Office for
Nuclear Regulation

ONR Assessment Report

Generic Design Assessment of the BWRX-300 – Step 2 Assessment Report - Structural Integrity

ONR Assessment Report

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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 structural integrity 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 Britain (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 (RGP), 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 targeted the following aspects in my assessment of the BWRX-300 SSSE:

- The high-level requirements in ONR's GDA guidance related to structural integrity for components classified as highest reliability.
- The application of these requirements to the Reactor Pressure Vessel.
- Other aspects that I have targeted and sampled are included in my assessment scope section, such as passive safety systems, pressure relief in the design and double reactor isolation valves which are all novel aspects.

Based upon my assessment, I have concluded the following:

- Chapter 22 (Structural integrity) of the RP's SSSE demonstrates an approach to the structural integrity safety case which is aligned with ONR's expectations.

- Chapter 22 is well integrated into the wider SSSE case, cross referenced appropriately in other chapters providing good visibility of structural integrity requirements.

- The Reactor Pressure Vessel topic report provides a clear and comprehensive outline of the RP’s future structural integrity substantiation.

- The RP’s documentation supporting the structural integrity claims is of a high standard.

- The RP has shown an approach which is consistent with ONR's expectations regarding structural integrity. However, a future SSSE will need to demonstrate the basis for classifying all high integrity components.

- A future SSSE will need to demonstrate that appropriate material properties can be achieved during manufacture and that there is a suitable margin between the end-of-life limiting defect size and the validated inspection size.

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.

My recommendation is as follows:

- Recommendation 1: ONR should consider the outcome from my assessment as part of the decision on fundamental adequacy of the generic BWRX-300 design.

List of abbreviations

ALARP	As Low As Reasonably Practicable
ABWR	Advanced Boiling Water Reactor
BL	Baseline
BTC	Basic Technical Characteristics
BWR	Boiling Water Reactor
CAE	Claim, Argument and Evidence
CNSC	Canadian Nuclear Safety Commission
CRDM	Control Rod Drive Mechanism
DAC	Design Acceptance Confirmation
DEC	Design Extension Conditions
DESNZ	Department of Energy Security and Net Zero
DR	Design Reference
DRP	Design Reference Point
ESBWR	Economic Simplified Boiling Water Reactor
FAP	Forward Action Plan
GB	Great Britain
GDA	Generic Design Assessment
GVHA	GE Vernova Hitachi Nuclear Energy Americas LLC
GSE	Generic Site Envelope
GSR	Generic Security Report
IAEA	International Atomic Energy Agency
ICS	Isolation Condenser System
LOCA	Loss of Coolant Accident
MDSL	Master Document Submission List
NPP	Nuclear Power Plant
NRC	Nuclear Regulatory Commission
NRW	Natural Resources Wales
ONR	Office for Nuclear Regulation
OPEX	Operational Experience
PA	Protected Area(s)
PSAR	Preliminary Safety Analysis Report
PSR	Preliminary Safety Report
QEDS	Qualified Examination Defect Size
RGP	Relevant Good Practice
RI	Regulatory Issue
RIV	Reactor Isolation Valve
RO	Regulatory Observation
RP	Requesting Party
RPV	Reactor Pressure Vessel
RQ	Regulatory Query
SAP	Safety Assessment Principle(s)
SBWR	Simplified Boiling Water Reactor
SMR	Small Modular Reactor
SSC	Structure, System and Component
SSCs	Structures, Systems and Components

SSSE	Safety, Security, Safeguards and Environment Cases
TAG	Technical Assessment 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 structural integrity 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 Requesting Party's (RP) Safety, security, safeguards and environment cases (SSSE) head document (ref. [1]), specifically chapters 1, 3, 4, 5, 22, 23, 27 (refs. [2], [3], [4], [5], [6], [7] and [8]) the associated revision of the Design Reference Report (DRR) (ref. [9]) and supporting documentation.

2. Assessment was undertaken in accordance with the requirements of the ONR's Management System and follows ONR's guidance on the mechanics of assessment, NS-TAST-GD-096 (ref. [10]) and ONR's risk informed, targeted engagements (RITE) guidance (ref. [11]. The ONR Safety Assessment Principles (SAPs) (ref. [12]) together with supporting Technical Assessment Guides (TAGs) (ref. [13]), have been used as the basis for this assessment.

3. This is a major report as per ONR's guidance on production of reports NS-TAST-GD-108 (ref. [14]).

1.1. Background

4. The ONR's GDA process (ref. [15]) calls for an assessment of the 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 Safety, Security, Safeguards and Environment cases (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. In Step 1, I prepared a detailed Assessment Plan for structural integrity (ref. [16]). 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. [17]) 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], [5], [6], [7], [8], [18], [19], [20], [21], [22], [23], [24], [25], [26], [27], [28], [29], [30], [31], [32], [33], [34], [35], [36], [37], [38], [39], [40], [41], [42], [43], [44], [45], [46], [47], [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 DRR (ref. [9]) and the Master Document Submission List (MDSL) (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 (NPP) 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 (PA) as well 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. [10]) and ONR's guidance on Risk Informed, Targeted Engagements (ref. [11]).

19. My assessment has considered the requirements of ONR's GDA guidance (ref. [51]) which featured as the GDA scope submitted by the RP (ref. [49]). I have targeted highest reliability components which in my judgement are the highest significance/risk. I have also targeted aspects/components that are novel for this design (either first of a kind use or novel in comparison to the Advanced Boiling Water Reactor).

2. Assessment standards and interfaces

20. 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.

21. 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

22. The ONR Safety Assessment Principles (SAPs) (ref. [12]) 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.

23. The International Atomic Energy Agency (IAEA) safety standards (ref. [52]) and nuclear security series (ref. [53]) 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.

