ONR Assessment Report

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

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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 probabilistic safety analysis (PSA) 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 (ref. [1]) and supporting documentation.

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

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

I targeted the following aspects in my assessment of the BWRX-300 SSSE:

- PSR Chapter 15.6 – Probabilistic Safety Assessment
- PSR Chapter 15.9 – Summary of Results of the Safety Analyses
- Passive Safety Claims
- Redundancy of Safety Systems
- PSA Modelling and Results

Based upon my assessment, I have concluded the following:

- The PSA models and reports that were submitted are of high quality and demonstrate that the risk arising from the design of the BWRX-300 is understood, and that PSA has been used to inform the design;

- As the design is still in an early stage, there are certain areas of the PSA that are not mature, namely human reliability analysis and C&I. I have identified these areas as shortfalls against IAEA SSG-3 (ref. [2]) and TAG-030 (ref. [3]) and the RP has agreed and identified these on a forward action plan. I am content with these actions and consider that it is appropriate to bring this part of the model up to maturity when the design has also matured;

- I have assessed the areas of the design which are represented in the PSA and found that the BWRX-300 design has appropriate levels of redundancy and defence in depth with respect to the PSA modelling; and,

- I have assessed the unique design aspect of BWR technology whereby radioactive steam exists outside the containment vessel, from a PSA perspective, and find that this aspect is acceptable. I have used learning from the advanced boiling water reactor (ABWR) GDA and am content that the likelihood of this type of an accident associated with active steam outside containment is expected to be acceptably low.

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

List of abbreviations

ALARP	As low as reasonably practicable
ABWR	Advanced Boiling Water Reactor
AOT	Allowable Outage Time
ATWS	Anticipated Transient Without SCRAM
BHEP	Basic Human Error Probability
BIS	Boron Injection System
BWR	Boiling Water Reactor
C&I	Control and Instrumentation
CAE	Claim, Argument and Evidence
CAFTA	Computer Aided Fault Tree Analysis
CDF	Core Damage Frequency
CNSC	Canadian Nuclear Safety Commission
DAC	Design Acceptance Confirmation
EA	Environment Agency
ESBWR	Economic Simplified Boiling Water Reactor
FAP	Forward Action Plan
GB	Great Britain
GDA	Generic Design Assessment
GVHA	GE Vernova Hitachi Nuclear Energy Americas LLC
HF	Human Factors
HVAC	Heating, Ventilation and Air Conditioning
I&C	Instrumentation and Control
IAEA	International Atomic Energy Agency
IEF	Initiating Event Frequency
ICS	Isolation Condenser System
LOCA	Loss of Coolant Accident
LRF	Large Release Frequency
MDSL	Master Document Submission List
MW	Megawatt
NPP	Nuclear Power Plant
ONR	Office for Nuclear Regulation
OPEX	Operational Experience
PA	Protected Area
PCCS	Passive Containment Cooling System
PIE	Postulated Initiating Event
POS	Plant Operating State
PRA	Probabilistic Risk Assessment
PSA	Probabilistic Safety Assessment
PSAR	Preliminary Safety Analysis Report
PSR	Preliminary Safety Report
RP	Requesting Party
RPV	Reactor Pressure Vessel
RQ	Regulatory Query
SBWR	Simplified Boiling Water Reactor
SC	Safety Classification

SSSE	Safety, Security, Safeguards and Environment Cases
SSCs	Structures, Systems and Components
SAP	Safety Assessment Principle(s)
SSC	Structure, System and Component
TAG	Technical Assessment Guide(s) (ONR)
TSC	Technical Support Contractor
TSM	Technical Specification Monitor
UK	United Kingdom
US	United States of America
USNRC	United States Nuclear Regulatory Commission
V&V	Verification and Validation
WENRA	Western European Nuclear Regulators' Association

Contents

1. Introduction

1. This report presents the outcome of my PSA assessment of the BWRX-300 design as part of Step 2 of the ONR GDA. My assessment is based upon the information presented in the Safety, SSSE head document (ref. [4]), specifically chapters (refs. [5], [6], [7], [8], [9], [10], [11], [12], [13], [14], [15], [16], [17], [18], [19], [20],) the associated revision of the Design Reference Report (DRR) (ref. [1]) and supporting documentation.

2. Assessment was undertaken in accordance with the requirements of ONR's Management System and follows ONR's guidance on the mechanics of assessment, NS-TAST-GD-096 (ref. [21]) and ONR's risk informed, targeted engagements (RITE) guidance (ref. [22]). The ONR Safety Assessment Principles (SAPs) (ref. [23]), together with supporting Technical Assessment Guides (TAGs) (ref. [24]), 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. [25])).

1.1. Background

4. ONR's GDA process (ref. [26]) calls for an assessment of the RP's 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 Americas LLC, United Kingdom (UK) Branch, is the RP. GE Vernova Hitachi Nuclear Energy Americas LLC (GVHA) is a provider of advanced reactors and nuclear services and is the designer of the BWRX-300. GVHA is headquartered in Wilmington, North Carolina, United States of America (US).

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

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

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

8. Step 1 is the preparatory part of the design assessment process and is 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 will complete 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 a site-specific basis and with a future licensee.

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

- The fundamental adequacy of the design and safety, security and safeguards cases; and

- The suitability of the methodologies, approaches, codes, standards and philosophies which form the building blocks for the design and cases.

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

12. Prior to the start of Step 2 I prepared a detailed Assessment Plan for PSA (ref. [27]). 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. [28]) and published on the regulators’ website.

