangements

97. I targeted the RP's RIV selection process to gain confidence that it has applied appropriate design assurance arrangements. This is due to their novelty and importance to nuclear safety.

98. I assessed the BWRX-300 General Valve Selection and Purchasing Guidance (ref. [94]) provided in response to RQ-01704 (ref. [95]). This contained valve design criteria including aspects relating to the flow medium, the valve body structure and internals, and other equipment interfaces. I consider that the valve selection criteria used proven engineering practices as valves are designed in accordance with relevant codes and standards. I compared the valve selection criteria (ref. [94]) to the choice of valve for the main steam RIVs and judge them to align. For Step 2, I consider this meets the expectations of IAEA SSR 2/1 requirement 9 (Proven engineering practices) (ref. [65]) and the expectations of ONR SAP ECS.3 (Codes and standards). I am content that there are no fundamental shortfalls preventing further development of the generic BWRX-300 design and associated SSSE evidence to support any future permissioning activities.

4.3.6.3. Piping design arrangements

99. I sought to determine if the RP's piping design arrangements considered piping features and piping effects, namely water hammer, dead-legs and embedded piping, which can impact mechanical components important to nuclear safety, for example the ICS, PCCS and RIVs.

100. I sampled the RP's 'General Mechanical Design Criteria' (ref. [96]) provided in response to RQ-01758 (ref. [97]). This includes requirements to minimise the potential for water hammer and adverse environmental conditions due to piping features including dead legs. I noted that the RP identifies (from operational experience) that the main causes of water hammer in nuclear power plants relates to line voiding (entrapment of air in liquid lines). The RP proposes design measures to mitigate line voiding including minimising high points in piping systems and vent addition where the mitigation by design is not practicable.

101. I identified that the RP sets out its design approaches to mitigate flow stagnation. This includes avoiding dead legs, system low points and sudden flow changes where hot and cold fluids exist. Where mitigation by design is not practicable, the RP incorporates additional drains and flushing points to enable periodic removal of radioactive crud and fluids in the line. I am content that the RP's piping design arrangements apply proven engineering practices as piping is designed in accordance with relevant codes and standards. It has adequately considered potential faults and taken account of design issues based on operational experience. I consider this to meet the expectations of IAEA SSR 2/1 requirement 9 (Proven engineering practices) (ref. [35]) and my expectations against ONR SAPs ECS.3 (Codes and standards) and EMC.2 (Use of scientific and technical issues).

102. I assessed the RP's plant piping layout criteria (ref. [98]) to gain confidence that piping containing radioactive liquid is routed in a location with visual and physical access. This reduces the potential for undetected degradation and unmonitored release. ONR Technical Inspection Guide (TIG) relating to License Condition 34 (ref. [99]) discusses piping in concealed or

inaccessible locations and the difficulties it can present in relation to EIMT. I recognise that the RP's design approach includes routing underground piping in accessible trenches and extending piping joints away from wall penetrations. The layout criteria (ref. [98]) outlines minimum clearances between piping and adjacent equipment and obstructions to ensure adequate piping access. I judge that the RP has adequately considered access to piping for EIMT activities. This is because it has demonstrated consideration of IAEA SSR-2/1 requirement 29 (Calibration, testing, maintenance, repair, replacement, inspection and monitoring of items important to safety) (ref. [65]) in its design for safe operation over the nuclear power plants lifetime. I also consider that the RP's approach meets my expectations against ONR SAPs ELO.1 (Access) and EMC.8 (Providing for examination). I am content that there are no fundamental shortfalls preventing further development of the generic BWRX-300 design and associated SSSE evidence to support any future permissioning activities.

4.3.7. Modularisation

103. My step 2 plan (ref. [19]) included an assessment of the RP's approach to modularisation. I assessed the PSR chapters relating to mechanical engineering and did not identify any novel or fundamental modularisation safety aspects. The extent of modularisation within the BWRX-300 design is limited to a Diaphragm Plate Steel Composite (DPSC) construction technique, which is assessed within the civil engineering assessment report (ref. [100]). I have therefore not identified any fundamental shortfalls which would prevent further development of the generic BWRX-300 design to support any future permissioning activities.