24. Furthermore, ONR is a member of the Western European Nuclear Regulators Association (WENRA). WENRA has developed Reference Levels (ref. [54]), which represent good practices for existing nuclear power plants, and Safety Objectives for new reactors (ref. [55]).

25. The relevant SAPs, IAEA standards and WENRA reference levels are aligned and expanded on in the TAGs (ref. [13]). The TAGs provide the principal means for assessing the structural integrity aspects in practice.

26. The key guidance is identified below and referenced where appropriate within Section 4 of this report. I also cited RGP, where applicable, within the body of this report.

2.1.1. Safety Assessment Principles (SAPs)

27. A list of the SAPs I consider relevant to this assessment is available in Appendix 1. These are the key SAPs applied within my assessment:

- To determine fundamental adequacy on the safety case I considered
    - SC.2 – Safety case process outputs
    - SC.3 – Lifecycle aspects
    - SC.4 – Safety case characteristics
    - SC.5 – Optimism, uncertainty and conservatism
- When assessing highest reliability claims for structural integrity I considered
    - EMC.1 – Highest reliability components and structures – safety case and assessment
    - EMC.2 – Highest reliability components and structures – use of scientific and technical issues
    - EMC.3 – Highest reliability components and structures – Evidence
- In addition to highest reliability claims, I also considered more generally –
    - EMC.8 – Design – providing for examination, was considered to determine adequacy of design for inspection.
    - EMC.9 – Product form, was considered in assessment of reduced safety systems as a consequence of this design.
    - EMC.13 – Materials, was considered as part of assessment of the requirements for materials in relation to structural integrity.

- The following were considered when assessing safety categorisation and classification:
    - ECS.1 – Safety categorisation
    - ECS.2 – Safety classification of structures, systems and components
    - ECS.3 – Codes and standards
- The following were considered when assessing ageing and degradation:
    - EAD.1 – Safe working life
    - EAD.2 – Lifetime margins
    - EAD.3 – Periodic measurement of material properties
    - EAD.4 – Periodic measurements of parameters
    - EAD.5 – Obsolescence

2.1.2. Technical Assessment Guides (TAGs)

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

- NS-TAST-GD-094 – Categorisation of safety functions and classification of structures, systems and components (SSCs) (ref. [56])
- NS-TAST-GD-096 – Guidance on Mechanics of Assessment (ref. [10])
- NS-TAST-GD-005 – Regulating duties to reduce risks As Low As Reasonably Practicable (ALARP) (ref. [57])
- NS-TAST-GD-051 – The purpose, scope and content of safety cases (ref. [58])
- NS-TAST-GD-016 – Integrity of Metal Structures, Systems and Components (ref. [59])

2.1.3. National and international standards and guidance

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

- IAEA SSR-2/1 – Safety of Nuclear Power Plants: Design (ref. [60])
- IAEA SSG-2 - Deterministic Safety Analysis for Nuclear Power Plants (ref. [61])

- IAEA SSG-56 – Design of the Reactor Coolant System and Associated Systems for Nuclear Power Plants (ref. [62])
- IAEA SSG-61 – Format and Content of the Safety Analysis Report for Nuclear Power Plants (ref. [63])
- IAEA TECHDOC-1936 – Applicability of design safety requirements to small modular reactor technologies intended for near term deployment (ref. [64]).
- WENRA safety reference levels for existing reactor 2020 (ref. [54])
- WENRA Safety of New NPP designs (ref. [55])
- WENRA Applicability of the Safety Objectives to SMRs (ref. [65])
- European Network for Inspection Qualification (ENIQ) methodology (ref. [66])
- American Society of Mechanical Engineers (ASME) Boiler & Pressure Vessel Code (ref. [67])

2.2. Integration with other assessment topics

30. To deliver the assessment scope described above I have worked closely with several 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.

31. The key interactions with other topic areas were:

- Fault studies - I have sought guidance on the adequacy of claims regarding tolerability of failure and consideration of how direct consequences of gross failure have influenced the classification of all SSCs, with focus on those designated as highest reliability;
- Internal hazards - I have sought guidance on the adequacy of claims regarding tolerability of failure and consideration of how indirect consequences of gross failure have influenced the classification of all SSCs, with focus on those designated as highest reliability.
- Chemistry - I have engaged with the chemistry topic to inform my judgement on how this has been considered from an ageing management perspective, in support of a robust materials selection methodology for through-life component integrity.
- Mechanical engineering and structural integrity have worked together on the assessment of some key SSCs and the topic of pressure relief. I have

led on the assessment of sub-components where safety claims are related to maintaining integrity of the primary circuit pressure boundary such as the reactor isolation valves (RIVs). A key topic where Mechanical Engineering has led is on the pressure relief aspects which is also of interest to Structural Integrity.

2.3. Use of technical support contractors

32. During Step 2 I have not engaged Technical Support Contractors (TSCs) to support my assessment of the Structural Integrity aspects of the BWRX-300 GDA.

3. Requesting Party's submission

33. 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.

34. The RP consolidated all four volumes to incorporate any commitments and clarifications identified in regulatory engagements, regulatory queries, and regulatory observations, in July 2025 . This consolidated revision is the basis of the regulatory judgements reached in Step 2.

35. This section presents a summary of the RP's safety case for structural integrity. 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

36. 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.

37. 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 Advanced Boiling Water Reactor (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 ONR assessed a UK version of the design 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 in the UK.

38. Chapter 22 of the SSSE states that the BWRX-300 design emphasises:

- Use of off-the-shelf components and proven materials.
- Design pressures and temperatures within the established BWR experience base.
- Compliance with ASME BPVC Section III for nuclear facility components.

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

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

41. Of relevance to structural integrity, there are a few notable features which are different to the ABWR. For instance, the RPV is taller than other BWR RPVs to facilitate natural circulation. The RPV is fitted with integral double RIVs on each leg of the reactor. Although described as “integral”, based on the SSSE, the RIVs will be bolted via a flanged connection which would be part of a set-in nozzle to the RPV. Additionally, the design reduces the number of penetrations to the RPV which are all located above the active fuel to prevent uncovering of the core in accident conditions. Noting these differences, the RPV design for the BWRX-300 is consistent with the ABWR design and other BWR RPV designs.