1.2. Scope

14. The assessment documented in this report is based upon the SSSE for the BWRX-300 (refs. [4], [5], [6], [7], [8], [9], [10], [11], [29], [12], [13], [14], [15], [16], [17], [18], [19], [20], [1], [21], [22], [23], [24], [25], [26], [29] [30], [31], [32], [33], [34], [35], [36], [37], [19], [20], [38]).

15. The RP’s GDA scope has been agreed between the regulators and the RP during Step 1. This is documented in an overall Scope of Generic Design Assessment report (ref. [39]). This is further supported by its DRR (ref. [1]) and the MDSL (ref. [40]). The GDA scope report documents the submissions which were provided in each topic area during Step 2 and provides a brief overview of the physical and functional scope of the nuclear power plant (NPP) that is proposed for consideration in the GDA. The 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. [21]) and ONR's guidance on Risk Informed, Targeted Engagements (ref. [22]).

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:

- Emergency Procedures
- As low as reasonably practicable (ALARP) Demonstration
- Fault evaluation of severe accidents
- Site-specific provisions which are not inherent to the design
- Level 3 PSA – the RP did not submit a Level 3 PSA during GDA.

20. My assessment has considered the following aspects:

- The PSA models and reports submitted during GDA to ONR
- PSR Chapter 15.6 (ref. [15])
- PSR Chapter 15.9 (ref. [16])
- Passive Safety Claims
- Redundancy of Safety Systems

2. Assessment Standards and Interfaces

21. Our 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 SAPs (ref. [23]) 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. [41]) and nuclear security series (ref. [42]) 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. [43]), which represent good practices for existing nuclear power plants, and Safety Objectives for new reactors (ref. [44]).

26. The relevant SAPs, IAEA standards and WENRA reference levels are embodied and expanded on in the TAGs (ref. [24]). The TAGs provide the principal means for assessing the PSA 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:

- FA.10 – Need for PSA
- FA.12 – Scope and Extent of PSA
- FA.13 – Adequate Representation of PSA
- FA. 14 – Use of PSA

29. These five SAPs are the PSA SAPs and reflect the expectations that ONR has related with PSAs that are submitted for assessment.

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

2.1.2. Technical Assessment Guides (TAGs)

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

- NS-TAST-GD-005, ALARP (ref. [45])
- NS-TAST-GD-030, PSA (ref. [3])
- NS-TAST-GD-042, Validation of Computer Codes and Calculation Methods (ref. [46])
- NS-TAST-GD-116, Use of Probabilistic Safety Analysis and Probabilistic Insights (ref. [47])

2.1.3. National and International Standards and Guidance

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

- IAEA SSG-3 - Development and Application of Level 1 Probabilistic Safety Assessment for Nuclear Power Plants (ref. [2])
- NUREG/CR-6268 - Common-Cause Failure Database and Analysis System: Event Data Collection, Classification and Coding (ref. [48])
- NUREG/CR-4772 - Accident Sequence Evaluation Program Human Reliability Analysis Procedure, USRNC (ref. [49])
- NUREG/CR-1278 - Handbook of Human Reliability Analysis with Emphasis on Nuclear Power Plant Applications Final Report (ref. [50])

2.2. Integration with other Assessment Topics

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

34. The key interactions with other topic areas were:

- Specialist ONR inspectors in Fault Analysis took the lead in assessment of the severe accident safety case, while I assessed the probabilistic results in the Level 2 PSA.

- Specialist ONR inspectors in Civil Engineering took the lead assessing the design of the containment structure, and I assessed the Level 2 PSA results that were derived from those analyses.

- Specialist inspectors in Human Factors (HF) took the lead regarding the RP's suitability and sufficiency of human actions in the design basis analysis part of the safety case, while I assessed the HRA calculations and HRA modelling used in the internal events Level 1 PSA.

- I took the lead assessing the modelling of software and instrumentation and control (I&C) in the internal events Level 1 PSA, while the ONR control and instrumentation (C&I)[1] specialist inspector supported my assessment.

- Specialist ONR inspectors in Internal Hazards took the lead assessing the deterministic parts of the internal hazards safety case, while I assessed the results of the internal hazards PSA.

- Specialist inspectors in External Hazards took the lead assessing the deterministic parts of the external hazards safety case, while I assessed the results of the external hazards PSA

2.3. Use of technical support contractors

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

[1] In this report, when referring to the ONR specialism of Control & Instrumentation, the acronym C&I has been used, whereas when referring to the BWRX-300 instrumentation & control design, the acronym I&C has been used.

3. Requesting Party's submission

36. The RP submitted the SSSE at the start of Step 2 in four volumes that integrate environmental protection, safety, security, and safeguards. This was accompanied by a head document (ref. [4]), which presents the integrated GDA environmental, safety, security, and safeguards case for the BWRX-300 design.

37. 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.

38. This section presents a summary of the RP's safety case for PSA. It also identifies the documents submitted by the RP which have formed the basis of my Step 2 assessment of the BWRX-300 design.

3.1. Summary of the BWRX-300 Design

39. The BWRX-300 is a single unit, direct-cycle, natural circulation, boiling water reactor with a power of ~870 megawatt (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.

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

41. The BWRX-300 reactor core houses 240 fuel assemblies and 57 control rods inside a steel reactor pressure vessel (RPV). It uses fuel assemblies (GNF2) that are already currently widely used globally (see Chapter 4 of the PSR - ref. [51]).