4.3.8. Containment

104. My step 2 plan (ref. [19]) included an assessment of the RP's approach to containment, specifically the HVAC systems. I reviewed chapters 6 and 9A (refs. [6] [7]) and identified that safety functions relating to mechanical engineering do not play a principal role in containment nuclear safety. I also identified that control and instrumentation systems do not rely on HVAC operability. This is being considered further in the control and instrumentation assessment (ref. [101]). However, as the design matures, a future BWRX-300 safety case should demonstrate that further heat loading analysis has been undertaken to underpin this claim during future permissioning activities. I also did not note any novel aspects to the HVAC system design. I have not identified any fundamental shortfalls which would prevent further development of the generic BWRX-300 design and associated SSSE evidence to support any future permissioning activities.

4.3.9. ALARP demonstration and application of relevant good practice

105. I assessed the adequacy of the RP's approach to reducing risks ALARP. My assessment targeted the following:

- the RP’s high-level arrangements to reducing risks ALARP
- application of RGP
- claimed risk reductions compared to the ABWR
- application of optioneering

4.3.9.1. ALARP evaluation

106. To assess the RP’s high-level approach to reducing risks ALARP, I assessed its ALARP evaluation report (ref. [11]). I note that this report:

  - Identifies that risk assessment will continue to inform the design as it develops;
  - Identifies the need for further optioneering;
  - Reviews the design against RGP and OPEX;
  - Identifies that an appropriate audit trail is required to justify key design decisions; and,
  - Recognises that to demonstrate risks are reduced ALARP, the RP may need to apply additional design changes.

107. I conclude that the RP has adequately demonstrated that it understands the principals of ALARP, and that production of supporting evidence is ongoing. I have not identified any fundamental shortfalls which would prevent the RP from further developing the generic BWRX-300 design and associated SSSE evidence to support any future permissioning activities.

4.3.9.2. Application of RGP – fuel route

108. I assessed the RP’s fuel route operations (ref. [7]) targeting mechanical handling requirements as this can influence the general plant layout.

109. Both new and irradiated fuel are stored in the fuel pool and are lifted by a:

  - Polar crane – used to import new fuel (to fuel pool elevators) and export the irradiated fuel cask to reactor building; and,
  - Refuelling machine – used to move fuel within the fuel pool.

110. The spent fuel pool is on the reactor building operations floor. This allows the irradiated fuel to be transferred at heights below that proposed for the UK ABWR design. Whilst I recognise that the RP has not completed its dropped load dose consequence assessment, I consider this to be a potential improvement on the ABWR design. This is because this potentially reduces the dropped load hazard consequences.

111. During Step 1, I identified that mechanical handling equipment is designed to US standards (ASME NOG-1 (ref. [102]), NUREG-0554 (ref. [103]) and ANS 57.1 (ref. [104])). The RP completed a high-level gap analysis against RGP (ref. [105]) and concluded that the US mechanical handling standards are unlikely to be accepted for a UK design. It identified that further post GDA analysis against ONR mechanical handling RGP would be required to justify its approach and show that its mechanical handling SSCs reduce risks ALARP. I noted that the RP's initial gap analysis did not consider ONR's nuclear lifting operations TAG (NS-TAST-GD-056 ref. [60]) so I sought clarification regarding what had been considered (RQ-1703 (ref. [106])). The RP responded stating that it will use RGP to inform its post GDA gap analysis. I am content that the RP has identified appropriate UK RGP to support its future gap analysis against US mechanical handling standards used in the generic design. For Step 2, this aligns with my expectations against ONR SAP ECS.3 (Codes and standards).

112. I assessed the irradiated fuel cask removal route. I noted that the polar crane does not transport the fuel cask over the spent fuel rack and is placed in a designated area adjacent to the spent fuel pool for processing. This route ensures that fuel cask is lifted not over safety significant SSCs (other than the spent fuel pool). This is depicted in figure 3b-1 within the RP's response to RQ-02026 (ref. [107]).

113. I note that the RP has not yet specified the irradiated fuel cask and therefore has not undertaken associated drop load analysis. The RP has acknowledged that appropriate hazard analysis is required and has captured this commitment in a forward action plan (ref. [35], 15.7-62 and 15.7-69). Given the potential cask drop height, I recognise that this future analysis could result in design changes to reduce risks ALARP. However, for Step 2, I consider that the RP's approach demonstrates its spent fuel cask route has considered principles of prevention to reduce fault effects and that this will continue as the design progresses. This aligns with my expectations of ONR SAP ELO.4 (Minimisation of the effects of incidents), EKP.5 (Safety measures) and IAEA SSG-63 (ref. [108]).