3.2. BWRX-300 Case Approach and Structure

42. The RP has submitted information on its strategy and intentions regarding the development of the SSSE (refs. [68], [69], [70] and [71]). The RP submitted these documents to ONR during Step 1.

43. The RP has submitted an SSSE for the BWRX-300 that claims to demonstrate that the BWRX-300 can be constructed, operated, and decommissioned on a generic site in 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.

44. The format and structure of the PSR largely aligns with the IAEA guidance for safety cases, SSG-61 (ref. [63]), 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 (PSAR) 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 RP’s case for structural integrity

45. The aspects covered by the BWRX-300 safety case in structural integrity is brought together in Chapter 22 (ref. [6]). The case is a “claims, arguments and evidence” structure for SSCs. The safety justification is based on Hitachi-GE (the RP for the UK ABWR GDA) structural integrity classification procedure (ref. [72]). The RP’s safety case structure claims for highest

reliability components follows a UK approach of a multi-legged structural integrity case from the Technical Advisory Group on Structural Integrity (TAGSI) (ref. [73]).

3.4. Basis of assessment: RP's documentation

46. The principal document that has formed the basis of my structural integrity assessment of the SSSE is Chapter 22 (ref. [6]). The structural integrity chapter is supported by an RPV substantiation document [74], with some links to Chapter 4 and 5 (refs. [4] and [5]).

3.5. Design Maturity

47. My assessment is based on revision 3 of the DRR (ref. [9]). The DRR presents the baseline (BL) design for GDA Step 2, outlining the physical system descriptions and requirements that form the design at the time of GDA.

48. The reactor building and the turbine building, along with the majority of the significant SSCs are housed with 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 PA boundary and PA access building.

49. The GDA Scope Report (ref. [49]) describes the RP's design process that extends from BL0: functional requirements are defined up to BL3: design is ready for construction.

50. In the March 2024 design reference , SSCs in the power block are stated to be at "BL1", which the RP defines as:

- System interfaces established;
- (included) in an integrated 3D model;
- Instrumentation and control aspects have been modelled;
- Deterministic and probabilistic analysis has been undertaken; and
- System descriptions developed for the primary systems.

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

4. ONR assessment

4.1. Assessment strategy

52. The objective of my GDA Step 2 assessment is to reach an independent regulatory judgement on the fundamental aspects of the BWRX-300 design, relevant to structural integrity 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 BWRX-300 GDA (ref. [75]).

53. GVHA is currently engaging with regulators internationally, including the NRC in the US 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.

54. 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. [76]). 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.

55. More detail on my assessment strategy can be found in my assessment plan (ref. [16]).

4.2. Assessment Scope

56. My assessment scope and the areas I have chosen to target for my assessment are set out in this section. This section also outlines the submissions that I have sampled, the standards and criteria that I will judge against and how I have interacted with the RP and other assessment topics.

57. 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.

58. In line with the objectives for Step 2, I have reviewed the highest level, fundamental claims and supporting arguments related to structural integrity. 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.

59. To fulfil the aims for the Step 2 assessment of the BWRX-300, I have assessed the following items, which I consider important:

- Safety, security and safeguards case - The RP's structural integrity safety case strategy and development of claims on SSCs important for safety. I have reviewed the RP's safety case structure, to determine whether the approach for defining and demonstrating structural integrity claims placed on SSCs important for safety are traceable and clearly linked to the safety functional requirements of the plant (requirement [2.11 b] in ref. [15]);

- Design basis and parameters - How structural integrity has been considered in the design to reduce risk ALARP (requirement [2.4] of ref. [15]). I have assessed the RP's approach for the design and classification of SSCs and associated support structures for operating in all normal and fault conditions to ensure that risks are ALARP. This includes:

    - Review of the RP's approach for the structural integrity classification of SSCs, along with examples of how it will be applied for a representative range of SSCs, including the identification of those requiring a highest reliability claim (requirement [2.6] of ref [15]);

    - The basis for applying avoidance of fracture demonstrations in support of any highest reliability component claims; and

    - An outline of the selection, application and compliance with relevant codes and standards is demonstrated, as well as how these may be supplemented for meeting UK expectations (requirement [2.11 c] of ref. [15]).

- Reactor coolant circulation – The choice of design of coolant pipework and systems is important to structural integrity, to minimise components of highest reliability but also to minimise degradation mechanisms. I have reviewed the RP's approach to ensuring structural integrity claims are ALARP in these design decisions;

- RPV – There are several unique aspects of the RPV identified in my assessment plan that I focus on in my assessment; and

- Challenges of BWR technology – There are several unique aspects in BWR technology and specific to the BWRX-300 that are of interest to structural integrity, and these are noted in my assessment plan. The items most dissimilar to the ABWR and of highest safety significance receive most of my focus.

60. I have leveraged work performed to assess the UK ABWR GDA, looking at similarities/differences in approach or in novelty of design. For example:

- I consider the RPV to be similar in design and will judge whether the process/approach to SI substantiation are also similar;

- The chimney section is novel and supports natural circulation, therefore I have targeted this component;
- The double RIVs are first of a kind, so I have targeted these;
- The containment is similar but I have reviewed the RPs approach as this is a fundamental component and lies on the boundaries of civil engineering and structural integrity topic areas; and
- I have targeted material selection as an important matter to assess.

61. I have not targeted:

- Control Rod Drive Mechanisms (CRDMs) as they are similar in design to ABWR; and
- ASME code compliance as compliance with the code is recognised international RGP.