42. 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.

3.2. BWRX-300 Case Approach and Structure

43. The RP has submitted information on its strategy and intentions regarding the development of the SSSE (refs [52], [53], [54], [55]). 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 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. [56]), supplemented to include UK specific chapters such as Structural Integrity and Chemistry. The RP has also provided a chapter on ALARP, which is applicable to all safety chapters. The RP has stated that the design and analysis referenced in the PSR is consistent with the March 2024 Preliminary Safety Analysis Report (PSAR) submitted to the US Nuclear Regulatory Commission (US NRC). The Security Assessment and PER are for the same March 2024 design but have more limited links to any US or Canadian submissions.

3.3. Summary of the RP's case for PSA

46. The aspects covered by the BWRX-300 safety case in the area of PSA is summarised as follows:

47. The RP has set out two quantitative PSA safety goals in Chapter 15.6 of the PSR (ref. [15]):

- Core damage frequency (CDF) <1x10-6 /year
- Large release frequency (LRF) <1x10-7 /year

48. BWRX-300 PSA models were submitted to ONR during Step 2.

49. To outline the PSA methodologies used for the BWRX-300, the RP has submitted a methodology document (ref. [57]). The RP claims in the methodology document (ref. [57]) that the BWRX-300 PSA methodology adheres to the ASME/ANS Level 1 PSA Standard (ref. [58]).

50. The PSR, Chapter 15.6 (ref. [15]) presents the objectives for the PSA and demonstrates how the PSA models and reports meet these objectives. The internal events Level 1 PSA analysed the design and operation of the plant to identify sequences of events that can lead to core damage, and the CDF is calculated. This includes:

- Internal events Level 1 PSA at full power and low power and shutdown
- Internal fire Level 1 at full power
- Internal flooding Level 1 PSA at full power
- Seismic PSA Level 1 PSA at full power
- High Wind Level 1 PSA at full power
- Spent fuel pool PSA
- Fuel and heavy load movements PSA

51. The Level 2 PSA calculated the LRF of radioactive releases from the reactor due to accidents in the reactor. This includes:

- Internal events Level 2 PSA at full power
- Internal fire Level 2 PSA at full power
- High wind Level 2 PSA at full power

52. The PSA scope includes all plant operating modes including full power, low power and shutdown. The BWRX-300 design operating modes are grouped for the PSA on the basis of common plant response as considered in the PSA and are described as follows in the PSR Chapter 15.6:

- Mode 1: Power operation, modelled in the full power internal events PSA
- Mode 2: Startup, bounded by the full power internal events PSA
- Mode 3: Hot shutdown, modelled in the low power shutdown PSA
- Mode 4: Stable shutdown, modelled in the low power shutdown PSA
- Mode 5: Cold shutdown, modelled in the low power shutdown PSA
- Mode 6a: Refuelling (reactor cavity drained): modelled in the low power shutdown PSA
- Mode 6b: Refuelling (reactor flooded to normal with the fuel pool gate installed): modelled in the low power shutdown PSA
- Mode 6c: Refuelling (reactor flooded to normal with fuel pool gate removed): modelled in the low power shutdown PSA

53. The RP also produced a PSA Summary Report (ref. [28]) to provide requested information that was not included in the PSR chapters relating to PSA.

54. The BWRX-300 PSA results are reported in Chapter 15.9 (ref. [16]) of the PSR. It is noted that the results are all lower than the two quantitative PSA safety goals listed in Chapter 15.6 (ref. [15]) of the PSR.

3.3.1. ALARP by Design

55. The appendix of Chapter 15.6 (ref. [15]) states that nuclear safety risks cannot be demonstrated to have been reduced ALARP within the scope of a two-step GDA, but rather points to future work that will be required to 'contribute to the development of a future ALARP statement'. An appendix table lists the forward action plans (FAPs) that were known at the time the PSR was written.

56. The RP states that the PSA results presented in the PSR Chapter 15.6 (ref. [15]) meet the BWRX-300 numerical safety objectives, set out in Chapter 15.6. It also states that scope of the PSA is such that all risk important SSCs are included. It is on this basis that the RP claims that risks have been reduced so far as is reasonably practicable and my assessment in this report judges the basis of this argument.

3.4. Basis of assessment: RP's documentation

57. The principal documents that have formed the basis of my PSA assessment of the SSSE are:

- PSR Chapter 15.6, PSA, which provides a summary of the suite of PSA models (ref. [15])
- BWRX-300 Probabilistic Safety Assessment Summary Report for UK GDA Review, which is a report that provides further information on the PSA scope, inputs and assumptions, screening tasks, results and design insights for the suite of PSA models submitted for GDA (ref. [28])
- BWRX-300 UK GDA Probabilistic Safety Assessment Methodology, which provides further information regarding the methods used in each of the various PSA models (ref. [57])
- BWRX-300 PSA Model Results and Insights, Rev. D (ref. [59])
- BWRX-300 PRA Initiating Event Analysis, Rev. E (ref. [60])
- BWRX-300 PRA RPV Rupture Frequency Study (ref. [61])
- BWRX-300 UK Generic Design Assessment (GDA) Probabilistic Safety Assessment (PSA) Project Plan (ref. [62])
- PSA Step 2 GDA Assessment Plan (ref. [63]) which outlines the approach to the assessment of the PSA.