114. I am content that there are no fundamental shortfalls preventing further development of the generic BWRX-300 design and associated SSSE evidence to support any future permissioning activities.

4.3.9.3. ABWR design variations assessment

115. I targeted the spent fuel pool cooling system within chapter 9A (ref. [7]). This differs from the UK ABWR design with one difference being the fuel pool cooling trains' redundancy and interconnectivity. Therefore, I sought assurance that the RP's design approach is likely to reduce risks ALARP.

116. In response to RQ-02044 (ref. [109]), the RP provided a comparison of nuclear safety requirements for the two designs. I identified that the UK ABWR requires supplemental cooling during certain core off-load scenarios

to maintain the spent fuel pool temperature. There are also systems delivering different levels of DiD that require the spent fuel pool. This spent fuel pool DiD vulnerability results in a higher safety categorisation for the UK ABWR and additional redundancy requirements. The BWRX-300 spent fuel cooling pool trains have sufficient cooling capacity and no other systems rely on the spent fuel pool to deliver their safety functions. The safety case also claims that the spent fuel pool water inventory provides adequate cooling for seven days in a loss of cooling event (ref. [7]). The design requirements for the BWRX-300 spent fuel pool result in a lower safety categorisation than the UK ABWR in line within its arrangements identified in chapter 3 (ref. [4]). I am content that the RP has applied an appropriate level of spent fuel pool cooling redundancy commensurate with the safety classification. I consider this meets the expectations of ONR SAP EDR.2 (Redundancy, diversity and segregation).

117. I note that there are no interconnections between the two BWRX-300 spent fuel cooling trains. I recognise that spent fuel pool cooling train interconnectivity is included on other reactor designs as a means of preserving cooling capability in accordance with IAEA SSG-63 (ref. [108]). I consider that the RP has not appropriately presented the arguments for omitting cooling train interconnectivity. I judge that a future SSSE should provide evidence that cross connectivity of the two trains is disproportionate in demonstrating that nuclear safety risks are reduced ALARP. However, I am content that there are no fundamental shortfalls preventing further development of the generic BWRX-300 design and associated SSSE evidence to support any future permissioning activities.

4.3.9.4. Reactor isolation valve optioneering

118. Due to the RIV novelty, I assessed the RP’s optioneering adequacy. I sought assurance that the RP had followed a methodical design process to select the RIVs. The RP provided its design selection rationale (ref. [110]) which was consistent with the requirements set out within PSR chapter 5 (ref. [5]) and the RIV specification (ref. [111]). These included:

- Layout considerations – space constraints and the mounting proximity to the RPV shell to reduce un-isolable line break potential;
- Performance requirements – pressure drop and leak tightness; and,
- Installation and maintenance – removable valve body for maintenance and a dual valve body to reduce mass.

119. The RP stated that the selection criteria led to discounting traditional gate, globe, and butterfly valves. The selected valve is a metal-seat ball valve with a hydraulic cylinder actuator in a single-forging double valve body configuration. The RP presented operational experience from non-nuclear high pressure steam applications and provided supplier engagement evidence. I am content that the RP’s optioneering process has been

undertaken early in the design process to consider nuclear safety performance requirements, EIMT requirements, RGP and supplier engagement. For Step 2, this aligns with my expectations against ONR SAP SC.4 (Safety case characteristics), ELO.1 (Access), EKP.2 (Fault tolerance) and ONR's design safety assurance TAG (NS-TAST-GD-057 ref. [61]). I have not identified any fundamental shortfalls which would prevent further development of the generic BWRX-300 design and associated SSSE evidence to support any future permissioning activities for the construction of a power station based on the design.

4.3.10. ABWR leveraging

120. I identified that the fine motion control rod drive mechanism design proposed for the BWRX-300, is similar to the UK ABWR. ONR previously assessed the UK ABWR fine motion control rod drive mechanisms and concluded that the design adequately met GDA expectations (ref. [112]). For Step 2, I have not identified any changes within the RP's submission or regulatory expectations that revises this regulatory judgement. I have applied ONR's risk informed and targeted engagement policy (ref. [14]) and have not assessed this further.

4.3.11. Categorisation of safety functions and classification of SSCs

121. Chapter 3 (ref. [4]), describes the RP's approach to categorisation of safety functions and classification of SSCs. The RP reviews the results of its safety analysis to categorise SSCs according to their nuclear safety importance. The classification reflects the importance of each SSC to plant safety and links engineering, such as codes and standards for design, manufacture, inspection, maintenance, and testing directly to the SSSE.

