ONR Assessment Report

Generic Design Assessment of the BWRX-300 – Step 2 Assessment of Internal Hazards

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Project Name: Generic Design Assessment of the BWRX-300 – Step 2

Report Title: Generic Design Assessment of the BWRX-300 – Step 2 Assessment of Internal Hazards

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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 internal hazards 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, on 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 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 sampled key hazards, fundamental layout considerations and novel aspects in my assessment of the BWRX-300 SSSE.

Based upon my assessment, I have concluded the following:

- Key hazards were identified at a high level, supported by methodologies fulfilling a GDA Step 2 fundamental assessment.
- Although, hazard identification and analysis were not exhaustive at this stage, fundamental principles have been included early in design development. The fundamental layout considerations adopt Engineering Key Principles, adequately considering segregation and separation. At GDA Step 2, this provides assurance of the design's intent to mitigate internal hazards risks. Further work in future will be required to align internal hazards analysis and the overall case to demonstrate the risks to the plant, operators and public will be reduced to so far as is reasonably practicable (SFAIRP).

- My sampling of novel aspects in safety systems and reactor isolation valves did not identify any significant safety shortfalls in the protection against internal hazards.

- The rationale for the design and the basis for any requirements was clearly set out, demonstrating the RP’s use of operational experience and international standards to safety risk reduction. By continuing with this approach, addressing Forward Action Plan (FAP) items, assessing the impact of future design changes/technological advances and incorporating the Relevant Good Practice (RGP) at the time, organisations taking this project forward should have a future ability to demonstrate the design will reduce risks to SFAIRP.

Overall, based on my assessment to date, I have not identified any fundamental safety shortfalls that would 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 to substantiate the claims and proposals made in the GDA Step 2 submissions.

List of abbreviations

AC	Alternating Current
ALARP	As Low As Reasonably Practicable
ABWR	Advanced Boiling Water Reactor
AISC	American Institute of Steel Construction
ASME	American Society of Mechanical Engineers
ANS	American Nuclear Society
ANSI	American National Standards Institute
AP	Assessment Plan
BEZ	Break Exclusion Zone
BL	Baseline
BLEVE	Boiling Liquid Expanding Vapour Explosion
BPVC	Boiler and Pressure Vessel Code
BWR	Boiling Water Reactor
CNS	Canadian Nuclear Society
CNSC	Canadian Nuclear Safety Commission
DAC	Design Acceptance Confirmation
DEGB	Double Ended Guillotine Break
DiD	Defence in Depth
DL	Defence Line
DRR	Design Reference Report
DSA	Deterministic Safety Analysis
ESBWR	Economic Simplified Boiling Water Reactor
FAP	Forward Action Plan
FHA	Fire Hazard Assessment
FMCRD	Fine Motion Control Rod Drive
FSF	Fundamental Safety Function
FSP	Fundamental Safety Property
FSSA	Fire Safe Shutdown Analysis
GB	Great Britain
GDA	Generic Design Assessment
GNF	Global Nuclear Fuels
GSR	Generic Safety Requirements
GVHA	GE Vernova Hitachi Nuclear Energy Americas LLC
HELB	High Energy Line Break
IAEA	International Atomic Energy Agency
ICS	Isolation Condenser System
IEC	International Electrotechnical Commission
IEEE	Institute of Electrical and Electronics Engineers
IEFHDC	Internal and External Flood Hazard Design Criteria
LfE	Learning from Experience
LLC	Limited Liability Company
LOCA	Loss of Coolant Accident
MDSL	Master Document Submission List
MELB	Moderate Energy Line Break

NFPA	National Fire Protection Association
NPP	Nuclear Power Plant
NRC	Nuclear Regulatory Commission
NUREG	Nuclear Regulatory (US NRC publications)
ONR	Office for Nuclear Regulation
OPEX	Operational Experience
OPG	Ontario Power Generation Inc.
PA	Protected Area(s)
PER	Preliminary Environmental Report
PPF	Pressure Part Failure
PRHA	Pipe Rupture Hazard Analysis
PSA	Probabilistic Safety Analysis (or Assessment)
PSAR	Preliminary Safety Analysis Report
PSR	Preliminary Safety Report
RCPB	Reactor Coolant Pressure Boundary
RIV	Reactor Isolation Valve
RGP	Relevant Good Practice
RITE	Risk Informed and Targeted Engagements
RO	Regulatory Observation
RP	Requesting Party
RPV	Reactor Pressure Vessel
RQ	Regulatory Query
SAP	Safety Assessment Principle(s)
SBWR	Simplified Boiling Water Reactor
SFAIRP	So far as is reasonably practicable
SMR	Small Modular Reactor
SRV	Safety Relief Valve
SSC	Structure, System and Component(s)
SSG	Specific Safety Guide
SSR	Specific Safety Requirement
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
3D	3 Dimensional

Contents

1. Introduction

1. This report presents the outcome of my internal hazard 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] and associated SSSE chapters (refs. [2], [3], [4], [5], [6], [7], [8], [9], [10], [11], [12], [13], [14], [15], [16], [17], [18], [19], [20], [21], [22], [23], [24], [25]), the associated revision of the Design Reference Report (DRR) (ref. [26]) and supporting documentation.

2. Assessment was undertaken in accordance with the requirements of the ONR Management System and follows ONR's guidance on the mechanics of assessment, NS-TAST-GD-096 (ref. [27]) and ONR's risk informed, targeted engagements (RITE) guidance (ref. [28]). The ONR Safety Assessment Principles (SAPs) (ref. [29]) together with supporting Technical Assessment Guides (TAGs) (ref. [30]), 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. [31]).

1.1. Background

4. The ONR's GDA process (ref. [32]) 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 mainly 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 completed in December 2025.

9. The RP has stated that at this time 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 was 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 (AP) for Internal Hazards (ref. [33]). 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. [34]) and published on the regulators' website.

1.2. Scope

14. The assessment documented in this report was based upon the SSSE for the BWRX-300 (refs. [1], [2], [35], [3], [4], [5], [6], [7], [8], [9], [10], [11], [36], [37], [38], [39], [12], [13], [14], [15], [16], [17], [18], [19], [20], [21], [40], [41], [42], [43], [44], [45], [22], [23], [24], [46], [47], [25], [48]).

15. The RP's GDA scope has been agreed between the regulators and the RP during Step 1. This was documented in an overall Scope of Generic Design Assessment report (ref. [49]). This is further supported by its DRR (ref. [26]) 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 was proposed for consideration in the GDA. The DRR provides a list of the systems, structures 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 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. [27]) and ONR's guidance on Risk Informed, Targeted Engagements (RITE) (ref. [28]).

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 therefore out of scope of this assessment:

- Detailed design which is ready for UK construction was out of scope. High-level conceptual information has been made available to inform my assessment, which was at a suitable level of detail to address fundamental design queries. If the design is intended to progress to the construction phase, then it must meet UK regulatory requirements in the future.
- The UK Advanced Boiling Water Reactor (ABWR) was assessed previously in GDA. Where there are like-for-like design features, the learning from UK ABWR was used to support my confidence on similar operations.

20. My assessment has considered the following aspects:

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

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. [29]) 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. [30]). The TAGs provide the principal means for assessing internal hazard 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:

- **Engineering Key Principles (EKP).** These are engineering design principles fundamental to nuclear safety.
- **External and Internal Hazards (EHA)**. These are relevant to how the design considers hazard impacts and the associated protection against the hazards.
- **Layout (ELO).** These are relevant on how the design minimises the effects of incidents.

29. These are supplemented by supporting SAPs. 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-004 – Fundamental Principles of Safety Assessment (ref. [55]).
- NS-TAST-GD-005 – Regulating duties to reduce risks ALARP (ref. [56]).
- NS-TAST-GD-006 – Design basis analysis (ref. [57]).
- NS-TAST-GD-014 – Internal Hazards (ref. [58]).
- NS-TAST-GD-036 - Redundancy, Diversity, Segregation and Layout of Structures, Systems and Components (ref. [59]).
- NS-TAST-GD-051 – The purpose, scope and content of safety cases (ref. [60]).
- NS-TAST-GD-067 – Pressure Systems Safety (ref. [61]).
- NS-TAST-GD-096 – Guidance on Mechanics of Assessment (ref. [62]).

2.1.3. National and international standards and guidance

31. 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. [63]).
- IAEA SSG-2 – Deterministic Safety Analysis for Nuclear Power Plants (ref. [64]).
- IAEA SSG-61 – Format and Content of the Safety Analysis Report for Nuclear Power Plants (ref. [65]).
- IAEA SSG-64 – Protection against Internal Hazards in the Design of Nuclear Power Plants (ref. [66]).
- IAEA SSG-68 – Design of Nuclear Installations Against External Events Excluding Earthquakes (ref. [67]).
- IAEA SSG-77 – Protection against Internal and External Hazards in the Operation of Nuclear Power Plants (ref. [68]).
- IAEA TECDOC-1791 – Considerations on the Application of the IAEA Safety Requirements for the Design of Nuclear Power Plants (ref. [69]).
- IAEA TECDOC-1936 – Applicability of design safety requirements to small modular reactor technologies intended for near term deployment (ref. [70]).
- WENRA Safety reference levels for existing reactors 2020 (ref. [53]).
- WENRA Safety of New NPP designs (ref. [54]).
- WENRA Applicability of the Safety Objectives to SMRs (ref. [71]).

2.2. Integration with other assessment topics

32. I worked closely with other assessors as part of my internal hazards 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**. I collaborated with civil engineering to seek assurance of civil structures / barriers to ascertain that there would be sufficient design capacity to accommodate hazard loadings or otherwise limit the spread of hazards.
- **External Hazards**. I collaborated with external hazards to seek clarity on the RP's approach for combined hazards.
- **Fault Studies**. They led on the assessment of the RP's methodologies for categorisation of safety functions and classification of structures, systems and components, and the assessment of a preliminary fault schedule, including the consideration of internal hazards.
- **Life fire safety**. They led on seeking assurance on the future substantiation of life fire safety claims and compliance with the Regulatory Reform (Fire Safety) Order 2005. I focused on the interfaces with fire compartmentation and fire protection for nuclear safety.
- **Mechanical engineering.** I collaborated with mechanical engineering on the interfaces between lifting and dropped load analysis.
- **Nuclear site health and safety.** They led on site safety aspects (also known as conventional health and safety). I focused on interfaces such as potential hazard impacts within modular structures and lifting aspects.
- **Probabilistic Safety Analysis (PSA).** They led on the PSA aspects of the combination hazards methodology.
- **Structural Integrity.** They led on the highest integrity component cases and assessment of the RP's Break Exclusion Zone (BEZ) methodology.