3.5. Design Maturity

58. My assessment is based on revision 3 of the DRR (ref. [1]). The design reference report presents the baseline design for GDA Step 2, outlining the physical system descriptions and requirements that form the design at that point in time.

59. 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 PA access building.

60. The GDA Scope Report (ref. [39]) 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).

61. In the design reference (revision 3), SSCs in the power block are stated to be at baseline 1. Baseline 1 is defined as:

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

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

4. ONR assessment

4.1. Assessment strategy

63. 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 the ONR sub-specialism of PSA as described in sections 1 and 3 of this report. My assessment strategy is set out in this section and defines how I have chosen which matters to target for assessment. My assessment is consistent with the delivery strategy for the BWRX-300 GDA (ref. [64]) and my plan for Step 2 of the GDA (ref. [63]).

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

65. Whilst there is no operating BWR plant in the UK, ONR has previously performed a four-step GDA on the Hitachi-GE UK ABWR (for the ABWR PSA Step 4 report see ref. [65]). I have taken learning from this previous activity, targeting my assessment on those aspects of the BWRX-300 which are novel or specific to this design. I have not looked to reassess inherent aspects of BWR technology which were considered in significant detail for the UK ABWR and judged to be acceptable.

66. I assessed the internal events Level 1 PSA model by sampling the areas of highest risk in the model (these are discussed in detail in later sections of this report). I also sampled the methodology document (ref. [57]) and the results discussed in the PSA summary report (ref. [28]). I also reviewed the results of the other PSAs submitted, but did not sample the other PSA models. Finally, I assessed the following design and safety case areas: defence-in-depth, passive safety systems, containment, redundancy of safety systems and a particular challenge of BWR technology in PSA – active steam being outside containment.

67. I focussed my assessment on the internal events Level 1 PSA because this is the most important model, from which all other models draw heavily. In addition, this PSA is of very broad scope, and thus my sampling assessment naturally runs across a very broad scope of PSA modelling techniques and approaches.

4.2. Assessment Scope

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

submissions that I have sampled, the standards and criteria that I will judge against and how I have interacted with the RP and other assessment Topics.

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

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

71. I did not sample the following parts of the PSA because they significantly share modelling from the internal events Level 1 PSA and they contain significant design assumptions at this early point in the design (however, I have included my judgement on the results and learning of these PSA models later in this report):

- Level 2 PSA
- Low Power Shut Down (LPSD) PSA
- Hazard PSA
- Seismic PSA
- Detailed assessment of the ultimate pressure regulation (UPR) system – this was in my plan for Step 2 but the RP demonstrated that the system is not risk-important in ref. [59] and thus I have removed it from my sample for Step 2
- Practical elimination claims – this was in my plan for Step 2 but I chose to not sample this due to the early nature of PSA modelling. Practical elimination claims are considered more by the fault studies specialist inspector, due to their deterministic claims (see ref. [66])

72. In order to fulfil the aims for the Step 2 assessment of the BWRX-300, I have assessed the following items, which I consider important:

- I have sampled the safety case, internal events Level 1 PSA results, modelling and summary reports and compared my sample with international and UK expectations, as referenced in Section 2.1, to understand the veracity of the PSA modelling;
- I have reviewed the reliability and probabilistic safety claims and arguments in the relevant internal events Level 1 PSA documents;

- In addition to the general PSA assessment, there are several areas of the design that I have also examined in more detail including:
    - defence in depth claims,
    - passive safety systems,
    - Level 2 PSA containment claims,
    - the redundancy of the safety systems,
    - the Level 1 PSA claims on the reactor pressure vessel,
    - the Level 1 PSA claims related to consideration of I&C and software.

4.3. Assessment

73. In the following sections I present the details of my assessment and conclusions of the BWRX-300 PSA.

4.4. Assessment of PSA Approaches and Methodologies

74. The scope of the PSA includes all sources of radioactivity at the facility, including the reactor core, spent fuel building, radioactive waste and new fuel during all plant operating states (POS)s. Fuel handling operations undertaken after spent fuel has been transported out of the fuel building, including work within spent fuel interim storage are outside of the scope for GDA.

75. The RP submitted a single report, BWRX-300 PSA Methodology (ref. [57]) to describe the approaches used for all PSAs submitted for GDA. The BWRX-300 PSA was stated to have been conducted to adhere to ASME/ANS RA-S (ref. [58]) and ASME/ANS-RA-S-1.2 (ref. [67]), the Level 1 and Level 2 PSA Standards for LWR Nuclear Power Plants. In addition, the RP has referenced a number of other reports that were used as methodologies in specific areas of the PSA such as:

- Identification of postulated initiating events (PIEs) - IAEA TECDOC-1804 (ref. [68]);
- Level 1 Low Power and Shutdown Internal Events PSA - ASME/ANS LPSD PSA Standard (ref. [69]);
- Human Reliability Analysis for PSA - SHARP - EPRI-NP-3583 (ref. [70]);
- Spent Fuel Damage PSA - ASME/ANS Standard for LPSD (ref. [69]);

- Internal Fire Hazard PSA - NUREG/CR-6850 (ref. [71]); and
- Seismic PSA – USNRC Regulatory Guide 1.200 (ref. [72])

76. I assessed the methodologies associated with the different PSA models and reports against ONR expectations in IAEA-SSG-3 (ref. [2]), the SAPs and PSA TAG-030 (ref. [3]). Overall, I was content with the approaches and the methodologies above mostly met my expectations for GDA. I found some minor shortfalls, and these are outlined in the relevant assessment sections in this report for each of the PSAs, however the shortfalls were not significant.