122. The RP's categorisation and classification approach differs from ONR guidance, NS-TAST-GD-094 (ref. [63]) as it only considers reactor faults. ONR's fault studies assessor has considered this further within their report (ref. [86]). However, it is consistent with the defence in depth principles in IAEA guidance (ref. [65]) and therefore RGP. I am satisfied with the RP's approach which, if properly applied should ensure that SSCs meet the desired safety function.

123. I assessed the RP's application of categorisation and classification to the BWRX-300 Reactor Building Polar Crane and Fuel Handling Machine. This is because I sought assurance that the potential nuclear risks had been adequately considered in the assigned nuclear safety classification.

124. I identified that the RP has applied a non-safety classification to the lifting equipment (ref. [35]). The RP states that this is due to the deterministic analysis undertaken to establish the classification of safety systems. This approach does not consider unmitigated consequences from non-reactor faults. This does not satisfy the expectations set out in ONR guidance, NS-TAST-GD-094 (Ref. [63]). The RP has identified that it should undertake

further work on consequence analysis in a forward action plan 15.7-62 and 15.7-69 (ref. [113]). I am content that the RP's delivery of these forward action plans should result in the appropriate classification of nuclear lifting equipment.

125. In my opinion, the impact of any future increase in nuclear safety classification on the fundamental crane design was unclear. I raised RQ-01702 (ref. [114]) to seek clarification. In response, the RP sought to provide assurance that the fundamental crane design will not be impacted should future safety analysis require a nuclear safety class (SC) 1 crane. The RP undertook a gap analysis (ref. [114]), between the proposed design standard in ASME NOG-1 (ref. [102]) and ONR expectations. The gap analysis (ref. [82]), compares the safety features required in ASME NOG-1 (ref. [102]) and those identified within the ONR nuclear lifting TAG (NS-TAST-GD-056 ref. [60]). It concludes that they are similar in their intent, and that the crane classification is unlikely to fundamentally impact the crane design. I am content that the gap analysis has considered appropriate codes and standards and shows that the BWRX-300 fundamental crane designs are unlikely to change to align with ONR expectations. For Step 2, this satisfies my expectations against ONR SAPs ECS.3 (Codes and standards) paragraph 170 and SAP EKP.5 (Safety measures).

126. I am satisfied that a future consequence analysis will inform the need for additional safety mechanisms or increased design margins. I expect this to form an integral part of any future ALARP justification. Whilst chapter 3 (ref. [4]) does not clearly demonstrate appropriate nuclear lifting equipment classification, I am satisfied, based on the information provided in RQ-01702 (ref. [114]), that nuclear safety classification has been considered by the RP in its BWRX-300 development. A future SSSE should justify the approach to nuclear lifting equipment design and not discount the potential for additional safety features outside of ASME NOG-1 requirements. It is my judgement that for Step 2, the omission of dropped loads consequence analysis from the SSSE, is not a fundamental challenge to the suitability of the BWRX-300 for deployment in GB.

4.3.12. Equipment qualification

127. Equipment qualification (EQ) enables the RP to demonstrate that SSCs will deliver their safety functions when required. I sought to determine if the RP's equipment qualification approach is appropriate by assessing its:

- EQ arrangements
- RIV concept qualification

4.3.12.1. Equipment qualification arrangements

128. I assessed the RP's high-level EQ arrangements. These are presented in its safety objectives and design rules (ref. [4]) and its EQ specification (ref. [115]). I noted that the arrangements consider:

- qualified life requirements
- ageing
- seismic qualification requirements
- SSC interactions
- equipment qualification standards
- harsh and mild environmental conditions
- how EQ is proportionate to an SSCs classification
- the need for a qualification programme and evidence
- configuration control arrangements

129. However, I found it unclear how the RP considered SSC mission times and appropriate SC2 and 3 SSC qualification.

130. The RP provided additional information (ref. [116]) demonstrating that SSC mission times are derived from fault analysis (ref. [80]). This informs SSC EQ and purchase specifications (ref. [115]). This highlights key SSC performance requirements including operation times. It also identifies an additional 10%-time margin on SSCs needing to operate for longer than 10 hours post-accident (mission time). For Step 2, I have not assessed the mission time adequacy. However, I am content that the RP has recognised the need for mission times within its EQ arrangements. A future SSSE should provide evidence that the mission time qualification activities and margins are justified.