2.3. Use of technical support contractors

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

3. Requesting Party's submission

35. The RP submitted the SSSE at the start of Step 2 in four volumes that integrated environmental protection, safety, security, and safeguards. This was accompanied by a head document (PSR SSSE Summary, ref. [1]), which presented 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 presented a summary of the RP's safety case for internal hazards. It also identified 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 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 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 (Global Nuclear Fuels GNF2) that are already currently widely used globally (PSR Chapter 4, Reactor, ref. [4]).

41. The reactor is equipped with several 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 utilises natural circulation and passive cooling rather than active components, reflecting the RP's design philosophy.

42. There are several novel aspects of the reactor design that are relevant to the internal hazards case, namely:

- Integral RPV isolation valves: the BWRX-300 RPV is equipped with isolation valves that are integral to the RPV that rapidly isolate a ruptured pipe to help mitigate the effects of a Loss of Coolant Accident (LOCA). All large fluid pipes with RPV penetrations are equipped with double isolation valves that are integral to the RPV. These isolation valves are designed to shut rapidly to prevent or limit the loss of coolant following a LOCA. (PSR Chapter 5, Reactor Coolant Systems, ref. [5]).

- No Safety Relief Valves (SRVs): the BWRX-300 does not make use of traditional safety or relief valves within the Reactor Coolant Pressure Boundary (RCPB). The design intent is to eliminate many historical sources of identified reactor coolant leakage such that total leakage is reduced through simplification. The large steam volume in the RPV and a large capacity Isolation Condenser System (ICS) provides overpressure protection for design basis events. (PSR Chapter 9A, Auxiliary Systems, ref. [9]). For design extension conditions (DEC) excess pressure in the RPV is dissipated through Ultimate pressure regulation (UPR) system to the containment volume (PSR Chapter 6, Engineered Safety Features, ref. [6]).

- Overpressure in containment: the vent flow path contains a bursting disc, a remote-actuated bypass valve, pressure indicator, isolation valve, check valve, sparger, and associated piping. The overpressure vent flow path connects to the containment exhaust flow path and terminates in the Reactor Equipment Pool. Containment can also be manually vented if required (PSR, Chapter 9A, Auxiliary Systems, ref. [9]).

- The below grade Reactor Building structure and its protection against various hazards and hazard combinations (PSR, Chapter 9B, Civil Structures, ref. [10]).

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 (overall strategy ref. [72]; and strategy pertaining to conventional safety, security and safety case respectively, (refs. [73], [74], [75]). 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 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, 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. [65]), supplemented to include UK specific chapters such as Structural Integrity and Chemistry. The RP has also provided a chapter on As Low As Reasonably Practicable (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.

46. Claims are categorised into SSSE types, with an overarching level 1 claim supported by level 2 and level 3 sub-claims. Security, safeguards and environment topics are covered in other assessments and are not discussed further. At Step 2, the RP has stated it aims to 'demonstrate a viable path towards substantiation for all claims' (SSSE Summary, ref. [1]).

47. The RP's overarching **level 1 safety claim** is that 'the safety risks to workers and the public during the construction, commissioning, operation, and decommissioning of the BWRX-300 have been reduced as low as reasonably practicable (ALARP)' (SSSE Summary, Appendix A, ref. [1]). Level 2 and level 3 claims are discussed in section 3.3 below.

48. The BWRX-300 defence-in-depth (DiD) concept uses Fundamental Safety Functions (FSFs) to define the interface between the Defence Lines (DL) and the physical barriers. For a given event sequence, 'if the functional DLs required to fulfil the FSFs are performed successfully, then the corresponding barriers remain effective' (PSR Chapter 3, Safety Objectives and Design Rules for SSCs, ref. [3]).

49. The BWRX-300 FSFs are; **control of reactivity, removal of heat from the fuel and confinement of radioactive materials**.

50. Detailed discussion of the five DLs are provided in PSR Chapter 3 (ref. [3]). However, the DL objectives are summarised as:

- Level 1: Prevention of abnormal operation and failures.
- Level 2: Control of abnormal operation and detection of failures.
- Level 3: Control of accidents within the design basis.
- Level 4: Control of severe plant conditions, including prevention of accident progression and mitigation of the consequences of severe accidents (divided into DL4a, independent and diverse system functions and DL4b, diverse accident monitoring in the BWRX-300 design).

- Level 5: Mitigation of radiological consequences of significant releases of radioactive material.

3.3. Summary of the RP's case for Internal Hazards

51. The RP acknowledged its internal hazards case required further development and, therefore, identified a high-level route map for the deterministic safety assessment and the approach to meeting relevant good practice (RGP). This is represented in the internal hazard strategy and methodologies within PSR Chapter 15.7, Internal Hazards (ref. [19]).

52. The RP completed an internal hazards identification and screening process. The high-level approach is documented in PSR Chapter 15.7 (PSR Chapter 15.7, Internal Hazards, ref. [19]). This included identifying internal hazards that will be considered in future developments.

53. The RP considered whether SSCs or fundamental safety properties (FSPs) will be challenged by internal hazards. FSPs were largely 'non-functional attributes of the design architecture and its SSCs. They provide assurance that the FSFs will be performed with the expected reliability as, when, and under the conditions required.' (PSR Chapter 15.7, Internal Hazards, ref. [19]).

54. The RP's approach identified that where internal hazards and combination hazards cannot be screened out, they will undergo Deterministic Safety Analysis (DSA) and Probabilistic Safety Assessment (PSA)[1] as part of future development. The current PSR does not present deterministic analysis for relevant internal hazards.

55. For each internal hazard, the RP's strategy includes:

- Definition of the hazard, including a description of the relevant FSPs and design provisions.
- Provision of hazard assessment methodologies.
- Outline of hazard-specific assumptions and conservatisms within the hazard analyses.
- Consideration of relevant cliff-edge effects.
- Presentation of illustrative analyses to demonstrate that associated assessment methodologies are appropriate for a future GDA step 3 or site-specific detailed assessment.
- Consideration of cross-cutting issues in the safety case.

[1] The RP used the term, "Probabilistic Safety Assessment." ONR uses "Probabilistic Safety Analysis." For the purposes of this report, the terms are treated the same.

- Inputs to fault schedule development and any future engineering substantiation required.

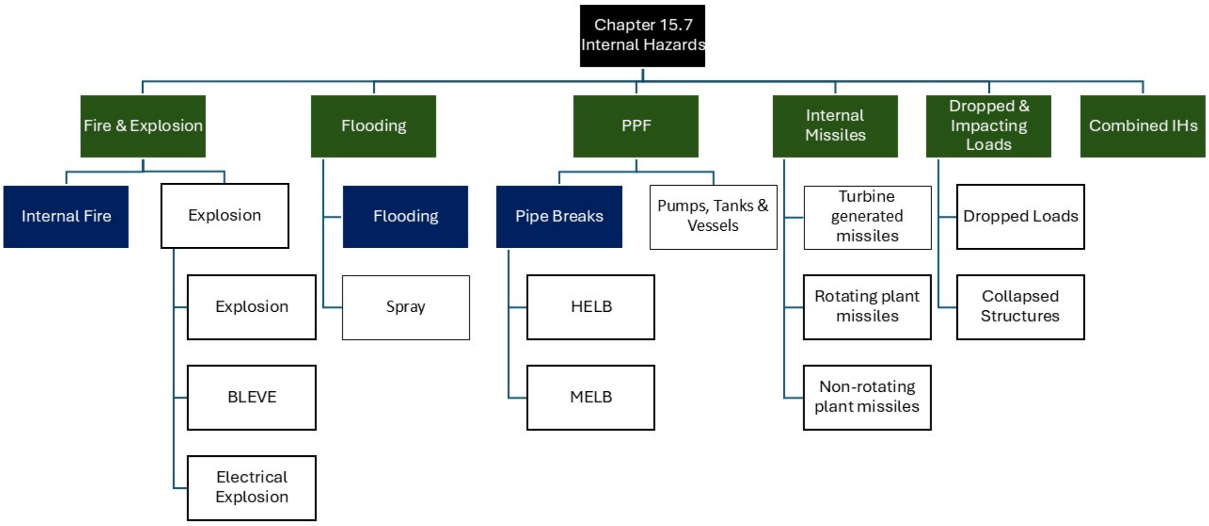

Figure 1 - Overview of the BWRX-300 Internal Hazards Safety Case

56. The internal hazards aspects covered by the BWRX-300 safety case included internal fire and explosion, internal flooding, internal missiles, pressure part failure (PPF)-related hazards (e.g. pipe whip, jet impact) dropped and impacting loads; and combined hazards. The RP's consideration of internal fire, flooding and pipe break-related hazards were at the highest level of maturity at this GDA Step. Consideration of other hazards was conceptual or not presented at Step 2. Figure 1 above illustrates the structure of the case, with the colour coding representing the differing levels of maturity discussed above.[2]

57. The RP presented the design's virtual 3D model of the power block to ONR at an internal hazards interaction (Interaction on 16 April 2025, ref. [76]). This was used to give early consideration of site layout, operational control and any operational constraints. This was not submitted formally as part of the RPs safety case.

58. For the internal hazards case at Step 2, the RP summarised the nature of the internal hazard, the analysis methodology to be developed, identifying where there was divergence from RGP. Where the RP noted gaps, it identified further development of the safety case needed and recorded key tasks via a Forward Action Plan (FAP) (ref. [77]).

[2] Figure 1 colour coding: (green) hazards presented at GDA Step 2; (blue) highest level of maturity at GDA Step 2; (unshaded) hazards at a conceptual level or not presented at Step 2. An outline of the combination hazard methodology was presented at Step 2 with sub-topics to be considered as part of future development of the safety case.

59. The key level 2 claim associated with internal hazards is:

- **Claim 2.3** – A suitable and sufficient safety analysis has been undertaken which presents a comprehensive fault and hazard analysis that specifies the requirements on the safety measures and informs emergency arrangements (PSR Chapter 15.7, Appendix A, ref. [19]).

60. Other pertinent level 2 claims are:

- **Claim 2.1 –** The functions of systems and structures have been derived and substantiated taking into account Relevant Good Practice (RGP) and Operational Experience (OPEX), and processes are in place to maintain these through-life (Engineering Analysis).
- **Claim 2.4** – Safety risks have been reduced as low as reasonably practicable (SSSE Summary, Appendix A, ref. [1]).

61. The relevant level 3 sub-claims associated with internal hazards are:

- **Claim 2.3.1** – All initiating events with the potential to lead to significant exposure or release of radioactive material, including the effects of internal and external hazards have been identified and appropriately assessed.
- **Claim 2.3.2** – Design basis events have been appropriately assessed to specify requirements on safety functions and on safety measures and their effectiveness.
- **Claim 2.3.3** – Beyond Design Basis and Severe Accidents have been appropriately assessed to identify further risk reducing measures and inform emergency arrangements.

62. Additionally, the civil structure level 3 claims supporting claims 2.1 and 2.4 respectively are also pertinent to ensuring the design is robust against internal hazards (PSR Chapter 9B, Civil Structures, Appendix A, ref. [10]).