4.3. Assessment

4.3.1. Safety Case

62. I have sampled the safety case structure and content, in particular PSR Chapter 22 - Structural Integrity of Metallic System Structures and Components (ref. [6]) which states the high-level structural integrity claims and arguments. This is consistent with ONR's expectations for the second step of GDA (ref. [51]). The overall safety case structure follows the IAEA methodology, SSG-61 (ref. [63]) noting that Structural Integrity is not a chapter within this guidance but the RP (and others previously, most notably, Hitachi-GE) have appended this chapter for completeness which I judge to be good practice.

63. The classification of SSCs is determined from the fault and consequence analysis, which aligns with ECS.1 and TAG 94 (ref. [56]). SSCs for which the consequences of gross failure could lead to a large radioactive release are designated as requiring the highest level of structural integrity requirements. These have been identified as High Integrity components with additional requirements above Safety Class 1. The RP's approach to classification of highest reliability components meets ONR's expectations under SAPs EMC 1 to EMC 3.

64. In my opinion, effective communication and interfaces between structural integrity and other topics is crucial. An example of the importance of this is fault analysis/hazard analysis. Many other topics require such links. Chapter 22 links to other topics e.g. Chemistry and to other chapters within the safety case which in my opinion shows consideration of this important matter of traceability of the safety case.

65. The content of (ref. [6]) aligns with my expectations for a PSR as set out by ONR SAPs SC 2, 3, 4 and 5,EMC SAPs 1-3 and IAEA SSG-61. I consider there are no fundamental shortalls for GDA Step 2. In my opinion, Chapter 22 (ref. [6]) and its supporting references provides a good framework for a future SSSE which supports the deployment of the BWRX-300 in GB.

4.3.2. Design basis and parameters

4.3.2.1. Categorisation and classification

66. Fundamental to design and reducing risks ALARP is a robust categorisation and classification process. The BWRX-300 is the 10th iteration of the BWR design GVHA and in my opinion embeds operating experience (OPEX) and the principles of reducing risk by learning from experience and improving the design. This has been made clear in the general description document submitted in step 1 (ref. [77]).

67. The BWRX-300 approach to classification of nuclear safety systems and components is based on principles of IAEA SSR-2/1 (ref. [60]) and IAEA SSG 30 (ref. [78]) as described in the BWRX-300 safety classification (ref. [79]). I have only considered this at the highest level to understand the general principles as well as incorporation and derivation of structural integrity requirements. The overall categorisation and classification approach has been considered by the fault studies inspector (Ref. [80]).

68. The RP has referenced the Hitachi-GE approach for determining structural integrity requirements based on SSC categorisation and classification used for ABWR (ref. [6]). ONR previously deemed this approach to be acceptable in our assessment of the UK ABWR (ref. [81]). The RP has suggested some updates to its methodologies to adapt the ABWR approach to meet the BWRX-300 design, for instance, to account for the yet to be developed break exclusion zone methodology (see section 4.3.6.3) and potentially removing the intermediate classification present in the ABWR methodology. It is important to note that within Step 2, no implementation of the structural integrity categorisation and classification process has been undertaken. A future safety case should include an update to these approaches to ensure alignment, prior to implementation.

69. I judge the BWRX-300 general classification and structural integrity classification approach incorporates the requirements from SSR-2/1 (ref. [60]), SSG 30 (ref. [78]) and SSG-56 (ref. [62]), which itself aids meeting SSR-2/1 (ref. [60]). I judge that the safety classification is used to inform the engineering design, with manufacturing rules applicable to each individual component selected with due account taken of the consequences of failure, in terms of both the fulfilment of the safety function and the prevention of a radioactive release (ref. [62]). The RP’s approach to classification is consistent with my expectations against SAPs ECS.1, ECS.2 and ECS.3. The RP’s approach is also consistent with the structural integrity aspects of Issue G of the WENRA reference safety levels and safety objectives for new

plants (ref. [55]) on safety classification of SSCs: ‘SSCs important to safety shall be designed, constructed and maintained such that their quality and reliability is commensurate with their classification’.

70. The RP has proposed a structural integrity classification approach that is deterministic and considers secondary consequences from other components. If any of these failures could lead to significant radiological consequences, they will be designated as High Integrity, which is the RP terminology for highest reliability components. These will include additional structural integrity requirements over and above recognised international codes as stated in SAPs EMC 1-3.

71. Whilst at this time, a detailed consequences analysis is not complete and the structural integrity categorisation and classification process has not started, it is recognised that the RPV is likely to be designated High Integrity and to understand how this might be substantiated (structure and content), the RP has produced a topic report on the RPV (ref. [74]), which outlines the approach at a component level.

72. This approach is similar to that used by other RPs previously, which is recognised to meet ONR’s expectations. Whilst I have not assessed the generic BWRX-300 safety categorisation and classification approach (ref. [79]), it assigns safety classes 1-3 to systems and components. A future SSSE would need to demonstrate the associated ASME structural integrity requirements. Further consequence analysis may require changes to the categorisation and classification of SSCs but the approach and linkage from classification to structural integrity requirements is appropriate, which in my opinion meets ECS.3 and TAG 94 (ref. [56]).

73. My discussions with the RP suggested that the RIVs could be classed as High Integrity (ref. [82]), though further fault analysis may support an alternative classification. I focused my review on the RPV as a more likely High Integrity component requiring enhanced requirements. A future safety case should examine all potentially High Integrity components to ensure ALARP risk reduction and proportionate protective measures.

74. The RP does not make any High Integrity claim within the PSR document. I consulted with the fault studies inspector, and he advised that RQ-01763 response (which describes the future consequence analysis) is applicable to 10^{-5} per reactor year events implying a High Integrity claim in the future (Ref. [80]). On this basis, I have concluded that the RP is intending to make High Integrity claims on this plant item and potentially others in the future. This is important because if this item is designated in a future SSSE as High Integrity, at this point in GDA there needs to be confidence it can meet all the expected structural integrity requirements. See paragraph 90 for my conclusion on this matter.