131. In response to RQ-02079 (ref. [117]) the RP set out its graded approach to EQ. It stated that the equipment specification (ref. [115]) defines minimum requirements applied to all equipment. SC2 and SC3 equipment is seismic classified, environmentally and functionally qualified according to the individual component requirements contained within component data sheets. The RP's approach is shown in its response to RQ-02079 table 1 (ref. [117]). I consider this starting point could exclude SC2 SSC seismic classification, environmental and functional qualification. This may reduce the level of confidence provided. However, I consider that this approach may identify additional qualification requirements for SC3 SSCs providing severe accident mitigation. I consider that SC2 SSC EQ procedures may not deliver qualification commensurate with the safety classification. This does not meet

expectations of SAP EQU.1 (Qualification procedures). I note that ONR's previous qualification guidance (NS-TAST-GD-094 (ref. [118]) annex 2) considered SC2 mechanical qualification expectations as equivalent with SC1 SSCs. For step 2, I am content a graded approach has been adopted which may result in proportionate equipment qualification being applied to SC2 SSCs. It is my opinion that a future SSSE should provide qualification confidence commensurate with an SSCs safety classification. This may be achieved by considering the starting point for SC2 SSCs qualification as equivalent to SC1 SSCs. However, I am content that there are no fundamental shortfalls preventing further development of the generic BWRX-300 design and associated SSSE evidence to support any future permissioning activities.

4.3.12.2. Reactor isolation valve concept qualification

132. I targeted the RIV qualification due to the novel nature and importance to nuclear safety. The RP submitted its RIV qualification approach (ref. [119]). I identified that the RP would undertake qualification in line with ASME QME-1 (ref. [120]) using a DN400 (16") valve for initial concept qualification. The RP is planning to undertake additional testing beyond the requirements of ASME QME-1 (ref. [120]). I noted that this includes full flow cycling and references radiation ageing. The RP expects that this will aid in demonstrating the valve internals durability over the plant lifetime. The RP also plans to undertake functional, dynamic, and hydrodynamic testing to demonstrate the RIV will meet its design requirements. The RP expects that qualification activities will be completed post GDA. I am content that the RP has identified appropriate concept qualification. This aligns with my expectations of SAP EQU.1 (Qualification procedures).

133. I note that the RP is undertaking RIV concept qualification with a DN400 (16") variation. Whilst appropriate for Step 2 GDA, qualification evidence for each RIVs nuclear safety performance requirements, commensurate with the safety classification, would be expected to underpin the final design specification and SSSE claims. However, I am content that there are no fundamental shortfalls preventing further development of the generic BWRX-300 design and associated SSSE evidence to support future permissioning activities.

4.3.13. Examination, inspection, maintenance, testing and ageing management

134. I targeted this area as SSC reliability claims is supported through completion of appropriate EIMT activities. I assessed the RP's EIMT approach to confirm:

- suitable EIMT arrangements are in place
- safe isolation aligns with RGP

- the layout minimises EIMT risks ALARP
- ageing management is adequately addressed

4.3.13.1. Examination, inspection, maintenance and testing arrangements

135. I assessed PSR Chapter 13 Conduct of Operations (ref. [27]). I noted that it identifies future EIMT arrangements should consider:

- planning
- monitoring
- scheduling
- work execution
- preventative maintenance
- corrective maintenance

136. I also identified that the SSC EIMT frequency and type will be consider:

- supplier recommendations
- safety analysis
- periodic inspection requirements
- OPEX
- service conditions

137. I sought further evidence that the RP's EIMT arrangements appropriately consider an SSCs safety classification. The RP responded to RQ-01759 (ref. [121]) referring to the BWRX-300 reliability, availability, maintainability, and inspectability program (ref. [122]). I noted that this provides a basis for establishing SSC equipment reliability classification to assign supplier requirements commensurate with safety classification. The RP uses this to inform its SSC maintenance strategies and performance criteria. The RP discusses a graded approach to SSC design and qualification where the highest level of rigour is applied to the highest safety classification components.

138. I consider that the RP has adequately outlined its EIMT requirements basis and that this appropriately considers SSC safety classification. I am content that the proposed arrangements appropriately demonstrate how EIMT requirements and frequencies will be addressed in a future SSSE. This aligns with my expectations against IAEA SSG-74 (ref. [123]) and ONR SAPs EMT.1 (Identification of requirements) and EAD.1 (Safe working life).