Relevant Sub-Claims of 2.1

- **Claim 2.1.2** – The design of the system/structure has been substantiated to achieve the safety functions in all relevant operating modes.
- **Claim 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.

Relevant Sub-Claims of 2.4

- **Claim 2.4.1** – Relevant Good Practice (RGP) has been taken into account across all disciplines.

- **Claim 2.4.2** – OPEX and Learning from Experience (LfE) has been taken into account across all disciplines.
- **Claim 2.4.3** – Optioneering (all reasonably practicable measures have been implemented to reduce risk).

63. The RP's internal hazards primary arguments supporting the claims at Step 2 were:

- **INHA.1** - National and international guidance has been used to derive a comprehensive list of internal hazards for consideration. Screening and grouping have been applied to identity those which require safety assessment.
- **INHA.2** - Design basis internal hazard sources and credible hazard combinations are systematically located and characterised.
- **INHA.3** - FSPs claimed to prevent or mitigate internal hazards and credible hazard combinations are appropriately identified.
- **INHA.4** - Appropriately conservative analysis demonstrates that the identified FSPs ensure that DL3 remains capable of reaching and maintaining a safe state and radiological releases are tolerable and ALARP.

3.4. Basis of assessment: RP's documentation

64. The principal documents that have formed the basis of my internal hazards assessment of the SSSE were:

- BWRX-300 Safety Strategy (from Step 1) (ref. [72]).
- BWRX-300 General Description (from Step 1) (ref. [78]).
- PSR Chapter 3 – Safety Objectives and Design Rules for Structures, Systems and Components (ref. [3]).
- PSR Chapter 5 – Reactor Coolant System and Associated Systems (ref. [5]).
- PSR Chapter 9B – Civil Structures (ref. [10]).
- PSR Chapter 15 – Safety Analysis (including Fault Studies, PSA and Hazard Assessment) (ref. [12]).
- PSR Chapter 27 – ALARP Evaluation (ref. [25]).
- Combined Hazards Topic Report (ref. [79]).

3.5. Design Maturity

65. My assessment was based on revision 3 of the DRR (ref. [26]). The DRR presents the baseline design for GDA Step 2, outlining the physical system descriptions and requirements that form the design at that point in time.

66. The reactor building and the turbine building, along with the majority 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 PA boundary and the PA access building.

67. 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 BL 3 (where the design is ready for construction).

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

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

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

70. Further hazard groupings were also identified (PSR Chapter 15.7, Internal Hazards – Site Specific and Future Work, ref. [19]). These hazards included:

- Pipeline Accident
- On-site hazardous materials
- Transportation Accidents
- Electromagnetic interference / radio frequency interference
- Vibration
- Static Electricity
- Methane
- Snow Melt
- Biological Agents
- Wildlife.

71. These hazards, in conjunction with those identified in Figure 1, will be addressed as part of future development of the case, including deterministic analysis to meet UK regulatory requirements using the proposed approach outlined in the BWRX-300 UK GDA Safety Case Manual Specification (ref. [80]).

4. ONR assessment

4.1. Assessment strategy

72. 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 internal hazards as described in sections 1 and 3 of this report. My assessment strategy set out in this section, defined how I chose which matters to target for assessment. My assessment was consistent with the delivery strategy for the BWRX-300 GDA (ref. [81]).

73. The RP engaged with regulators internationally, including the US NRC and the Canadian Nuclear Safety Commission (CNSC). It proposed 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.

74. Whilst there are no operating BWR plants in the UK, ONR has previously performed a four-step GDA on the Hitachi-GE UK ABWR (ref. [82]). My assessment scope included aspects of the BWRX-300 which are novel or specific to the BWR technology. Using learning from UK ABWR GDA, I also focused on the fundamental adequacy of the design on facilities where the impact of hazards should be considered at early design stages and where risks would be the highest or least well controlled. I have not reassessed inherent aspects of BWR technology which were considered in significant detail for the UK ABWR and judged to be acceptable.

75. Given the level of design maturity, my strategy involved sampling fundamental layout considerations and novel aspects of the BWRX-300 SSSE case to ascertain whether the RP provided adequate assurance that sufficient design capacity will be provided to protect and mitigate against key hazards and these had been considered at Step 2.

4.2. Assessment Scope

76. My assessment scope was consistent with the GDA scope agreed between the regulators and the RP during Step 1 and detailed in Section 1.2 of this report. Key areas of focus identified for internal hazards were the screening criteria (e.g. frequency; pressure or temperature for line breaks) for hazard scenarios and assessment of low frequency, high consequence events such as turbine disintegration and combined hazards. This section also outlines the submissions that I sampled, the standards and criteria that I judged against and how I interacted with the RP and other assessment topics.

77. In line with RITE principles, I focused my sampling on buildings of greatest nuclear safety significance, namely the Reactor Building and Turbine

Building. I also used learning from the UK ABWR and other LfE to focus on key hazard scenarios that should be considered in the early design stages. For example, these included internal fire within the Reactor Building, pressure part failure of main steam lines / feedwater lines, impact from turbine disintegration and the dropped impact from a spent fuel cask. I have undertaken a broad review of the fundamental claims and supporting arguments related to internal hazards. To support this, I sampled a targeted set of the claims and arguments identified under Section 3.3 (Summary of the RP's case for Internal Hazards).

4.3. Assessment

78. My assessment covered the following, incorporating the RP's safety case structure:

- Safety Case Submissions (s.4.3.1.).
- Internal Hazard Identification (s.4.3.2.).
- Internal Fire (s.4.3.3.).
- Internal Explosion (s.4.3.4.).
- Flooding (s.4.3.5.).
- Pressure Part Failure (s.4.3.6.).
- Internal Missiles (s.4.3.7.).
- Dropped and Impact Loads (s.4.3.8.).
- Combined Hazards (s.4.3.9.).
- Novel Aspects (s.4.3.10.).
- Segregation, Redundancy and Diversity (s.4.3.11.).
- ALARP (s.4.3.12.).

4.3.1. Safety Case Submissions

79. I sampled the RP's safety case submissions in line with IAEA standards SSG-61 and SSG-64 (refs. [65], [66]), WENRA safety reference levels (ref. [53]) and ONR guidance (SAPs and Internal Hazards TAG, refs. [29] and [58]). I focussed on determining the fundamental adequacy of the RP's safety case. This included the assessment of the RP's safety case claims, arguments and where relevant, evidence commensurate with Step 2 to demonstrate internal hazards that may have the potential to impact nuclear safety are adequately identified and considered in design.

80. The RP claims it has an overarching suitable and sufficient analysis (**Claim 2.3**), supported with recognised guidance (**Argument INHA.1**), adequate characterisation (**Argument INHA.2**) and FSPs (**Arguments INHA.3, INHA.4).**

81. To review the 'Golden Thread' link to the claims, I sampled the DL1 approach (prevention; summarised in section 3.2), where the objectives are to:

- Eliminate or minimise sources of internal hazards.
- Contain or mitigate consequences of internal hazards.
- Ensure continued availability of DL3 functions (control; summarised in section 3.2).
- Fail-safe features are included in DL3 function implementation.
- Provisions to monitor FSF performance and physical barrier integrity.
- Equipment not supporting DL3 functions are appropriately protected or qualified.

82. I sampled the following submissions to assess the adequacy of information to underpin the above claims and arguments:

- The RP's Safety Strategy (ref. [72]) which provided an overall framework of the safety analysis.
- The RPs Safety Objectives and Design Rules (PSR Chapter 3, ref. [3]). This provided high-level information on SSCs, design assumptions and FSPs.
- PSR Chapter 15 and sub-chapters which covered safety analysis. Internal hazards were addressed in Chapter 15.7 (ref. [19]). The RP defined the hazard and assessed this against FSPs being met. I sampled high level methodologies with certain hazards more developed than others at Step 2.
- PSR Engineering Chapters. These are referenced under the specific internal hazards below where relevant.

83. No additional analysis or detailed substantiation beyond the submission was provided at Step 2 (RQ-01711, ref. [83]).

84. It was my expectation that a safety case 'should be accurate and demonstrably complete*'* in line with SAP SC.4 (Safety case characteristics, SAPs, ref. [29]) relative to fulfilling a fundamental assessment in line with ONR's guidance to Requesting Parties (ref. [32]). The RP acknowledged that further development of the case was required in subsequent project steps and developed a Forward Action Plan (FAP) for future organisations to address items to meet UK regulatory requirements (FAP, ref. [77] and RQ-01711, ref. [83]).

85. The RP's high level approach to internal hazards and my sampling of PSR chapters provided confidence that key information relating to fundamental internal hazard aspects was available to underpin **Claim 2.3** (an overarching suitable and sufficient analysis) (Safety strategy and PSR Chapter 9B, Civil Structures, refs. [72] [10]).

86. The potential impact of internal hazards on civil structures and layout considerations should be considered in the early design stages (SAP ELO.4, Layout). I therefore sampled supporting information of potential hazard impacts on civil structures (PSR Chapter 9B, Civil Structures, ref. [10]) associated with **Claim 2.1** (derivation of system and structure functions) and **Claim 2.4** (safety risks reduced to SFAIRP). This included sampling of the Power Block General Arrangement drawing and the Reactor Building section drawing provided as part of RQ-01710, (ref. [84]) and a walkthrough of the 3D model (interaction on 16 April 2025, ref. [76]). This information provided confidence that fundamental segregation and separation principles have been included early in design development. Further details of layout specifics sampled are included as part of hazard considerations below.

87. High-level supporting information such as SSC robustness (PSR Chapter 3, Safety Objectives and Design Rules for SSCs, ref. [3]) and systems including water and fire protection systems (PSR Chapter 9A, Auxiliary Systems, ref. [9]) also informed my assessment of **Claim 2.1** (derivation of system and structure functions) and **Claim 2.4** (safety risks reduced to SFAIRP).

88. Based on the safety strategy, the available information in other PSR Chapters, the arrangement and section drawings and the identification of future commitments relating to internal hazards analysis, I judge the RP provided confidence that the internal hazards aspects of the safety case are fundamentally sufficient at Step 2.

4.3.2. Internal Hazard Identification

89. The RP's hazard identification approach is covered in the Safety Strategy (ref. [72]) and the Safety Objectives and Design Rules (ref. [3]). The RP indicates it followed:

- IAEA Specific Safety Requirements, SSR-2/1, Safety of Nuclear Power Plants: Design (ref. [63]).
- IAEA General Safety Requirements, GSR Part 4, Safety Assessment for Facilities and Activities: General Safety Requirements (ref. [85]).
- IAEA Specific Safety Guide, SSG-64, Protection against Internal Hazards in the Design of Nuclear Power Plants (ref. [66]).

90. Table 15.7-1 of the PSR summarises the internal hazards reviewed by the RP at GDA Step 2. These were grouped under internal fire, internal explosion, internal flooding, pressure part failure, internal missiles, dropped / impacting loads and certain combined hazards. Further resolution or granularity beyond the high level hazard groups above was not provided at this stage. Combined hazards are discussed separately below (section 4.3.9).