75. In my opinion, the BWRX-300 design philosophy is focused on reducing the number of critical components and the severity of potential faults. If

subsequent analysis aligns with the design intent, it could lead to a low number of components designated ‘High Integrity’. This approach would align with the guidance in our SAPs (paragraphs 286-288). I judge the RPs described approach to structural integrity categorisation and classification meets my expectations against ECS 1-3 and TAG 94 (ref. [56]).

4.3.2.2. Highest reliability demonstration

76. Components designated as highest reliability components require a robust structural integrity safety case, through a multi-legged approach (EMC.1-3). The RP has chosen to follow the TAGSI structure, which has been long established within the UK and recognised as RGP, as identified in TAG 016 (ref. [59]). The RP recognises that code compliance, which it states will be to the ASME design code, is only a starting point when it comes to components that require enhanced evidence. I judge the use of ASME to be RGP as a design code, noting Sizewell B was built and is maintained to this design code.

77. There are five sub-claims in ref. [6], these are:

- Structural Integrity is assured by good design and considering relevant BWRX-300 OPEX (sub-claim 2.1.3);
- Structural integrity is assured by material selection and quality manufacturing (sub-claims 2.1.4 and 2.1.5);
- Functional testing provides a demonstration of integrity at start of life (sub-claim 2.1);
- Through-life integrity is demonstrated by analysis and inspection (sub-claims 2.1.3 and 2.1.5); and
- Inspection and monitoring regularly validate integrity through-life (sub-claim 2.1.5).

78. Each of these claims has a series of arguments but at this stage the RP has not provided any supporting evidence. This is acceptable within the scope of a step 2 GDA (Section 3.20 of ref. [51]).

79. All these legs will contribute to an avoidance of fracture demonstration. I have reviewed the overall approach to avoidance of fracture, which is based on the UK ABWR approach, utilising recognised R6 defect tolerance assessment, to establish a Qualified Examination Defect Size (QEDS). From this QEDS, the proposal is to use qualified inspections to detect defects at this size or larger using the ENIQ approach (ref. [66]). In addition, a material selection process, high quality manufacture and material testing is recognised as important to ensure the highest level of confidence in component quality. I judge that the overall approach is consistent with RGP and based on the approach previously submitted for the UK ABWR GDA. It

meets my expectations with regards to highest reliability demonstration, SAPs EMC 1-3.

4.3.2.3. Code compliance

80. As discussed above, the RP has chosen the ASME design code for the BWRX-300 for pressure vessels and piping design, which it has experience with, and is internationally recognised. Although design code compliance is important and if used correctly will lead to high quality components, I have not focused my assessment on this area as the RP is experienced in design and operating plant in compliance with this code and it has been used for decades at Sizewell B. Instead, I have focused on beyond code compliance, on those components which are likely to be categorised as High Integrity, notably the RPV, as these components carry the most significant consequences if they were to fail.

81. In my opinion, the use of ASME aligns with IAEA SSR 2/1 (ref. [60]) and SSG-56 (ref. [62]) for use of proven practices and established codes and standards. ASME is recognised as established RGP in the UK for reactor designs (Sizewell B and other GDAs).

4.3.3. The reactor pressure vessel

82. The RP produced a topic report (ref. [74]) to outline the substantiation at a component level for the RPV. Currently this is limited to claims and arguments. The design of the RPV is similar to that of the ABWR but there are notable differences I recognised in my assessment plan (ref. [16]). My assessment plan suggested these differences should be the focus of my assessment, however, there are no additional structural integrity requirements available for some components of the RPV above their current classification, such as the chimney. Instead, I discuss these at a high level and judged whether there are any potential fundamental shortfalls as a consequence of these components being used in the design from a structural integrity perspective.

83. The RPV has three fundamental safety functions, as follows:

- control of reactivity
- removal of heat from the fuel
- confinement of radioactive materials

84. I focused my assessment on the confinement function which is most applicable to structural integrity but have also considered the removal of heat claimed through natural circulation.

85. Similar to (ref. [6]), the topic report has sub-claims, as follows:

- 2.1.2 – the design of the system/structure has been substantiated to achieve the safety functions in all relevant operating modes;
- 2.1.3 – the system/structure design has been undertaken in accordance with relevant design codes and standards (RGP) and design safety principles and taking account of Operating Experience to support reducing risks ALARP;
- 2.1.4 – system/structure performance will be validated by suitable testing throughout manufacturing, construction, and commissioning;
- 2.1.5 – ageing and degradation mechanisms will be identified and assessed in the design. Suitable examination, inspection, maintenance, and testing will be specified to maintain systems/structures fit-for-purpose through-life; and
- 2.1.6 – the BWRX-300 will be designed so that it can be decommissioned safely, using current available technologies, and with minimal impact on the environment and people.

86. I judge these are key areas of structural integrity substantiation for a component such as the RPV.

87. The chimney is a large diameter cylindrical component typically found in BWRs to direct coolant flow inside the RPV. It is a key component of the BWRX-300 design for natural circulation and passive cooling. It is not a novel aspect in terms of the component design, but it is novel in terms of its function of providing natural circulation. The chimney is an extension of the internal core shroud present on most reactor pressure vessels (BWRs and Pressurised Water Reactors) which provides directional flow within the vessel to the incoming cooler water. This function is applicable for the BWRX-300 chimney/shroud. In addition, the significant length of this chimney helps provide a pressure differential between the water in the channel between the chimney and the RPV wall and the water/vapour inside the chimney above the fuel, which in turn enables natural circulation. By doing so, the overall design has eliminated the need for large coolant pumps.

88. Currently the RP does not propose any specific structural integrity claim for the chimney. At this stage of the design, it is unlikely the chimney would require any specific classification above the typical reactor internal components. Of note, the RPV wall and the chimney will have a penetration for the ICS return. A future safety case should consider this feature and ensure all the relevant degradation mechanisms, such as fatigue and thermal fatigue, and that ALARP considerations are included in the design process. I did not sample this aspect of the design any further in step 2 as the purpose of my assessment is limited to the general aspects of the design.