139. I have not identified any fundamental shortfalls preventing further development of the generic BWRX-300 design and associated SSSE evidence to support any future permissioning activities.

4.3.13.2. Establishment of safe plant isolation configuration

140. I assessed the RP's safe plant isolation approach. I targeted this given its importance to reducing SSC EIMT personnel risks ALARP.

141. I sought assurance that the RP's arrangements aligned with safe isolation RGP (HSG253, ref. [73]). In its response to RQ-01759 (ref. [121]), the RP identified that there is no direct US equivalent to HSG253. The RP therefore undertook a gap analysis of its plant isolation arrangements against HSG253 expectations. The RP showed how its safe plant isolation arrangements align with HSG253. I noted that the RP's safe isolation arrangements (ref. [124]) provide guidance to designers to include, 'vents, drains & suitable isolation valves and no undrained piping low points'. The RP recognises further analysis against HSG253 will be undertaken as its design develops (ref. [121]). I am content that the RP's arrangements align with RGP for the safe isolation of plant and equipment. For Step 2 this meets my expectations of ONR SAP ECS.3 (Codes and Standards) and EMT.1 (Identification of requirements) and HSG 253 (ref. [73]).

142. I have not identified any fundamental shortfalls preventing further development of the generic BWRX-300 design and associated SSSE evidence to support any future permissioning activities.

4.3.13.3. Plant layout

143. I sampled the RP's plant layout arrangements to gain confidence that the plant design would not inhibit EIMT activities.

144. I assessed the BWRX-300 plant layout criteria, (ref. [98]). I identified that this provides layout-related requirements. This includes minimum access dimensions to ensure adequate space is available for operations, maintenance, inspection, and testing activities and that risks to personnel are considered and minimised. I noted that the EIMT programme (ref. [122]) specifies plant layout design considerations, which designers may apply to the 3D model to satisfy these requirements. This includes SSC space envelope provisions 'ringfenced' for plant personnel to perform activities and adequately sized laydown areas. Transit routes for large equipment are also identified. This provides assurance that SSC replacement has been considered in the design and is achievable. The RP has recognised that 50Hz equipment is often larger and heavier than the 60 Hz equipment modelled in its design currently under development for the Darlington nuclear site. To show compatibility of its layout with 50Hz rated equipment, SSCs including pumps, fans, air compressors and cable trays are modelled 10-20% larger and heavier than the 60 Hz equipment (ref. [98], appendix A).

145. In its response to RQ-02026 (ref. [107]), the RP used its 3D model to show, how RIV maintenance space envelopes have been applied. The RP stated that it uses the 3D model to develop major lifting and routing plans. I noted that this includes the steam dryer which is the largest item that may require replacing within the plant lifetime. The refuel floor truck bay hatch, which the steam dryer must pass through, is sized accordingly.

146. I consider that the RP's plant and equipment layout approach makes adequate provision for EIMT activities in different plant states. This aligns with my expectations against ONR SAP ELO.1 (Access), SC.4 (safety case characteristics), IAEA SSR-2/1 requirement 29 (ref. [65]).

147. I have not identified any fundamental shortfalls preventing further development of the generic BWRX-300 design and associated SSSE evidence to support any future permissioning activities.

4.3.13.4. Ageing and degradation management arrangements

148. I assessed the RP's ageing and degradation management arrangements to provide assurance that related risks are effectively managed. I sought to determine whether:

- Potential degradation mechanisms were considered in the design phase; and,
- Ageing management requirements traceability was evident throughout the safety case.

149. I assessed chapter 3 (ref. [4]) and chapter 13 (ref. [27]). I noted that ageing and degradation mechanisms and their effects on SSC safety, reliability, and performance are considered throughout the design process. This includes mechanical, thermal, chemical, electrical, physical, biological, and radiation aspects. Significant ageing mechanisms are also used to inform EQ activities. The RP's arrangements set out what should be included within a future ageing management programme. This includes:

- condition assessments
- obsolescence management
- SSC specific ageing management plans

150. I judge that the RP has adequately considered ageing management and identifies how it relates to a future ageing management programme and EQ. I consider the ageing management requirements to be evident throughout the SSSE. This aligns with my expectations against ONR SAPs EAD.1 (Safe working life), EQU.1 (Qualification procedures), ageing management TAG (NS-TAST-GD-109 ref. [64]) and IAEA SSG-48 (ref. [125]).