91. The RP also identified further internal hazards, indicating additional groupings (section 3.5, para. 69) and recognised the need to address them as the design develops. This is recorded in FAP items for hazards not addressed at GDA Step 2 (ref. [77]).

92. The RP applies screening criteria based on frequency of occurrence of internal hazards of 10^{-7} per year; meaning that hazards below that frequency of occurrence are not considered further. I judge that this screening criterion is consistent with EHA.19 and paras. 235 and 631 of ONR SAPs, providing frequencies of internal hazard-related events are appropriately derived. The IAEA guidance on protection against internal hazards (IAEA, SSG-64, ref. [66]) states that DSA should be developed, to include characterisation of the internal hazards and their consequential impact on SSCs. I sought clarification on the specific set of internal hazards that would be considered and characterised in the DSA versus a reduced set that had been identified for the PSA through RQ-02085 (ref. [86]). The RQ-2085 response confirmed that the deterministic approach will not be limited by the reduced PSA hazard scenario list. The internal hazards DSA will be developed independently of the PSA process.

93. I am content that the RP presented several key hazards it has considered at Step 2 and identified further hazards to be assessed as part of future development. I am satisfied the RP applied an effective process to hazard identification (SAP EHA.1), adopting relevant good practice (RGP) in the derivation of hazard scenarios (IAEA SSG-64, WENRA Safety Reference Levels, refs [66], [53]). I consider this supports its argument of **INHA.1** (use of national and international guidance to derive internal hazards).

4.3.3. Internal Fire

94. My assessment focused on internal fire due to the significance of fire protection considerations in design for nuclear safety. I sampled the RP's approach to internal fire analysis and potential impacts on key SSCs. The assessment of life fire safety aspects was covered in assessment AR-1361 (life fire safety assessment, ref. [87]).

95. The RP identifies internal fire as a hazard, with the potential to impact the nuclear safety classified buildings, SSCs and challenge FSFs.

96. It is my expectation that fire hazards should be identified and characterised (SAP EHA.1, identification and characterisation and EHA.14, sources of harm). As part of the PSR submission (PSR Chapter 15.7, Internal Hazards, ref. [19]), the RP shared the design philosophies incorporated in the design for protection against internal hazards. I raised RQ-01933 (ref. [88]) to seek further detail on the methodologies for identification and assessment of internal hazards, in addition to the high level design philosophies included in the PSR submission. In response to the RQ, the RP highlighted that evaluation of site fire hazards and demonstration of fire safety adequacy at the site are underpinned by both the Fire Hazard

Assessment (FHA) and the Fire Safe Shutdown Analysis (FSSA). The RP noted that these documents were not developed for a UK context, however they do broadly align with expectations in ONR Internal Hazards TAG and IAEA SSG-64. These documents state that the building and fire protection design follow fire protection standards such as the National Fire Protection Association (NFPA) Fire Protection Standard (ref. [89]), US NRC Fire Protection Guidance 1.189 (ref. [90]), and Canadian Nuclear Society (CNS) Fire Protection Guidance CSA N293 (ref. [91]).

97. The BWRX-300 design provides for segregation by fire barriers between three independent divisions, containing both mechanical and electrical safe shutdown equipment and components. I considered segregation is in place to ensure that an internal fire is contained to one division and shutdown capability may be initiated via an alternative division. I judge this meets internal hazards expectations of segregation and redundancy as per SAP EDR.2. The FHA and FSSA state that the walls, floors, ceilings and enclosures shall be of sufficient fire resistance based on combustible loadings in the area, to ensure fires do not spread to adjacent divisions. IAEA SSG-64 (ref. [66]) also requires that the plant can be brought to, and maintained in a safe state in the event of an internal hazard occurrence, including when equipment is unavailable owing to planned maintenance. The RP's high level principles indicate that for a most limiting single failure, there are available DLs, including DL4 functions with diverse engineering safety systems to maintain FSFs.

98. The fire containment approach, if applied consistently throughout the design, is in line with the Internal Hazard TAG (ref. [58]) and IAEA SSG-64 (ref. [66]) fire containment approach. In general, the fire barriers will be rated to three hours fire resistance. However, the RP also noted that the fire resistance may be lower than three hours where a suitable case may be made e.g. based on lower hazard consequence. At this stage I am content that the RP is working toward underpinning suitable and sufficient safety analysis **Claim 2.3** (an overarching suitable and sufficient analysis) and the civil structures **Claim 2.1.2** (substantiation of design of the system/structure). However, internal fire hazards will require further assessment (in either a future Step 3 GDA or during site specific activities) with suitable and sufficient supporting evidence to demonstrate preference of fire containment approach in design and to substantiate the claims and proposals made in the GDA Step 2 submissions. The RP has created FAP items (PSR15.7-62 and PSR15.7-73, ref. [77]) covering the development of deterministic assessment to assess internal fire in future design phases.

99. The RP presented an example illustrative analysis for assessing fire hazards. The examples chosen were rooms which have potential ignition sources, combustible inventory and also house systems claimed for shutdown; including an area where no FSFs fail and an area where the FSFs fail (Interaction on 12 March 2025, ref. [92]). The illustrations showed the methodology currently in place to identify areas where safe shutdown is

at risk. A detailed methodology for fire safe shutdown analysis is still to be developed and will form part of the DSA. However, I consider that for Step 2, with the segregation principles adopted (discussed above), and the use of appropriate fire retardant materials, both within SSCs and as consumables, are a good starting point for minimising the fire impact to nuclear safety. As the design and methodologies are still developing, this will require further assessment to confirm that fire risk is minimised throughout design and the fire impact on all SSCs is adequately considered. This includes but is not limited to application of fire containment approach, consideration of all combustible materials when quantifying the fire loading in different areas and use of applicable fire growth curves, in line with UK regulatory expectations.

100. I considered that the example scenarios, along with the fire safety strategy provides confidence that the RP has a systematic and methodical approach for assessing the impact from internal fires on SSCs, which is in line with SAP EHA.1 (identification and characterisation).

101. To ensure FSF are fulfilled and resilient against fires, the RP applied a defence in depth (DiD) strategy. The RP's safety case (PSR Chapter 15.7, Internal Hazards, ref. [19]) highlights the following elements:

- Design measures to reduce or eliminate, where practical, combustible materials and ignition sources. Materials used in plant design minimise the likelihood of starting or propagating fires.
- Means provided to quickly detect and extinguish or control fires to minimise the adverse effects of fires on SSCs.
- Limit the effects/spread of fire using measures such as fire separation, segregation and design of mechanical, electrical systems to prevent the spread of fires.

102. From this, I was content the RP had embedded defence in depth considerations in the design of the plant for resilience against internal fire, conversant of SAP EKP.3 (DiD).

103. The RP did not analyse cliff-edge effects at Step 2. PSR Chapter 15.7 (ref. [19]) states that the cliff-edge effects will be considered in the future when the deterministic assessment for the Internal Fire Hazard has further matured, and this was covered by FAP item PSR15.7-70 (ref. [77]).

104. The RP uses recognised design standards (e.g. NFPA (Fire Protection Standard, ref. [89]), US NRC (Fire Protection Guidance 1.189, ref. [90]), Canadian Nuclear Society (CNS) standards (Fire Protection Guidance CSA N293, ref. [91]) for building and fire protection design and I consider it a relevant argument to support **Claim 2.1.3** (use of RGP and OPEX to support reducing risks to ALARP). Further development in future design stages will be required for all key internal fire scenarios to ensure associated analysis is suitable and sufficient when reviewed against IAEA

and SAPs (EHA.1, identification and characterisation; EHA.3, design basis events and EHA.7, cliff-edge effects).

105. Further characterisation will be required to align the internal fire hazards and the overall case. However, I have not identified any fundamental shortfalls with the RP's analysis of internal fires and the potential impact on key SSCs. I am content that the RP has made sufficient progress for Step 2 in line with recognised international guidance and standards (as discussed in the paragraph above). The FAP process gave me confidence that gaps will be closed in future design development. I judged this as adequate at this stage of the design's development.

4.3.4. Internal Explosion

106. The RP identifies internal explosions as an internal hazard. The RP indicates that explosion, Boiling Liquid Expanding Vapour Explosion (BLEVE) and electrical explosion will be topics of future assessment (PSR Chapter 15.7, Internal Hazards, ref. [19]). Internal hazard topics are summarised in paragraph 56. In addition to these, IAEA advises that:

> 'chemical explosions, boiling liquid expanding vapour explosions induced by fire exposure, oil mist, pressure vessel blast, high energy arcing faults accompanied by rapid air expansion and plasma buildup should be considered. Also, consequent effects of explosions (e.g. the rupture of pipes conveying flammable gases) should be taken into account in the identification of explosion hazards.' (IAEA, SSG-64, ref. [66]).

107. It is my expectation that the safety case systematically identifies and characterises all explosion sources in accordance with SAPs EHA.1 (identification and characterisation), EHA.6 (analysis) and EHA.14 (sources of harm), alongside clear descriptions of inventories, flammability / explosivity properties, storage and operating conditions (pressure, temperature) and locations.

108. As identified in my Assessment Plan (AP) (ref. [33]), it is also my expectation that accumulation of gases (including radiolytic gases) / vapours / stable mists reaching flammable concentrations at the release point or elsewhere in or outside process are considered. This should consider specific learning from past incidents (e.g. Brunsbüttel and Hamaoka). Consideration of the off-gases related to fuel failure is included in the ONR Chemistry assessment (ref. [93]).

109. Learning from UK ABWR also highlighted that inherent to the BWR design is the formation of hydrogen and oxygen in stoichiometric quantities under normal operations, due to the radiolysis of water in the Reactor Pressure Vessel (RPV). The internal hazards aspects of the safety case for the generation, accumulation, management and mitigation of radiolytic gases during normal operations was not presented at Step 2. However, this is

expected to be covered as part of FAP item PSR15.7-68 (ref. [77]). With the combination of hydrogen management (incorporating learning from past incidents related to accumulation of explosive hydrogen concentrations within pipework) and the reactor building primarily below grade, vapour cloud explosions and battery induced explosions should also be analysed (Internal Hazards TAG, ref. [58]).

110. The identification of explosion sources and development of methodologies for characterisation of internal explosions was not addressed as part of Step 2 submissions, and is covered in FAP item PSR15.7-68 (ref. [77]) supporting future detailed characterisation. I note this includes explosion characterisation, assessment of the impact on key safety measures, SSCs and incorporation of any subsequent changes required to be made to the design. However, the RP has incorporated design measures to minimise the explosion risk e.g. the primary containment is inerted with nitrogen during normal operation of the facility, to support hydrogen management requirements under standard operating conditions. Reactor chemistry to manage hydrogen concentrations in vessel and ex-vessel also provides DiD and is considered further in the ONR Chemistry assessment (ref. [93]).