89. During step 1, the RP confirmed that the RIVs would not be integrated to the RPV by either means of welding or forging. Instead, the RIVs will be bolted via a flange that is part of a set-in welded nozzle to the RPV (Figure 3-3 from ref. [77]). Whilst the double valves in one body and positioned close to the RPV is a novel design, the RP is uncertain at this time whether they will be classified as High Integrity until a complete hazard analysis is carried out. If these were to be classified as High Integrity, this could place significant requirements on the integrity of the valve bodies (of relevance to structural integrity as a pressure boundary component).

90. Whilst RIVs may be common in the process industries, these designs are not common to nuclear and substantiation to a high level of reliability will be novel. However, in my opinion, the design and assessment process should be comparable to other components such as the main steam isolation valves or reactor coolant pump bowls at Sizewell B. Previous GDAs have referred to other difficulties with valve bodies, particularly those that are cast. The RP has suggested these will be forged valve bodies. At this stage, I consider the RP's approach to design and substantiation of the RIVs to be adequate, particularly if they do not require classification of High Integrity. Should this change, I would expect a future safety case to demonstrate a robust level of substantiation as required for this level of reliability. The RP's approach to classification presented within step 2 is consistent with my expectations as per ECS.1.

91. One matter for consideration as a consequence of the RIV design is the loading in normal and accidental conditions due to placement of these valves onto the RPV set-in nozzles. I expect a future safety case should demonstrate appropriate assessment of these conditions. I do note the RP's proactive approach with regards to the placement and design of the RIVs. Usually, designs have an inboard and an outboard containment isolation valve. The BWRX-300 design consists of an outboard containment isolation valve as standard and an inboard isolation valve (for most lines except the ICS), but the system also incorporates two RIVs for redundancy.

92. While there is a possibility of common cause failure, the BWRX-300 design is proposing an additional isolation valve to most designs of reactor technology. Also, by placing it close to the reactor, if adequately conceived, the design reduces the likelihood of a main steam loss of coolant accident and potentially reduces the categorisation/classification requirements for downstream SSCs. These design measures are, in my opinion, a good demonstration of the RP's considerations to reduce risks to ALARP.

93. In discussion, ONR's fault studies inspector explained that the design places a high reliance/reliability on the function of these valves. This is in part due to reduced safety injection systems (reliance on passive safety). Whilst the BWRX-300 design may incorporate an additional valve and may make the design more resilient to main steam line breaks, the design basis of the reactor still relies on the RIVs to operate as intended. The design basis

analysis will therefore be an important consideration for the future safety case.

94. I judge the RP's approach to design is consistent with ONR's technical advice to inspectors in TAG 16 (ref. [59]) in that highest reliability claims should be avoided by a robust plant design where possible.

95. Another key feature of the design of the RPV is a reduction in the number of active safety injection systems and therefore the number of penetrations in the RPV. In addition, where possible, penetrations have been moved to above the level of the top of the fuel to support accident situations that may lead to fuel being uncovered. In terms of supporting the containment function of the RPV, the reduction of the number of penetrations is a positive measure, particularly from a loss of coolant accident perspective but also for the integrity of the vessel, to limit the number of the length of welds on the RPV (and other safety systems), which I judge, meets EMC.9.

96. I have not targeted the CRDM penetrations in the bottom head of the RPV as part of my assessment as the BWRX-300 design is similar to the ABWR assessed in (ref. [81]). ONR's specialist assessment of the ABWR did not raise any significant findings for the design.

97. Reactor coolant chemistry is subject to assessment by the Chemistry specialists and is relevant to structural integrity through the environment to metal interactions. A good material selection process is key to ensuring risks from degradation are reduced to ALARP. I have reviewed the Reactor Chapter 5 (ref. [5]) and the Chemistry Chapter 23 (ref. [7]). There is good alignment between these two chapters in my opinion. The additional requirements for the RPV in Chapter 22 are also consistent with the RPV topic report (ref. [74]).

98. The RPV topic report sub-claims include:

- "Structural Integrity is assured by material selection and quality manufacturing"; and
- Argument 5 (sub claim 2.1.4) "Components are manufactured through judicious material selection".

99. This argument aligns with my expectations as set out in EMC 3 and 13 and the requirement in GD-007 (ref. [51])– "material selection and specification with tighter control of composition") which states:

- "Supplementary material specification exceeding ASME BPVC requirements, applies to RPV pressure boundary materials."

100. The RP has not submitted substantial evidence to support a robust material selection process within this stage of GDA. However, I judge that the evidence provided within the submissions at step 2 is sufficient to ensure high quality material through a robust material selection process.

101. Chapter 5 (ref. [5]) explicitly discusses the significant threats to the components, such as stress corrosion cracking and general corrosion under oxidation and provides information on mitigations through control of water quality from chemical additions and chemical processes, which in some cases serve multiple purposes. In addition to controlling the chemistry of the water to mitigate crack initiation and other corrosion mechanisms to determine the material selection, irradiation embrittlement is another key degradation mechanism of the RPV. Table 5-3 in Chapter 5 (Ref. [5]) provides confidence that the RP has considered irradiation embrittlement effects on the RPV as it notes:

- "There are special controls of key elements and interstitials to address different embrittlement processes including radiation embrittlement and reduction of toughness in the RPV."

102. Chapter 5.2.2 on material properties also highlights the consequences of irradiation and the rationale for obtaining high quality material properties in manufacture.