77. Overall, I judge that the PSA methodologies and approaches met favourably with my expectations compared with IAEA SSG-3 (ref. [2]), paragraphs 3.10 to 3.11, TAG-030 (ref. [3]) and SAP FA.10.

4.5. PSA Assumptions

78. For the BWRX-300 PSA, the RP explained in one of our Level 4 engagements (ref. [73]) that the project used four different tools for tracking assumptions:

- Design Input Open Item (DIOI) process – a tool/database created to satisfy the requirements of tracking unverified/preliminary design inputs;
- Design Control of PSA Attributes - a specification that defines the requirements for the functions credited in the BWRX-300 PSA supporting licensing activities;
- Model Charge Tracking database - tracks all design and model changes not included in the baseline Scoping PRA model and analysis supporting documentation. Requests for design optioneering and sensitivities are also tracked here for resolution; and,
- PSA Configuration and Model Maintenance - the focus of PSA configuration control is to ensure consistency with the configuration control high level requirement and supporting requirements from the LWR ASME standard. For consideration of plants in development as well as operating, the non-LWR standard was also reviewed to ensure compliance with these configuration control requirements.

79. In addition to the above tools, the PSA Summary Report (ref. [28]) contains a few paragraphs outlining how assumptions were tracked for specific PSA reports.

80. I sampled the assumptions recorded in the PSA report, BWRX-300 PRA RPV Rupture Frequency Study (ref. [61]). The assumption section in this report contained an ‘overall assumption’, and a list of ‘Specific Key Assumptions’ that are used to calculate the RPV rupture frequency. I found

that the assumptions listed in this report were clear and sensible. I did find that there were other assumptions noted in the later text of the report that were not listed in the sub-section on Assumptions. These appeared to be mostly modelling assumptions, rather than design or operational assumptions and in my opinion it is sensible to include these in the paragraphs where the modelling is discussed, rather than in the earlier sub-section on assumptions.

81. Overall, I judge that the requesting party documented assumptions appropriately in the sample that I assessed during GDA.

4.6. Computer Codes and Inputs

82. The RP performed some support analysis using several computer codes and input information. Much of the thermal-hydraulic analysis was performed using ‘MAAP’ and ‘TRACG’. While MAAP is well known to the industry, and ONR from previous GDAs, TRACG is a General Electric proprietary version of the Transient Reactor Analysis Code (TRAC). It is a best estimate code for analysis of BWR transients ranging from simple operational transients to design basis LOCAs, and anticipated transient without SCRAM (ATWS). TRACG is based on a multi-dimensional two-fluid model for the reactor thermal hydraulics and a three-dimensional neutron kinetics model.

83. In the PSA TAG-030 (ref. [3]), ONR’s expectations for computer codes and inputs are that for any codes used, they have been verified, validated or qualified, as appropriate, and that the codes meet ONR quality expectations as outlined in SAPs para 678 (ref. [23]) and TAG-042 (ref. [46]).

84. Assessment of the verification and validation (V&V) of MAAP and TRACG were conducted during the GDA for the ABWR.

85. The ABWR Fault Studies Step 4 AR (ref. [74]) assessed TRACG and concluded: “I judge both the code and its use to be consistent with my expectations (as established by the SAPs) and I have no issues with its appropriateness for the UK ABWR GDA.”

86. The Severe Accidents Step 4 AR (ref. [75]) assessed MAAP and concluded: “I therefore conclude that Hitachi-GE’s use of MAAP meets my GDA expectations for the AV series of SAPs”.

87. As these codes have already been assessed and found appropriate for use in the ABWR GDA, I took credit for this assessment and did not re-assess these codes in the BWRX-300 GDA. I am content that these are reasonable codes to support claims made in the BWRX-300 PSA.

4.7. Level 1 PSA: Initiating Events

88. There are three steps in Initiating Event Analysis for PSA: creating an initial long list of postulated IEs (PIEs), grouping these IEs down to a manageable

number for further analysis, and finally assigning a frequency to each of the IEs to be analysed in the PSA.

89. The RP's approach for identifying initiating events (IEs) is found in BWRX-300 PRA Initiating Events Analysis report (ref. [60]). The RP identified all of the IEs through systematic analytical methods such as hazard and operability (HAZOP) analysis, failure mode and effect analysis (FMEA) and system description documents and an examination of existing lists of IEs that are in the public domain for similar plants. To begin the process of identification and grouping of IEs, the RP used different criteria to make the long list of IEs to be used in the PSA for all operating modes of the plant more manageable. The following sub-sections describe the steps taken to process the IE list and my assessment.

90. The next step was to perform grouping/bounding and screening analysis in order to limit the analysis required for the PSA. The RP used four criteria to group IEs outlined in the Methodology document (ref. [57]). These criteria were:

- Similarity in plant response;
- Similarity in success criteria;
- Similarity in accident progression timing; and
- Similarity in effect on the performance of operators and relevant mitigating systems.

91. This grouping process resulted in a final set of 13 IE groups.

92. Finally, IE frequencies (IEFs) were assigned for those IEs in the final list. The RP assigned a frequency using generic data, and NUREG-6268 (ref. [48]) was the preferred source for most IEs.

93. I found that the RP's approach for deriving and grouping the set of IEs was logical and followed RPG, such as ONR's TAG-030 (ref. [3]).