111. I consider the design is being developed in accordance with appropriate IAEA guidance. Furthermore, it includes provision of engineered safety systems to prevent the formation of flammable atmospheres. The FAP item PSR 15.7-68 (ref. [77]), supporting future detailed characterisation, gives me confidence that explosion hazards have been reviewed at a fundamental level, and is aligned to SAP EHA.1 (identification and characterisation). I judge this as adequate at this stage of the design development.

4.3.5. Internal Flooding

112. I sampled the RP's approach to internal flooding analysis. I raised a RQ to request the RP's flooding methodology and the additional information provided enabled fundamental assessment (RQ-01771, ref. [94]). The Internal and External Flood Hazard Design Criteria (IEFHDC) (ref. [95]) identifies sources of flooding such as pipe ruptures, tank ruptures, overfill of tanks, pump seals, fire suppression piping, sprinkler systems, eye wash stations and chemical showers.

113. The RP proposed IEFHDC identified and characterised flooding scenarios using US NRC NUREG 0800 (ref. [96]) and ANS/ANSI 56.11 Design criteria, (ref. [97]). I consider these as an adequate starting point for internal flooding hazard characterisation.

114. Deterministic analysis of internal flooding and hazard reduction was not presented at Step 2. This was identified by the RP as FAP items PSR15.3 and PSR15.7-64 (FAP, ref. [77]) to be addressed as design progresses. Cliff edge effects from internal flooding were also not identified at Step 2

but FAP item PSR15.7-70 was noted for future development (FAP, ref. [77]).

115. The RP's case centres on preventative and mitigation measures using IAEA guidance SSG-64 (ref. [66]). The methodology considers inexhaustible sources of flood water from pipe breaks. The methodology also identifies the further requirement to consider other inexhaustible flooding sources such as steam, fire water and feedwater (RQ-01771, ref. [94]).

116. The RP states internal flooding will be mitigated by layout of the plant. This includes physical separation of redundant SSCs or relocation of SSCs to elevations above assumed flood levels. The RP identified design features such as leak detection and isolation equipment to minimise the impact of flooding. Also bunding, sumps, pumps and drainage paths are identified to divert flood waters to the base of the reactor building where they would collate, this was represented in layout diagrams and the virtual 3D model (Interaction on 16 April 2025, ref. [76]).

117. The RP presented its preference for mitigation, I noted that vulnerable equipment is relocated to elevations above assumed flooding levels, or where this cannot be achieved, an engineered barrier or enclosure is provided. A further measure discussed was environmental qualification. The methodology explains that if it is determined that environmental qualification is necessary for the SC1 equipment to remain operable, then equipment shall be qualified for the maximum flood level (RQ-01771, ref. [94]).

118. This approach gives me confidence that SSCs would either be located above assumed flooding levels, separated, installed in waterproof enclosures, or environmentally qualified (RQ-01771, ref. [94] and design rules specified in PSR Chapter 3, Safety Objectives, ref. [3]). At this stage of the design's development, flooding levels are assumed, based on the inventory of the ICS pool. This was discussed during the 3D model virtual walkthrough (Interaction on 16 April 2025, ref. [76]). I am content that this assumption is conservative given the ICS is largest water source in the design.

119. At a fundamental level, the RP also indicated that the design of pipework was to the American Society of Mechanical Engineers (ASME) Boiler Pressure Vessel Code (BPVC-III NB, ref. [98] and BPVC-III NCD, ref. [99]), which would also reduce risks of pipe breaks. I considered this is as recognised RGP. While the RP did not refer to the recent 2025 update of the BPVC (ref. [100]) the changes do not materially alter my judgement.

120. Spray release hazards were identified by the RP, as well as the outline of a characterisation methodology. The RP stated that buildings that contain SC1 equipment and components shall be evaluated for areas where spray wetting can occur. Furthermore, the RP identified factors to be considered,

these included; assumptions and conservatisms, consideration of cliff-edge effects, illustrative analysis and cross cutting issues (PSR Chapter 15.7, Internal Hazards, ref. [19]).

121. I sampled the 3D virtual model to ascertain the flooding routes and where flood waters would collate. The RP claimed that loss of the largest water source (the ICS pool), would not have an adverse effect on SSCs. This was demonstrated in the virtual walkthrough of the 3D model (Interaction on 16 April 2025, ref. [76]). The RP explained that discharging of the ICS pool would cause any water to be directed away from any vulnerable safety systems via bunding and drainage paths which would divert waters to collate in the base of the reactor building. The assumed flooding levels would not challenge SSCs as they would either be protected by engineered barriers, enclosures, elevated above flooding levels, or where this is not possible, qualified as suitable to be submerged. This gave me confidence that there are no fundamental issues in the design at this design reference point.

122. The combination of the 3D model, the methodology applying IAEA guidance/ASME code, along with the associated FAP items provided me with confidence that the approach for internal flooding was in line with RGP (SAPs EHA.3 (design basis events) and EHA.6 (analysis); IAEA SSG-64 and the Internal Hazards TAG, refs. [29], [66], [58] respectively) and the case was progressing toward **Claim 2.3.1** (identification and assessment of all initiating events). I consider the proposed methodology made progress against **Claim 2.1.3** (use of RGP and OPEX to support reducing risks to ALARP) and **Claim 2.4.1** (use of RGP in all topic areas). I judged this as adequate at this stage of the design's development.

123. I consider further development is required to characterise unmitigated consequences of internal flooding to assure the adequacy of DiD (SAP EKP.3, defence in depth). This has been identified as part of future development of the DSA (FAP item PSR15.7-62, ref. [77]).

4.3.6. Pressure Part Failure (PPF)

124. Hazards associated with failure of pressurised equipment, commonly referred to as Pressure Part Failure (PPF) are an important part of the internal hazards assessment scope in PWR and BWR reactor technology. As part of Step 2, the RP focused on pipe break, which I sampled. I raised RQ-01830 to seek clarity on the proposed methodology for PPF. This was not included in the original submission. In the Step 2 submission the RP focussed specifically on high energy line breaks (HELBs) and moderate energy line breaks (MELBs), and this is described in the RP's Rupture Hazard Analysis (PRHA) (RQ-01830, ref. [101]). The RP identified that hazards arising from failure of pumps, tanks and other pressurised sources would be topics of future assessment (PSR Chapter 15.7, Internal Hazards ref. [23] and refer to Figure 1 discussed earlier).

125. The RP classified pipework into high energy pipework if the temperature is above ≥ 95ºC, pressure (≥ 1.9MPa(g)) and/or gas is held above atmospheric pressure. I am content the classification aligns with NUREG-0800 and IAEA SSG-64 (refs. [66], [96]).

126. The proposed methodology has identified circumferential and longitudinal pipe breaks in several locations for high-energy pipework to support the Step 2 submission. These break locations are identified as:

- Terminal ends of pipe runs at points of maximum constraint for example, vessel connections, pumps, valves and fittings that are rigidly attached to structures (PRHA) (RQ-01830, ref. [101])
- Intermediate break locations:
  - Breaks in SC1 pipework are postulated at intermediate locations based on usage and at stressed locations.
  - Breaks in SC2 and SC3 pipework are postulated at intermediate locations such as; elbow, tee, cross, flange or non-standard fittings. Furthermore, consideration is given to high stress locations (PRHA) (RQ-01830, ref. [101]).

127. For moderate energy pipework the following are postulated:

- Through-wall cracks that are located in areas containing systems and components important to safety, but where no high-energy fluid systems are present, through-wall leakage cracks are postulated at the most adverse location to determine the protection needed to withstand the effects of the resulting water spray and flooding.
- Through-wall cracks at axial and circumferential locations that result in the most severe environmental consequences
- Through-wall cracks in moderate-energy piping are postulated in piping located adjacent to essential SSCs (PRHA) (RQ-01830, ref. [101]).

128. I noted the methodology had some exclusions from the through-wall pipe cracks in moderate energy pipework, namely:

- Moderate-energy through-wall cracks need not be postulated in DN 25mm (1-inch NPS) and smaller piping.
- Through-wall cracks in moderate-energy pipework located in an area in which a break in high-energy piping is postulated, provided such through-wall cracks would not result in more limiting environmental conditions than the high-energy piping break (PRHA) (RQ-01830, ref. [101]).

129. The RP has considered a double ended guillotine break (DEGB) for HELB as a failure mode. At this stage, the RP did not consider DEGB or pipe whip for MELBs. As outlined in ONR's internal hazards TAG (ref. [58]), it is

generally considered that pipes operating below the high energy threshold would not whip, however, consideration should be given to cliff-edge effects especially for systems operating close to the above pressure / temperature levels (SAP EHA.7, cliff-edge effects). The assumption for no pipe whip below 2.0Mpa is not rigorous. As the design matures, further consideration is expected on the potential for some level of whip with flexible pipework or some low schedule pipework on lower pressure systems. The impacts should not be damaging to walls or structures but should be considered for the potential effects on more sensitive plant components.

130. Where arguments are made based on hazards associated with high energy pipework failure bounding consequences of the failure of pipework operating in states with lower stored energy, the RP should demonstrate that those effects are comparable and not overly simplified. For example, moderate energy pipework with a larger bore, holding a larger inventory, can pose a more significant flooding or steam release / humidity challenge than higher energy pipework. Similarly, the consequences of failures of moderate energy pipework should not be limited to consideration of partial failures in line with SAPs EHA.1 (identification and characterisation), EHA.3 (design basis events) and EHA.6 (analysis). FAP items PSR15.7-64 and PSR15.7-65 were identified to close this gap in future development (FAP, ref. [76]).

131. The RP's PRHA stated that high energy pipelines that operate for less than 2% of the time are classified as moderate energy pipelines. This is based on US NRC guidelines but is not consistent with ONR SAP NT.2 (time at risk) or the ONR Internal Hazards TAG (ref. [58]). ONR notes that although an operation may be infrequent, in the event of a failure of a high energy line, the consequence is not reduced, and hence appropriate protection and mitigation is required. The RP has stated the 2% rule will not be applied to the UK design FAP item PSR15.7-21 (FAP, ref. [77]).

132. The methodology for pipe breaks (PHRA) shows a degree of alignment with the requirements of IAEA, SSG-64 (ref. [66]) and the ONR Internal Hazards TAG (ref. [58]). This is a good starting point and gives confidence that the RP is working towards meeting UK expectations.

133. The RP's proposed methodology considers dynamic effects of pipework failure. These include pipe whip and jet impingement. The methodology considers the direct and indirect effects of a whipping pipe on essential SSCs inside the zone of influence, similarly jet thrust and spray wetting on safety significant SSCs are evaluated. The RP has considered mitigatory actions if dynamics effects on essential SSCs are unacceptable, this includes the relocation of SSC, pipe whip restraints resulting in a smaller zone of influence, protective barriers or jet shields or qualification of SSCs to operate following the pipe rupture interaction (RQ-01830, ref. [101]).