103. The process is consistent with previous BWR designs. I judge that material selection and properties are appropriately considered in the RP's case. In consideration of these points, there are no significant shortfalls with the RP's approach to materials selection, to provide confidence that the design uses proven materials. In my opinion, this approach aligns with Requirement 47 of the IAEA SSR2/1 (ref. [60]) for quality of materials and SSG-56 for material properties and characteristics as a specific consideration in the design of the reactor coolant system pressure retaining boundary and associated systems (ref. [62]). In addition, the RP's safety justification meets my expectations for SAPs EAD 1-5 and EMC. 3. In reaching my conclusion, I also accounted for the RP's response to RQ-01760 (Ref. [83]) on ageing management. The RP identified criteria for the BWRX-300 design such as degradation mechanisms, reliability, chemical control, and material selection which meet ONR (EAD 1-5) and international guidelines.

104. In Chapter 5 (ref. [5]), there is a specific section 5.11 on access for inspection and maintenance. This aspect must be considered in design and is a focus area within the structural integrity guidance (ref. [51]), as reflected in SAP EMC 8. This also aligns with requirement 32 of IAEA SSR 2/1 (ref. [60]) and SSG-56 (ref. [62]). It is positive to see access for design considerations being explicitly addressed in the SSSE.

105. I targeted this aspect further by asking the RP for any information related to inspection of the RPV bottom head welds performed from the outside. The RP provided documentation (ref. [84]) which outlined a 3D model set up to evaluate access for inspection configurations around plant and to identify any potential restrictions for inspections. The model provides a useful visual representation to identify any potential difficulties and the necessity in early provision for access. Additionally, a civil engineering meeting presented requirements for positioning SSCs to ensure adequate spacing for

inspection and maintenance (ref. [85], [86]). I consider that the RP has applied due diligence and that the RP's approach with regards to design for inspectability is aligned with international and ONR guidance. I judge the RP's approach with regards to inspection access and design for inspection is consistent with my expectations (EMC.8).

106. Overall, I conclude that, from a structural integrity perspective, the RPV design and the novel features of the BWRX-300 design are fundamentally acceptable and consistent with ONR's SAPs.

4.3.4. Passive cooling and pressure relief systems

107. The BWRX-300 design uses the ICS as a primary means to cool and depressurise the reactor circuit. Using this system, the RP has chosen to remove reliance on safety relief valves, claiming that these components can be unreliable and lead to loss of coolant accidents. The safety relief aspect has been assessed by the Mechanical Engineering specialist (ref. [87]).

108. This design concept was introduced in the ESBWR, which was assessed and approved by the US NRC (ref. [88]). This was achieved through a full scale mock up demonstration of this equipment. In my opinion, this provides suitable evidence to support its substantiation.

109. Currently, there are no specific structural integrity claims on the ICS. The ICS consists of stainless-steel pipework between the RPV and the submerged heat exchanger. The system includes return valves, connecting systems such as the boron injection system and shut down cooling and a beyond design basis safety relief system, named the ultimate pressure regulation subsystem. This subsystem includes provision of a rupture disc and a remote actuated isolation valve. Currently, the RP classes the system as 'class 1' based on its nuclear safety function. The approach is consistent with my expectations (ECS.1, ECS.2).

110. An issue which I have given careful consideration is the return condensate lines which remain isolated from the reactor unless demanded, usually on shutdown/trip. Therefore, this return line may spend a considerable time with little or no circulation and may be subject to a faster rate of degradation than anticipated if impurities concentrate on areas of little or no circulation. I have not seen evidence of this being accounted for in the RP's design justification. Whilst it is not a significant matter for step 2, a future safety case should consider this aspect, noting a damage assessment is standard practice for class 1 components under the ASME code (Ref. [67]).

111. The mechanical engineering specialist raised RQ-01758 (Ref. [89]) on the RP's approach to address uncertainty in the pipework due to the novelty of the design, in particular, how the RP accounts for appropriate operational experience. This aligns with structural integrity expectations for ageing management of pipework and so I reviewed the output from this perspective (EAD.1 and EAD.2). The RP's design principles consider water hammer,

dead-leg phenomenon and inaccessible or buried systems. The RP's response provides confidence in the design, manufacture and maintenance of essential connections between the pipework, the vessels and other SSCs. The RP's response also demonstrates how OPEX was considered within the design process.

112. Both the ICS and Passive Containment Cooling System (PCCS) are passive in design. The only active components on the ICS system are the isolation valves which are designed to fail in a safe position. For example, the isolation valves on the return condensate line from the ICS are energised when closed. Loss of power opens the valves. The PCCS is designed to ASME Section III Class 2, NCA and NCD, and currently, similar to the ICS, has no specific structural integrity claims. It is a heat exchanger of steel pipework on the containment walls, which routes through the equipment pool to reject heat through convection. It would operate under certain conditions, such as a significant steam leak in containment, and it would do so without operator action being required. I judge that compliance with the relevant sections of the ASME code would be sufficient for the design of this system at this current level of classification. As this system is within containment, a future safety case should consider any interactions with other safety systems during consequence analysis which may impact the classification of this system (or others).

113. ONR considers the ASME design code as international RGP (ref. [59]). The RP has provided sufficient evidence that these systems will be designed and operated to the ASME standard if they retain their current classification level. The RP's safety submission therefore meets my expectations with regards to codes and standards (ECS.3).

4.3.5. Containment vessel

114. The containment vessel is proposed as a composite steel/concrete structure that is modular in form and will be welded together to install.

115. As the vessel is not proposed as a free-standing pressure vessel, it will not be designed/constructed to the ASME BPVC and is therefore not a focus for my assessment. I attended a civil engineering meeting which gave detailed information on the design and intent of the containment (and other aspects). The design will incorporate significant welding of these modules, many of which will be performed on site during installation.

116. At final construction, the containment facing element will be a steel pressure vessel. From this perspective, despite it not strictly being a free-standing pressure vessel, the design intent is similar and therefore whilst at this time I do not challenge the construction methods or build code, a future safety case should demonstrate appropriate consideration of structural integrity requirements.