94. I next sampled the RP's assigning of loss of coolant accident (LOCA) IEFs for my assessment. I chose LOCA IEFs for my sample because they represent the most risk-important set of IEs in the PSA. The RP tested three different methods to assign LOCA IEFs (aside from Excessive LOCA – RPV rupture which was assigned an IEF using a BWRX-300 PRA RPV rupture frequency study (ref. [61]) which I have assessed separately in the next sub-section). The RP's first method was simply noting the generic LOCA frequencies provided in NUREG-6928 (ref. [76]), however the RP decided that this approach was too conservative. The RP then used an alternative method that counted piping segments for various systems, multiplied this by a frequency per piping segment for various systems and then calculated the overall LOCA frequency by aggregating these by size of break. A third

method was also proposed, using an EPRI derived rupture frequency per foot per year, and then counting up all the length of piping of various bore radii through the plant.

95. All of these methods use generic values that include outside-containment pipe failure data, whereas for breaks inside containment, the quality and inspection/maintenance requirements are assumed by the RP to be far better. Thus, the RP assumed that whatever data is used from the three proposed methods, it would be overly conservative. The RP decided to use a scaled version of the third method, by dividing the final figures by 5 to account for the relative amount of piping that is inside containment compared to outside for most plants, and to account for the higher standard of piping quality and inspection requirements.

96. The final figures used for LOCA IE frequencies are clearly explained as to how they were derived. I found the RP's explanations logical and sensible. I judge that the LOCA IEs are consistent with my expectations for PSA compared with IAEA-SSG3 (ref. [2]) paragraphs 5.13 to 5.22 and 5.28 to 5.31, TAG-030 (ref. [3]) and SAPs FA.12.

4.7.1. Spurious Rupture of RPV Initiating Event Frequency Derivation

97. Modern nuclear reactors have attempted to improve design and operations to the point where the very unlikely event of an RPV failure is now a risk important event in PSAs. I sampled the RP's approach in the RPV rupture frequency study (ref. [61]) for deriving the IEF for spurious rupture of the RPV because this basic event is the most risk important potential failure in the internal events Level 1 PSA.

98. The BWRX-300 RPV design is unusual compared with the UK ABWR or other traditional BWR design, because of the design intention to make the at-power circulation be fully passive.

99. RPV rupture is stated by the RP to be an extremely unlikely event, and no such events are reported in the industry (although there have been significant precursor events, such as the Davis-Besse PWR incident). Thus, the RP claims that its work has significant uncertainty.

100. The RP listed several design assumptions, such as RPV design pressure, the lack of axial welds, no nozzles to be allowed in the core region, etc, which the following approach depended on.

101. To assign an RPV failure frequency, the RP reviewed previous industry estimates for both PWR and BWR RPV rupture frequency, focusing on BWRX-300 RPV design potential weaknesses, such as weld locations, proximity of welds to high neutron flux areas and locations with the highest potential stress. This literature survey resulted in an estimated BWR rupture frequency of between $1x10^{-07}$/year and $1x10^{-08}$/year, with outliers at $1x10^{-10}$/year.

102. The RP then reviewed the treatment of RPV rupture frequency in the ESBWR PSA (an early design from which the BWRX-300 is claimed to have partially evolved from). The ESBWR PRA assigned a value of $1x10^{-10}$/year for RPV rupture, claiming lower than normal BWR operating pressure, the positive effect of natural circulation on the RPV, improved materials, the absence of nozzles and welds at the core level, improved in-service inspections and other factors.

103. The RP next reviewed relevant BWRX-300 design features which could either justify a reduction or increase in using a generic RPV rupture frequency from the literature survey activity.

Table 1: BWRX-300 RPV Design Features and Effects on Rupture Frequency

RPV Design Feature	RP Stated Effect on Generic Rupture Frequency
No axial welds	Reduction
No vessel penetrations near core	Reduction
Design pressure 20% higher than current BWRs and operational pressure is consistent with current BWRs, meaning increased margin compared with current BWRs	Reduction
Taller but thinner RPV	No effect (approximately same surface area)
Core area and lower head contain numerous penetrations	Increase in frequency for core area and lower head (as the generic frequency has PWR data, which typically have less penetrations)
Nozzle design -nozzles attached directly to RPV -nozzles attached further away from RPV	Unknown effect – the nozzles and attached isolation valves are novel
RPV structural support	No effect – areas of high stress have had wall thickened

104. The RP examined uncertainty in the RPV rupture frequency study. Even though the world-wide nuclear industry has accrued thousands of reactor-

years of operating experience, there is still insufficient data to validate the very low frequencies estimated above and included in most PSAs. There are two types of uncertainty that are studied for situations such as this: aleatory (randomness) and epistemic (modelling) uncertainty.

105. Modelling uncertainty is associated with a lack of complete knowledge of the phenomena. The BWRX-300 RPV design differs in various aspects from the existing BWR fleet, upon which the current industry rupture frequency estimates are based. It is observed in Table 1 that certain features may justify a reduction in the frequency and some features may justify an increase in the frequency. Then there are some aspects where it is unknown what the effect will be on the frequency because they are novel in the design such as:

- the inclusion of dual isolation valves adjacent to each RPV nozzle;
- support of the RPV from the mid-vessel;
- an assumption to reduce the BWR generic rupture frequency median value by 3 due to the above mentioned BWRX-300 design features; and
- apportioning of RPV rupture frequencies amongst three regions of the vessel.