134. The RP also reviewed the inclusion of pipe whip restraints. These would be designed to ASME and American Institute of Steel Construction (AISC) codes (BPVC-III-1-NF, ref. [102] and ANSI/AISC N690-18, ref. [103]). I noted the 2024 update of the AISC code (ANSI/AISC N690-24, ref [104]) but this did not materially alter my judgement.

135. Cliff-edge effects such as lump masses, elbows/bends, the production of secondary missiles, and the assessment of simultaneous whip and jet effects were identified in the PHRA methodology (RQ-01830, ref. [101]), in line with SAP EA.7 (cliff-edge effects), but were not fully characterised at Step 2. As the design develops, the RP will consider specific sources and locations, this will be used to inform the safety case.

136. The PRHA methodology (RQ-01830, ref. [101]), also considered the consequences of environmental effects such as temperature, pressure and flooding; these include the relocation of SSCs, providing barriers or enclosures for SSCs, installing SSCs above assumed flooding heights or ensuring SSCs are qualified for operation whilst being submerged. The RP used US NRC guidance and the ANS design basis guidance (NUREG-0800, ref. [94]; NUREG/CR-7275, ref. [105] ANSI/ANS 58.2, ref. [106] to consider environmental and dynamic effects and their impact to SSCs. Those effects identified as part of the RP’s hazard identification via the PHRA methodology broadly aligned with the expectations ONR Internal Hazards TAG (ref. [58]) and IAEA SSG-64 (ref. [66]).

137. The RP’s high-level approach did not identify specific vulnerability of SSCs at Step 2. An early indication of hazard prevention was summarised from the codes and standards used at Step 2. The RP presented arguments that source pipework would be designed to ASME codes (BPVC, refs. [98], [99]). Safety critical electrical and mechanical equipment would be relocated, protected by barriers, enclosures and jet shields or where appropriate environmentally qualified to IEEE /IEC 60780-323-2016 (ref. [107]) for electrical equipment. The RP’s approach broadly aligns with NUREG-0800 (ref. [96]), which is good starting point, and appropriate for Step 2, but as the design progresses, vulnerability of SSCs should be further analysed through suitable DSA.

138. The RP referred to the use of break exclusion zones (BEZ) as an argument when considering pipe rupture in PSR Chapter 15.7 (ref. [19]). The RP used BEZ to define areas where the dynamic effects (like pipe whip and jet impingement) of postulated pipe breaks in high-energy fluid systems could not be tolerated. By defining these zones, designers aim to ensure that essential systems are not concentrated in potential break locations and are adequately shielded or separated from the consequences of a pipe failure, meeting the requirements for protection from environmental and dynamic effects. This approach is based on US NRC guidance NUREG-0800 (ref. [96]).

139. When compared with UK expectations, the RP's approach indicates potential for areas of plant relying on highest integrity claims, which are onerous and should only to be considered when other avenues to improve the resilience of the plant such as adequate protection against the consequences of DEGB have been exhausted and none are deemed to be reasonably practicable. This presents a potential gap versus UK RGP (ONR SAPs and Internal Hazards TAG, refs. [29], [58]).

140. The ONR structural integrity assessor has taken the lead on the assessment of highest reliability components (ref. [108]). There have been several meetings held between ONR, US NRC and the CNSC to understand the different perspectives on the Break Exclusion Zone (BEZ) methodology.

141. In my assessment I have taken cognisance of the technical assessment undertaken by the CNSC of Ontario Power Generation (OPG) submissions of their BEZ approach to the BWRX-300 design.

142. I noted that the Structural Integrity assessor raised an RQ to acquire evidence regarding the final position with regards to BEZ methodology and its alignment with the CNSC's requirements. The RP explained that a final position had not yet been reached, and discussions were ongoing.

143. The RP concluded that given the on-going production and submission of the documents to CNSC and potential iteration of the 'final' position, the resolution of the BEZ topic will be available after GDA Step 2 (RQ-01921, ref. [109]).

144. Given the on-going communication with the CNSC and the RPs recognition of the gaps by the integration of the solutions into their FAP item PSR15.7-63 (ref. [76]), I judge the position as adequate at this stage of the design's development.

145. The RP has identified appropriate initiating events and consequences relating to PPF in PSR Chapter 15.7 (ref. [19]). This methodology satisfied the requirements of the US NRC guidance. The bounding failure of HELB has been considered and the defence in depth strategy gave me confidence that the RP has considered the effect of dynamic and environmental effects on SSCs as a result of a pipework failure. This shows a route to demonstrating **Claim 2.1** (derivation of system and structure functions) and **Claim 2.4** (safety risks reduced to SFAIRP) with fundamental aspects at Step 2. Although further work is required to develop the PPF analysis, I judged this as adequate at this stage of the design development.

146. In considering PPF hazards, the RP demonstrated use and application of national and international standards such as; the SAPs, IAEA SSG-64, NUREG-0800 series, and IEEE/IEC 60780-323-2016 guidance (refs. [29], [66], [96], [107]).This demonstrates alignment with **Claim 2.1.3** (use of

RGP and OPEX to support reducing risks to ALARP) and **Claim 2.4.1** (use of RGP in all topic areas). The FAP process gave me confidence that gaps can be closed in future design development. I judged this as adequate at this stage of the design's development.

4.3.7. Internal Missiles

147. The RP described its general approach and methodology to internal missiles which includes the consideration of turbine generated missiles, rotating plant missiles and non-rotating plant missiles (see Figure 1). The RP identified that missile sources can fail disruptively ejecting fragments which could potentially damage SSCs important to nuclear safety and compromising the delivery of FSFs. The RP also identified chemical and physical explosions could produce missiles which have the potential to result in similar unmitigated consequences.

148. The RP's DiD strategy on the fundamental protection against internal missiles (PSR Chapter 15.7, ref. [19]) is provided by:

- Design measures to prevent internal missile generation.
- Missile-proof structures and SSCs designed to withstand missile impact.
- Provision of barriers and equipment placement. This includes considering equipment orientation to minimise the consequences of missile strike.
- Redundant equipment is physically separated to minimise common-cause failure.

149. I sampled the RP's approach for turbine generated missiles. The RP considered low trajectory missile fragments within a 25 degree angle to the turbine wheel planes, based on US NRC guidance (NUREG-0800, ref. [96]). I sought clarification of how the design and layout was optimised such that the missile impacts are eliminated or minimised (RQ-01710 (ref. [84]). I sampled the reference design GDA layout drawings which show that buildings in the power block (Reactor Building, Control Building, Service Building and Radwaste Building) would be located within areas least susceptible to low trajectory missile strikes. The RP acknowledged the need to carry out sensitivity analysis as part of ongoing development of the case, including missile fragments occurring wider than 25 degrees from the turbine plane of rotation and high trajectory missiles. I am content that the RP has proposed a favourable layout for safety significant buildings in the power block (RQ-01710, ref. [84]), consistent with SSG-64 and SAP ELO.4 (layout) at Step 2 (refs. [66] and [29]).

150. I noted that further work would be performed at later design stages to characterise the different types of internal missiles, determine the potential consequences of any subsequent impacts and implement further safety measures as required.

151. At Step 2, the RP's key protection strategy of 'missile-proof structures and SSCs designed to withstand missile impact' is limited to the Reactor Building. Missile loadings will be characterised as part of future development. I observed that the Reactor Building outer shell is designed to maintain structural integrity from impact hazards, which is considered in the ONR civil engineering assessment (ref. [110]). The presence of an inner steel liner provides DiD from scabbing. At Step 2, the ONR civil engineering assessor is content that that there was sufficient design capacity of the barrier thickness to accommodate turbine missile hazard loadings (RQ-02000, ref. [111]).

152. It is my expectation that the future development of the case incorporates turbine disintegration, learning from UK ABWR and other OPEX, including a wider angle for missile strike zones, plus consideration of high-trajectory missiles. I anticipate this should be addressed by future development of the case, tracked by FAP items. FAP items PSR15.7-66 pertains to turbine missiles, and PSR15.7-67 pertains to other internal missiles (FAP, ref. [77]).

153. To seek further confidence on fundamental layout aspects, I utilised the RP's 3D virtual model walkthrough, to check the turbine orientation and potential impact of turbine missile hazards on nuclear safety significant buildings in the power block (Interaction on 16 April 2025, ref. [76]). With the aid of the 3D model, I reviewed the relative position, separation and segregation arrangements. I saw that safety significant buildings were located within areas least susceptible to low trajectory missile strikes and did not identify any other fundamental facilities of significance to nuclear safety within the low trajectory missile strike zone at this stage.
I considered this as an indication that the RP is adopting appropriate layout design choices, segregation and separation principles to protect against internal missiles, in alignment with SAP ELO.4 (Layout). I judge this as adequate at this stage of the design's development.

4.3.8. Dropped and Impact Loads

154. The RP described its general approach to drops and impact loads which includes the consideration of dropped loads and collapsed structures. The RP identified that this hazard encompassed equipment and items that are dropped, swung, fall or are lowered out of control which could potentially damage SSCs important to nuclear safety. The RP recognised dropped and impacting loads could be caused by failure of cranes, lifting rigs and structural failures (PSR Chapter 15.7, ref. [19]) and also considered dropped fuel load, swing loads and slinging faults. The RP considered temporary structures and committed to consider collapsed structures as part of future development of the case (refer to Figure 1), which I expect to also include the loss of integrity of handling equipment. These matters should be characterised, in line with SAP EHA.1 (Identification and characterisation).

155. In the UK ABWR GDA, a number of dropped loads scenarios associated with spent fuel cask lifting operations proved challenging and the internal hazards assessment queried the design choices made (UK ABWR Step 4 Assessment of Internal Hazards, ref. [112]). I sampled dropped loads hazards to satisfy myself that learning had been incorporated into the BWRX-300 design. I sought clarification on the significant lifts within the Reactor Building, SSC locations and barrier thickness. I also sampled BWRX-300 layout drawings (RQ-01710 and RQ-01712, refs. [84] and [113]).

156. I identified three lifts around the reactor which could result in challenging dropped load hazards, namely:

- Spent fuel cask lifting operations.
- Fine Motion Control Rod Drive (FMCRD) lifting operations.
- Isolation Condenser System (ICS) lifting operations.

157. The heaviest lift involved a spent fuel cask, with technical details (weight and lift height) provided in RQ-01710 and RQ-02026 (refs. [84] and [114]). The RP applied a conservative assumption of a commercial cask and its associated lifting device based on three approved cask vendors. The layout drawing is in RQ-01710 (ref. [84]), the clarification of drop / lift heights is in RQ-01712 (ref. [113]) and the cask vendor assumption is in RQ-02026 (ref. [114]). The hazard magnitude was similar to that in the UK ABWR design.