117. Another feature of note is that the containment will employ a dry inert atmosphere, however, this is not novel for the previous designs and is similar to the ABWR (also dry and inert – Ref. [77]). I am therefore not considering these two aspects any further in this assessment and from a structural integrity perspective, the containment vessel meets my expectations.

4.3.6. Break exclusion zone (BEZ)

4.3.6.1. Background

118. As part of an international collaboration with US NRC and CNSC, ONR has considered the application of BEZ in parallel to the BWRX-300 GDA step 2.

4.3.6.2. Step 2 assessment of BEZ methodology

119. I introduced the RP BEZ proposal in section 4.3.2.1 of my assessment report. A brief explanation of the methodology is provided at the start of the next section. The RP's approach to the structural integrity assessment does not include BEZ methodology for highest reliability claims. I judge the approach in step 2 for the demonstration of highest reliability claims remains consistent with my expectations (EMC 1-3).

4.3.6.3. International collaboration on BEZ methodology

120. BEZ methodology is applied under US NRC guidance, which allows the use of pre-approved analysis (by US NRC), to eliminate dynamic effects of pipe ruptures from the design basis and then specifically, that breaks and cracks in areas designated as BEZ, do not need to be postulated.

121. There have been several meetings held between ONR, US NRC and the CNSC to understand the different perspectives on the BEZ methodology. US NRC explained potential updates to the methodology, which includes greater consideration of gross failure and locations of failure, a key requirement in hazard analysis for structural integrity in the UK.

122. The structural integrity topic within ONR has extant views on the use of BEZ methodology from previous similar applications. CNSC have raised questions to GVHA during their vendor design review of the BWRX-300 related to the methodology.

123. ONR expressed views on the use of BEZ at this forum and explained ONR's requirements for detailed hazard analysis and subsequent structural integrity requirements under the SAPs for highest reliability components, for which no lines of protection are available against the consequences of gross failure. At a meeting in November 2024 (ref. [90]), ONR, US NRC and CNSC concluded that further work would be required to harmonise the approach with regards to BEZ.

124. It became apparent from further discussions between ONR, the RP, US NRC and CNSC that BEZ was an area which could benefit from increased

collaboration. I raised RQ-01921 to understand the direction of the proposed UK BEZ methodology noting that submissions to CNSC had incorporated some changes to the standard US BEZ methodology. I have considered this information to aid collaboration rather than inform my judgement on adequacy for step 2, which is stated at the end of this section, however, any information that may affect it (positive or negative) will be reflected in the following paragraphs.

125. The response to the RQ (Ref. [91]) included reference to documents previously shared with ONR, which are correspondence between Ontario Power Generation and CNSC. GVHA acts as technical support for Ontario Power Generation but has explained it does not directly influence decisions made by them. Noting this, Ontario Power Generation has suggested some modifications to the approach currently within the submission for Tennessee Valley Authority, who have submitted their design to US NRC. These modifications, include approaches that may be greater aligned with UK expectations, such as defect tolerance assessment. This gives some confidence to the UK GDA, that such modifications may be implemented (noting uncertainty around the current position).

126. The RQ included questions around whether a position for the UK can be established based on the current positions for the other two applications. The RP could not confirm this at this stage. The RP did provide notes from their Gap Analysis Review Panel, which showed that the RP does understand the gap between the US BEZ methodology and requirements for highest reliability components in the UK. This again, provided some confidence related to the UK GDA.

127. In relation to international collaboration, based on the response from the RP and engagements internationally, there remains uncertainty regarding the use of BEZ within the UK context. Discussions between ONR, CNSC and US NRC will however continue. These conclusions have not altered my judgement in paragraph 119.

5. Conclusions

128. Based upon my assessment, I have concluded the following:

- Chapter 22 (Structural integrity) of the RP's SSSE demonstrates an approach to the structural integrity safety case which is aligned with my expectations.
- Chapter 22 is well integrated into the wider SSSE case, cross referenced appropriately in other chapters providing good visibility of structural integrity requirements.
- The RPV topic report provides a clear and comprehensive outline of the RP's future structural integrity substantiation.
- The RP's documentation supporting the structural integrity claims is of a high standard.
- The RP has shown an approach which is consistent with my expectations regarding structural integrity. However, a future safety case will need to demonstrate the basis for classifying all High Integrity components.
- A future SSSE will need to demonstrate that appropriate material properties can be achieved during manufacture and that there is a suitable margin between the end-of-life limiting defect size and the validated inspection size.

129. 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.

6. References

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[21] GE-Hitachi, NEDO-34170 BWRX-300 UK GDA Chapter 8 - Electrical Power, Revision C, 15 July 2025, ONRW-2019369590-22501.

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Appendix 1 – Relevant SAPs considered during the assessment.

SAP reference	SAP title
SC.2	Safety case process outputs
SC.3	Lifecycle aspects
SC.4	Safety case characteristics
SC.5	Optimism, uncertainty and conservatism
EMC.1	Integrity of metal components and structures: Highest reliability components and structures – Safety case and assessment
EMC.2	Integrity of metal components and structures: Highest reliability components and structures – Use of scientific and technical issues
EMC.3	Integrity of metal components and structures: Highest reliability components and structures – Evidence
EMC.8	Integrity of metal components and structures: Design – Providing for examination
EMC.9	Integrity of metal components and structures: Product form
EMC.13	Integrity of metal components and structures: Materials
ECS.1	Safety classification and standards: Safety categorisation
ECS.2	Safety classification and standards: Safety classification of structures, systems and components
ECS.3	Safety classification and standards: Codes and standards
EAD.1	Ageing and degradation: Safe working life
EAD.2	Ageing and degradation: Lifetime margins
EAD.3	Ageing and degradation: Periodic measurement of material properties
EAD.4	Ageing and degradation: Periodic measurement of parameters
EAD.5	Ageing and degradation: Obsolescence