106. To finally arrive at an estimate for the BWRX-300 RPV rupture frequency, the RP started with the median value of RPV rupture frequency for both PWRs and BWRs from NUREG/CR-6928 (ref. [76]), reduced this by an order of magnitude to isolate for BWRs, then reduced the frequency by a further factor of 3 due to the novel design features described in Table 1, and then assigning a relatively high error factor of 30 to this figure. They then assumed a lognormal distribution for the rupture frequency, using the error factor of 30, and calculated the mean from this. The final RPV rupture frequency was derived in this way.

107. I have compared the approach used to estimate the RPV rupture frequency with TAG-030 (ref. [3]). The RP has included details of the design and relevant equipment. Early information on the design and operation has also been provided. The approach used (adapted literature survey) is a common approach used to estimate difficult figures that do not have sufficient operational experience (OPEX). The documentation of the approach was thorough. The RP also included thorough descriptions of its uncertainty. Reasonable assumptions have been made and a sensible effort to be best-estimate has been applied.

108. In my opinion, the RP has used a balanced well documented approach, and while the RPV rupture frequency is very low, I am content that they have documented and justified the approach and final figure.

109. While assessing the RPV Rupture Frequency Study (ref. [61]) I reviewed the RP's engineering claims on the RPV and associated nozzles, with integrated isolation valves. The early BWRX-300 RPV design has been documented well, both in this report and in Chapter 4 (ref. [51]) of the PSR. A full list of IEs was developed for the RPV and potential weaknesses in the design (such as the interface with the isolation valves and nozzles) has been discussed and accounted for in the PSA. I did not observe any fundamental safety shortfalls in the design of the RPV with respect to PSA.

110. Overall, I compared the IE list development, bounding and screening approach and the IEF methodology with IAEA-SSG3 (ref. [2]) paragraphs 5.13 to 5.22, TAG-030 (ref. [3]) and SAPs FA.12 and FA.13 and found them to be adequate. I judge that the IE methodology is adequate.

4.8. Level 1 PSA – Determination of Success Criteria

111. Determining success criteria for PSA is a critical step after deciding on the list of IEs to be analysed in the PSA. The main objective of success criteria formulation is to determine for a given IE what represents a successful or unsuccessful plant response, and to translate this information into detailed PSA modelling choices. Front-line systems are studied to understand which can provide mitigating functions for an IE and the minimum requirements for fulfilling those functions, for example how many trains of a safety system are required to fulfil its safety mission after an IE has occurred. The RP outlined the approach used for determining the success criteria used in the BWRX-300 PSA in the PSA Methodology (ref. [57]).

112. The RP claims that each success criterion in the BWRX-300 PSA is based on either a plant design parameter or a thermal-hydraulic calculation. The PSA Summary Report (ref. [28]) states that success criteria for the BWRX-300 PSA was developed using MAAP thermal-hydraulic analysis. The MAAP analysis was not included in my scope of assessment for Step 2, but I did review the results of the MAAP sequence in ref. [28]. The results showed that for all IEs included in the BWRX-300 PSA how many trains of the various safety systems would be required (and hence modelled) to avoid a severe accident.

113. I sampled the BWRX-300 PSA model and observed that the success criteria of the isolation condenser system (ICS) system matched the results of the MAAP analyses in ref. [28]. I have listed this in Table 2 below.

Table 2: BWRX-300 ICS Success Criteria for Various IEs

IE	ICS Success Criteria from MAAP (ref. [28])	Observed ICS Modelling in the BWRX-300 PSA Model
General Transient with ICS Pressure Control and RPV Injection	1 ICS train required	At least 1 ICS train available
Excessive LOCA in RPV Steam Space	3 ICS trains required	All three ICS trains must be available
Unisolated MLOCA in CUW supply line	3 ICS trains required	All three ICS trains must be available
SLOCA	1 ICS train required	At least 1 ICS train available

114. I judge that the RP has used codes that are well known in the industry and in my sample, the success criteria results are reasonable. I sampled the PSA modelling for the ICS system in the PSA fault trees and found that the PSA modelling matched the success criteria for the ICS system outlined in (ref. [28]). Overall, the success criteria approach is consistent with IAEA SSG-3 (ref. [2]) paragraphs 5.45 to 5.54, TAG-030 (ref. [3]) and SAPs FA.13 and in my opinion is consistent with my expectations.

4.9. Level 1 PSA – Event Sequence Modelling

115. Event sequence modelling is usually portrayed in event tree development in coordination with success criteria analysis. Event trees start from the IE considered and describe the various possible sequences of success or failures of safety systems designed to stop the accident or mitigate its consequences. Thus, they give the picture of the progression of failures that can lead to severe accidents, and are one of the primary building blocks for a PSA. The resulting core damage sequences can then be coupled with parametric models of severe accident progression and fission product behaviour to describe pant damage states with potential pathways for offsite radiological release.

116. The BWRX-300 event trees are stated by the RP to have been modelled in accordance with guidance outlined in the ASME Level 1 standard (ref. [58]). After an IE, the event trees first consider reactivity control, then fuel cooling, then long term heat removal and then containment integrity safety functions. The event tree accident sequences then conclude in an end state, of which there is a single non-core damage end state, and six different grouped core damage end states.

117. While I did not sample event sequence modelling in detail, I am content with the approach. I reviewed the event sequence modelling at a high level and found that it met my expectations compared to IAEA SSG-3 (ref. [2]) paragraphs 5.59 to 5.63, TAG-030 (ref. [3]) and SAP FA.13.