158. The longest lift involved a FMCRD travelling the distance from grade all the way to the basement (approximately twice the length of the spent fuel lift), with technical details (weight and lift height) provided in RQ-01712 (ref. [113]). This is a long travel distance in terms of what has been assessed as part of ONR's GDA projects to date. However, the weight of this lift is significantly less than the fuel cask, and I judged the impact would be within the bounds of the fuel cask drop. As the design develops, I expect other factors such as load shape, consideration of sharp edges, centre of gravity, load stability to be included in the methodology in line with SAP EHA.1 (Identification and characterisation).

159. The Isolation Condenser lift is carried out when the reactor is fully shutdown and under maintenance. At Step 2, although the GDA scope included all modes of operation, the RP focussed its safety analysis of reactor faults during power operations only. Drops and impacts during all modes of operation will be addressed as part of future development of the case. The RP identified this as part of FAP item PSR15.5-30 (FAP, ref. [77]).

160. With the ICS pools and associated condensers located near the vicinity of the reactor, I considered there was a potential impact on nuclear safety. Therefore, I sought clarification on how challenges to the reactor would be minimised during an Isolation Condenser lift. I sampled the design rules

(PSR Chapter 3, Safety Objectives and Design Rules for SSCs, ref. [3]), which confirmed that lifting equipment would be designed to appropriate standards and safety factors; and the lifting equipment will perform its intended functions in the event of a single component failure. Also, heavy loads will not be lifted over safety classified components and will be managed via the use of safe-load paths and safety interlocks which prevent movement outside of safe load paths. Physical limits and administrative controls provide additional DiD.

161. The RP also provided a walkthrough using their 3D virtual model (Interaction on 16 April 2025 and RQ-01724, refs. [76] [115]), to indicate the lift operations and potential impact of dropped and impact hazards. The Reactor Building contains the reactor, which is effectively surrounded by 6 key compartments. Nuclear safety systems are contained in three alternative compartments (3 divisions), with the three remaining compartments containing minimal systems or systems not fundamental to nuclear safety. If one division is impacted by a dropped or impact load, redundancy is provided by the remaining two divisions. The cask lift and the FMCRD lift are carried out within the same vertical compartment. As mentioned earlier, IAEA SSG-64 (ref. [66]) also requires that the plant can be brought to, and maintained in a safe state in the event of an internal hazard occurrence, including when equipment is unavailable owing to planned maintenance. The RP's high level principles indicate for a most limiting single failure that there are available DLs, including DL4 functions with diverse engineering safety systems to maintain FSFs. At this stage, I did not identify any fundamental systems in the compartment where these lifts are carried out which could impact on nuclear safety (Layouts from RQ-1710, ref. [84], and Nuclear lifts from RQ-02026, ref. [114]).

162. From a conceptual level at Step 2, the ONR civil engineering assessor was content that there was sufficient design capacity of the Reactor Building floors and barriers to accommodate potential dropped and impact hazard loadings (ONR civil engineering assessment, ref. [110]).

163. The RP advised the development of the hazard methodology was ongoing, with FAP item PSR15.7-69 used to track progress (FAP, ref. [77]). It is my expectation that the RP incorporates the learning on the drop and impact loads, and spent fuel route from UK ABWR in the development of the case. The RP identified the 'focus on dropped loads in other GDA processes,' and acknowledged the requirement to address Lessons Learned (FAP item PSR15.7-69, ref. [77]). In sampling the above three lifting scenarios, I considered this as an indication that the RP is adopting appropriate layout design choices, segregation and separation principles to protect against drop and impact hazards, and there is adequate design capacity should a heavy load impact on the compartment, in line with SAPs ELO.4 (Layout) and EKP.2 (Fault Tolerance) (ref. [29]). I judge this as adequate at this stage of the design's development.

4.3.9. Combined Hazards

164. Combined hazards can pose significant challenge to nuclear reactor designs. During the UK ABWR GDA, this was a significant focus, recognising learning from the Fukushima nuclear accident. I sampled the RP's consideration of combined hazards during Step 2 to assure myself that a suitable identification, screening and characterisation methodology has been developed and is informing the design from an early stage.

165. In considering expectation regarding combined hazard analysis, the RP referenced IAEA SSG-64 and SSG-68 (refs. [66] [67]), WENRA Safety Reference Levels (ref. [53]) and ONR SAPs (ref. [29]). The RP identified the key combinations to be as two or more internal hazards; combinations of external and internal hazards; and combinations of two or more external hazards (Combined Hazards Topic Report, ref. [79]).The IAEA, WENRA and ONR guidance expects that combined internal and external hazards are adequately identified and analysed deterministically.

166. The RP acknowledged that identification of hazard combinations has not yet been undertaken at this stage, but would be addressed as safety case and design development progresses. I acknowledge that the RP has captured several forward actions which will need to be addressed in future as the design matures. These include:

- PSR15-3 Deterministic analysis of hazards and combined hazards.
- PSR15.7-71 Approach to deterministic assessments, including inputs to hazard schedule.
- PSR15.7-72 Identification of combination hazards.

167. Learning from OPEX, such as the Fukushima Daiichi accident of 2011 should also be incorporated. IAEA states:

> 'The lessons learned from the Fukushima Daiichi accident have led to the reinforcement of some requirements in SSR-2/1… related to important topics such as the robustness of the design against external natural hazards exceeding those derived from the site hazard evaluation, the independence of different levels of defence in depth, the emergency power supply, the capability for using of non-permanent sources of electric power and coolant and the reliability of the heat transfer to the ultimate heat sink.' (IAEA, SSR-2/1, ref. [63]).

168. Given the Reactor Building construction being below grade, I sampled potential external or internal flooding induced hazard combinations impacting on the battery rooms. The RP indicated that batteries will be located in independent rooms that are both fire and flood protected (PSR Chapter 8, Electrical Power, ref. [8]). The RP stated that this will be considered in further detail as part of the FAP items PSR15-3 and PSR15.8-144 relating to the deterministic analysis and external hazard

initiated event combinations. The RP has captured the learning from Fukushima explicitly under FAP item PSR 15.8-151 (FAP, ref. [77]).

169. I concurred with the ONR external hazards assessor that a meaningful assessment of combined hazards could not be carried out at Step 2. However, as the future project intends to apply principles of protection against combination hazards using guidance such as IAEA SSG-64 (ref. [66]) and WENRA Safety Reference Levels (ref. [54]) and based on the confidence I gained from my sample of individual hazards at Step 2, including application of DiD principles in design, I am content with the identification of the forward actions to address this in the future.

4.3.10. Novel Design Features

170. There are some key differences from the ABWR design in the BWRX-300 that are relevant to my assessment, the elimination of safety relief valves, the integrated reactor double isolation valves and below grade construction. I discuss each in turn below.

Safety Systems - Elimination of safety relief valves

171. The RP's general description document (ref. [78]) stated that safety relief valves have been eliminated from the coolant system in the BWRX-300 design. To relieve excess pressure from the vessel the design relies on the isolation condenser system (ICS).

172. The RP stated that the ICS has been designed to relieve pressure in line with the American standard (ASME) Boiler and Pressure Vessel Code (BPVC), Section III, Class 1 equipment (ref. [98]). The ICS removes decay heat, reduces steam pressure and RPV pressure; this is the RP's key rationale supporting the elimination of safety relief valves in this system (PSR Chapter 6, Engineered Safety Features, ref. [6]). The ICS consists of three independent loops each containing a heat exchanger. I noted an internal hazard event in the heat exchanger system could affect the integrity of the cooling system; however, ICS loops are separated by distance to prevent an incident impacting multiple loops. One loop is required for design basis incidents whereas two loops would be required for LOCA mitigation, hence one additional loop is designed in for redundancy purposes, the safety function should not be challenged.

173. The removal of thermal heat condenses steam by heating/evaporating water in the ICS pools which are vented to atmosphere. The ICS pools have sufficient inventory to provide adequate decay heat removal and fuel cooling for seven days following the loss of AC power. I noted from the 3D walkthrough of the reactor building that the pipework relating to the steam system was not convoluted, therefore reducing the risk of unrevealed cracks, water hammer or pipe whip (RQ-01724, ref. [115] and Interaction on 16 April 2025, ref. [76] respectively).

174. I note that a break in the pipework within the ICS system can quickly be isolated by the reactor isolation valves, these valves are situated at the section of pipework immediately penetrating from the RPV. This system can isolate or reduce the flow of coolant dependent on the function of the pipework. This is discussed further below in paragraph 175 (PSR Chapter 1, ref. [2]) and the general description document (ref. [78]). I note that further analysis will be required as the design develops to support reliability claims relating to the ICS. This has been discussed with the Fault Studies assessor and is addressed in the Fault Studies assessment (ref. [116]).

175. At this stage of the design's development, I consider that adequate measures to address internal hazards have been identified particularly in relation to the redundancy and separation of ICS trains. The RP's approach has given me confidence that internal hazards will be managed as the design develops.

Safety Systems - Reactor Isolation Valves

176. The RP's approach to loss of cooling accidents (LOCAs) and passive fuel cooling following transients is discussed in PSR Chapter 1 (ref. [2]). The BWRX-300 design has integral reactor isolation valves (RIVs) which are designed to isolate or reduce flow in a line following a downstream medium or large pipe break. Two RIVs are installed in series on a flange connection at the closest piping terminal to the RPV.

177. The RP claimed that RIVs have a SC1 classification, designed with mechanical redundancy and have diverse control and instrumentation. Each RIV can act independently in the event of a transient and close within ten seconds, achieved by SC1 battery backed AC power. I noted that during the 3D walkdown that there are two battery rooms, each is designed above assumed flooding levels. Furthermore, battery rooms are segregated from each other, which ensures redundancy in the event of a loss of a division following a transient. This would ensure power remains available to fulfil safety functions (RQ-01724, ref. [115] and Interaction on 16 April 2025, ref. [76] respectively). All RIVs fail-safe in a closed position upon loss of power, except on the ICS steam supply and condensate return lines. These lines perform safety functions during transient and LOCA events to aid cooling, but flow may be limited by partial closure of the RIVs. As the design develops the reliability of the RIVs will require assessment. This should include loss of power as a credible scenario, as this may limit the RIVs functionality, albeit battery back-up is available. The consideration of all operational modes is included under FAP item PSR 15.5-30 (FAP, ref. [77]).

178. PSR Chapter 15.2 states, 'a break between the RPV and the RIVs is not evaluated as a Design Basis Analysis fault. Breaks inside containment are postulated to occur at any arbitrary location between the outer RIV, or the flow limiter for Main Steam pipes, and the containment boundary.' (PSR

Chapter 15.2, ref. [14]). If a break between the RPV and the nearest RIV were to occur, an un-isolatable LOCA could result. This presents a challenge to safety of the design. This was discussed with the Fault Studies assessor and further analysis is required as the design develops to underpin design features and claims associated with the RIVs to ensure alignment with RGP as set down in ONRs SAPs EHA.3 (Design basis events) and EHA.18 (Beyond design basis events). The ONR Fault Studies assessor has raised a regulatory observation (RO-BWRX300-004, ref. [117]) to seek further clarity on how large un-isolatable pipe breaks will be addressed in the future (Fault Studies assessment report, ref. [116]).