151. I assessed the RP's ageing management arrangements application to the ICS. SSSE chapter 23 (ref. [44]) considers that degradation mechanisms are

mitigated by the chemistry management and control philosophy. This is discussed further within the chemistry assessment report (ref. [126]).

152. I judge that the RP has appropriately identified ageing mechanisms and mitigation methods and incorporated RGP. This aligns with my expectations against ONR SAP EAD.2 (Safe working life) and IAEA SSG-48 (ref. [125]). I have not identified any fundamental shortfalls preventing further development of the generic BWRX-300 design and associated SSSE evidence to support any future permissioning activities.

5. Conclusions

153. This report presents the Step 2 mechanical engineering assessment for the GDA of the BWRX-300 design. The focus of my assessment in this step was towards the fundamental adequacy of the design and safety case. I have assessed the SSSE chapters and relevant supporting documentation provided by the RP to form my judgements. I targeted my assessment, in accordance with my assessment plan (ref. [19]), at the content of most relevance to mechanical engineering against the expectations of ONR's SAPs, TAGs and other guidance which ONR regards as relevant good practice.

154. Based upon my assessment, I have concluded the RP has:

- Produced an SSSE case which is logical and hierarchical aligning with RGP. A future SSSE should consider improving the traceability between the safety analysis and engineering requirements.
- Adequately demonstrated how its limits and conditions development considers RGP and is still developing.
- Adequately demonstrated that its DiD arrangements align with RGP and implemented appropriate independence in mechanical engineering SSCs in different DiD levels;
- Adequately demonstrated the ICS concept design qualification approach. A future SSSE should demonstrate adequate BWRX-300 ICS qualification;
- Adequately considered overpressure resilience in the BWRX-300 design. It has proposed use of a proven engineering technology in a novel application by including a bursting disc to relieve reactor pressure under accident conditions. A future SSSE should qualify the bursting disk and demonstrate that EIMT requirements are achievable and reduce risks ALARP;
- Adequately implemented design assurance arrangements which support valve selection including the reactor isolation valves. This includes redundancy and independence design rules. A future SSSE should consider and justify the reactor isolation valve mechanical diversity and demonstrate adequate qualification;
- Implemented some design improvements relative to previous BWR designs. A future SSSE should consider opportunities to reduce fuel route dropped load consequences, such that risk are reduced as ALARP;

- Demonstrated it is working to reduce risks ALARP. It has identified appropriate RGP, identified and mitigated risks and demonstrated an adequate optioneering approach. A future SSSE should continue to undertake analysis and produce evidence that risks have been reduced ALARP as the design matures;

- Established an appropriate methodology for its categorisation and classification principles' development. A future SSSE should include dropped load analysis to inform the nuclear lifting equipment categorisation and classification;

- Not yet identified appropriate qualification arrangements specific to SC2 components. However, a graded approach has been adopted which may result in proportionate equipment qualification being applied to safety class 2 SSCs. A future SSSE should provide evidence that equipment qualification procedures provide a level of confidence commensurate with SSCs safety classification;

- Implemented EIMT arrangements, which ensure adequate space is available to conduct planned activities. Its arrangements align with safe plant isolation RGP; and,

- Implemented adequate ageing and degradation management arrangements which identify foreseeable SSC degradation mechanisms.

155. Overall, based on my assessment, and subject to the provision and assessment of suitable and sufficient supporting evidence in either a future Step 3 GDA or during site specific activities, I have not identified any fundamental safety shortfalls that could prevent ONR permissioning the construction of a power station based on the generic BWRX-300 design.

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Appendix 1 – Relevant SAPs considered during the assessment.

SAP reference	SAP title
SC.2	Safety case outputs
SC.4	Safety case characteristics
EKP.2	Fault tolerance
EKP.3	Defence in depth
EKP.4	Safety function
ECS.3	Codes and standards
ECS.5	Use of experience, test or analysis
EQU.1	Qualification procedures
EDR.2	Redundancy diversity and segregation
EDR.3	Common cause failure
EDR.4	Single failure criterion
EMC.4	Procedural control
EMT.1	Identification of requirements
EMT.2	Frequency
EAD.1	Safe working life
ELO.1	Access
ELO.4	Minimisation of the effects of incidents
EHT.1	Design
EHT.2	Coolant inventory and flow