Below Grade Construction

179. The novel aspect of the additional reactor void volume could act as a fluid catcher. This is important, considering learning from the Fukushima Daiichi accident, which highlighted the requirement to consider external or internal flooding that could induce hazard combinations. This is expected to be considered in further detail as part of the FAP items PSR15-3 and PSR15.8-144 (FAP, ref. [77]). This was discussed in section 4.3.9.

180. Learning from UK ABWR also highlighted the requirement for hydrogen management. With the reactor building primarily below grade, vapour cloud explosions and battery induced explosions are expected to be covered as part of FAP item PSR15.7-68 (ref. [77]). This was discussed in section 4.3.4.

4.3.11. Segregation, Separation and Redundancy against Internal Hazards

181. The RP outlined design rules incorporating segregation, separation and redundancy principles (Safety Objectives and Design Rules, ref. [3]). Component details (including penetrations across nuclear safety significant barriers) were at a high level at this stage. At this stage of design development, I focussed my assessment primarily on segregation and separation aspects; considering the application of high-level principles across each individual hazard covered in the PSR.

182. I raised a RQ to gain an understanding in the robustness of overall design through optimisation of layout, and provision of barriers. I sampled the GDA layouts and the 3D model (RQ-01710, ref. [84] and Interaction on 16 April 2025, ref. [76] respectively).

183. The layouts provided confidence that fundamental segregation and separation principles[3] have been included early in the design development of the power block. The design incorporated three divisions that are

[3]The RP used the terms, “physical separation” and “spatial separation.” ONR uses “segregation” and “separation” respectively. For the purposes of this report, ONR terminology is used.

physically and electrically independent of each other, separated by hazard barriers. This ensures there is redundancy should one division be lost to an internal hazard event. The primary containment has multiple safety trains which apply the separation approach. In these areas, the RP acknowledged the need for spatial separation and segregation of supporting equipment and components to protect against all relevant internal hazards (Safety Objectives and Design Rules, ref. [3]).

184. The RP's proposed methodologies identified several measures which I consider demonstrated defence in depth. In my sampling of measures to protect against pipe break events and the subsequent consequences:

- Barriers will be provided to prevent whipping pipe or jet impingement from impacting SSCs in separate divisions.
- Barriers, bunds and flood diverting arrangements such as walls, dikes, trenches etc. can be used to divert flood water away from vulnerable SSCs both in the vertical and horizontal planes and there is some evidence as to the withstand of hydrostatic loads placed on them. This was presented in the RP's IEFHDC (RQ-01771, ref. [94]).
- To minimise damage to vulnerable SSCs by dynamic effects, the RP will either relocate SCCs, provide protective barriers, jet shields or qualify of SSCs to operate following a dynamic event such as pipe whip.
- To minimise damage from environmental effects SSCs are either be relocated, placed at elevated positions above flooding heights, placed in watertight housings, and/or be qualified for submersion.

185. Decay heat removal the design incorporated three cooling trains on the ICS. One is required for 'anticipated operational occurrences'; two trains are required for LOCA mitigation with the third provided for redundancy. Although there are barriers segregating each pool, each train is spatially separated by virtue of the connections to the RPV and through venting to the containment. Through my sampling of the case, I did not identify an internal hazard that would challenge multiple trains of the ICS and would not be surmountable through design development (e.g. protection from dynamic effects, environmental qualification etc). I therefore conclude that the case aligns with my expectations and ONR SAP EKP.3 (defence in depth)

186. I also took cognisance of layout considerations sampled by the civil engineering (RQ-01724, ref. [115]) and mechanical engineering (RQ-02026, ref. [114]) topic areas.

187. With additional responses to specific information requests and the 3D model walkthrough, I am content that the RP has applied their design rules discussed above to consider the fundamental aspects of segregation and

separation in minimising the impact of internal hazards to the overall layout design at GDA Step 2, in line with SAP ELO.4 (Layout).

4.3.12. ALARP

188. In terms of reducing risks to ‘so far as is reasonably practicable’ or more commonly known as ALARP;

- The RP has identified and presented its approach to hazard identification and analysis methodology and identified gaps with RGP.
- The RP has implemented appropriate consideration of the hierarchy of safety measures set out in the Engineering Key Principles (EKP.1 to 5) and supporting guidance, with requirements set out to eliminate or minimise the impact of hazards, primarily using segregation.

189. The design and safety case are in development. Therefore, consideration of internal hazards and combination of hazards are not exhaustive at this stage. Further work is required by the RP to align its internal hazards analysis and overall case to demonstrate the risks to the plant, operators and public will be reduced to ALARP in a future version of the safety case.

190. However, at Step 2 the RP presented the following to provide confidence that an ALARP demonstration can be made in the future:

- BWRX-300 design evolution summarising safety design changes (PSR Chapter 27, ALARP, ref. [25]).
- Methodologies for assessment of key internal hazards (PSR Chapter 15.7, Internal Hazards, ref. [19]).
- The development of design rules combined with the general arrangement and building section drawings used to demonstrate fundamental segregation and separation principles applied at Step 2 (Safety Objectives and Design Rules, ref. [3] and RQ-01710, ref. [84]).
- A 3D model walkthrough to provide confidence of the arguments used to support **Claim 2.1** (derivation of system and structure functions) **and Claim 2.4** (safety risks reduced to SFAIRP) (Interaction on 16 April 2025, ref. [76]).
- High level information relating to RGP and OPEX used in the safety case, underpinning **Claim 2.4.1** (use of RGP in all topic areas) and **Claim 2.4.2** (use of OPEX in all topic areas)**.** Details of these were discussed in previous sections of my report.

191. I have assurance that DiD principles have been applied to the BWRX-300 design. For example, the RP’s safety case highlighted DiD elements which centred on preventing, detecting, suppressing fires, and limiting their

effects. Similarly measures to reduce the risk of flood propagation were also identified (discussed above).

192. Notwithstanding the ALARP case yet to be made, I am content that the RP has made sufficient progress in line with RGP, with the actions identified for further development of the case to meet UK regulatory requirements. Subject to FAP items being addressed in the future by continuing the existing approach (RQ-01711, ref. [83]), taking account of further design changes, technological advances and latest RGP, I have adequate assurance that risks from internal hazards can be reduced to ALARP in the future.

5. Conclusions

193. This report presents the Step 2 Internal Hazards 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. [33]), against the expectations of ONR's SAPs, Internal Hazards TAG and IAEA SSG-64 (refs. [29], [58], [66]). The assessment considered key hazards, fundamental layout considerations and novel aspects.

194. The RP provided indicative strategies, methodologies and approaches in the consideration of internal hazards which form the building blocks for the design and safety cases. Although at varying levels of maturity, I judged that key hazards were considered at GDA Step 2.

195. With the currently available information from the safety case, responses to specific information requests and the 3D model walkthroughs, I was content with the development of the fundamental adequacy of the design and safety case of internal hazards aspects for GDA Step 2.

196. These internal hazards, other hazard combinations, along with supporting substantiation and evidence may be sampled further in the future as more detailed design and safety case information becomes available and/or if there are design changes which may have an impact on the design robustness against internal hazards and hazard combinations.

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

- Key hazards were identified at a high level, supported by methodologies fulfilling a GDA Step 2 fundamental assessment.
- Although, hazard identification and analysis were not exhaustive at this stage, fundamental principles have been included early in design development. The fundamental layout considerations adopt Engineering Key Principles, adequately considering segregation and separation. At GDA Step 2, this provides assurance of the design's intent to mitigate internal hazards risks. Further work in future will be required to align internal hazards analysis and the overall case to demonstrate the risks to the plant, operators and public will be reduced to so far as is reasonably practicable (SFAIRP).
- My sampling of novel aspects in safety systems and reactor isolation valves did not identify any significant safety shortfalls in the protection against internal hazards.
- The rationale for the design and the basis for any requirements was clearly set out, demonstrating the RP's use of operational experience and international standards to safety risk reduction. By continuing with

this approach, addressing Forward Action Plan (FAP) items, assessing impact of future design changes/technological advances and incorporating the RGP at the time, organisations taking this project forward should have a future ability to demonstrate the design will reduce risks to SFAIRP.

198. Overall, based on my assessment, I have not identified any fundamental safety shortfalls that would 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 to substantiate the claims and proposals made in the GDA Step 2 submissions.

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Appendix 1 – Relevant SAPs considered during the assessment

SAP reference	SAP title
EDR.2	Engineering principles: design for reliability Redundancy, diversity and segregation.
EHA.1	Engineering principles: external and internal hazards. Identification and Characterisation.
EHA.3	Engineering principles: external and internal hazards. Design basis events.
EHA.6	Engineering principles: external and internal hazards. Design basis event operating states.
EHA.7	Engineering principles: external and internal hazards. ‘Cliff-edge’ effects.
EHA.12	Engineering principles: external and internal hazards. Flooding.
EHA.14	Engineering principles: external and internal hazards. Fire, explosion, missiles, toxic gases etc – sources of harm.
EHA.16	Engineering principles: external and internal hazards. Fire detection and fighting.
EHA.18	Engineering principles: Beyond design basis events.
EKP.1	Engineering principles: key principles. Inherent Safety.
EKP.2	Engineering principles: key principles. Fault tolerance.
EKP.3	Engineering principles: key principles. Defence in depth.
EKP.4	Engineering principles: key principles. Safety function.
EKP.5	Engineering principles: key principles. Safety measures.
ELO.4	Engineering principles: layout. Minimisation of the effects of incidents.
NT.2	Numerical targets and legal limits: Time at risk
SC.4	The regulatory assessment of safety cases. Safety case characteristics.

Appendix 2 – Regulatory Queries used in my assessment

RQ#	Title	Lead Specialism
RQ-01710	Request for General Arrangement Drawings	Internal Hazards
RQ-01711	Clarification of Hazards FAP Commitments	Internal Hazards
RQ-01712	Clarification of Lifting Height / Drop Impact	Internal Hazards
RQ-01724	Layout and Spatial Provision	Civil Engineering
RQ-01771	Clarification of Internal Flooding Methodology (GOTHIC model)	Internal Hazards
RQ-01830	Clarification of Pipe Rupture Hazard Analysis Criteria	Internal Hazards
RQ-01933	Clarification on Internal Fire / Internal Missile Methodologies	Internal Hazards
RQ-02000	Turbine Disintegration: Reactor Building Roof	Civil Engineering
RQ-02026	BWRX-300 3D Model, Layout considerations and Nuclear Lifting	Mechanical Engineering
RQ-02085	Clarification on approach to deterministic assessment	Internal Hazards