4.10. Level 1 PSA - Human Reliability Analysis

118. The BWRX-300 design includes provisions for significant human actions that are necessary during accidents for mitigation. In addition, there are some human actions that can result in an initiating event, in effect causing an accident to begin. As PSA is expected to cover all significant sources of radioactivity, all permitted operating states and all relevant initiating faults, human error is expected to be included as human reliability analysis (HRA).

119. The BWRX-300 PSA includes consideration of human errors in the various PSA models. The RP based its PSA approach (ref. [57]) for modelling human error in PSA on the Systematic Human Action Reliability Procedure (SHARP) in ref. [70].

120. The RP has considered three types of human failure events (HFEs):

- Pre-IE HFEs (Type-A);
- Human failure events (HFEs) that lead to IEs (Type-B); and
- Post-IE HFEs (Type-C).

121. For Type-A and Type-C HFEs, the RP has used HRA documents (refs [50], [49], [77]) to calculate the human error probabilities (HEP). These HEPs are then assigned to different basic events in the PSA models.

122. I compared the RP's consideration and modelling of HRA in the PSA with TAG-030 (ref. [3]) and found a few shortfalls. I noted that although the RP considered the three human error types in its methodology, the RP has not modelled Type A or Type B errors in the PSA model (specifically in the model that was submitted during GDA), but only Type-C errors. This represents a gap against SAP FA.12 and FA.13 in that by not considering Type A or Type B errors there is a potential area of risk in the design that is not understood or accounted for.

123. During the course of my assessment, I wrote two regulatory queries (RQ-01964 and RQ-01998) to seek more information on the RP's position on this matter. The RP stated that Type-A errors were initially identified in the early phase of creating the PSA but were generally screened out on the basis of not being credible for the design. The RP stated that a novel system, the Technical Specification Monitor (TSM), detects miscalibration errors and alerts operators before they become an issue. Thus, the RP has not modelled Type-A human errors in the BWRX-300 PSA.

124. I compared the RP’s approach for Type-A human error probability modelling with the ASEP approach (ref. [49]). I expected an approach whereby all identified Type-A human errors are assigned a basic human error probability (BHEP) which can be reduced by ‘recovery factors’ such as having a checker available when the first worker performs a calibration task, or the use of a written checklist that is confirmed by at least two people. However, ref. [49] also allows for ‘optimum conditions’ due to the excellence of recovery factors, in which case a Type-A HEP may be assigned a negligible number. These ‘optimum conditions’ include a compelling signal such as an annunciator in the control room when the task is finished or before power operation is resumed.

125. The RP’s approach, using the TSM appears to be using this ‘optimum condition’ approach by assuming that the TSM would include a compelling signal in the control room in the right conditions. In this case, ref. [49] allows for a negligible HEP for these Type-A errors. However, the design of the TSM is still notional and for all Type-A errors to be assigned a negligible HEP would require the TSM to have the ability to monitor all applicable plant equipment. In addition, it is noted that in the present design, the TSM has been stated to be a Class 3 system and thus not of a very high reliability in itself. If this system should fail, and a Type-A error was committed, the PSA model is not able to include these sources of risk as it is presently modelled.

126. In response to my questions in RQ-01964 and RQ-01998, the RP stated that the HRA will continue to develop as the design develops. More detailed HRA will be performed, and this was included in the FAP item PSR15.4-191.

127. Whilst the RP’s approach does not fully meet my expectations as set out in SAP FA.12 and IAEA SSG-3 paragraphs 5.103 and 5.122, the information shared with me in response to RQ-01964 and RQ-01998 demonstrates to me that the RP has acknowledged that more work on Type-A human error modelling needs to be provided and has captured this as a commitment in FAP item PSR15.4-191.

128. I compared the RP’s approach for Type-B human error probability modelling with the IAEA SSG-3 (ref. [2]) paragraphs 5.105, 5.113, 5.117, and 9.46, and the ASEP approach (ref. [49]). Errors of commission leading to an initiating event are usually modelled explicitly as IEs with an accompanying event tree, and assigned a frequency based on OPEX. Modern reactor designs, such as the BWRX-300, aspire to reduce the number of potential Type-B human errors through design, and thus it is typical to see fewer of these in PSA models for modern plants. In the PSA Methodology (ref. [57]), the RP states that no specific Type B actions are modelled in the PSA. In subsequent discussions the RP stated that Type B human errors are accounted in the derivation of IEFs.

129. It was not obvious how the derivation of IEFs accounted for Type-B human errors and so I addressed this with the RP via RQ-01964. The RP responded

that they have used generic OPEX to quantify IEs and any contribution from human error to IEFs would be included in such OPEX (such as NUREG/CR-6928 (ref. [76])). I reviewed NUREG/CR-6928 (ref. [76]) briefly and noted that there is some indication that operator error has been included as a data source in quantifying all its basic events and IEFs. However, I did not find explicit mention that it should be assumed that Type-B errors should not be modelled due to this potential inclusion.

130. The RP also stated that no Type-B events were identified at this point in the project, but that as more information becomes available, if Type-B events are identified that are not represented by existing IEFs, they will be explicitly included and quantified in the PSA.

131. Whilst the RP's approach does not fully meet my expectations as set out in SAP FA.12 and/or IAEA SSG-3 paragraphs 5.103 and 5.122, the information shared with me in response to RQ-01964 demonstrates to me that the RP has acknowledged that more work on Type-B human error modelling needs to be provided and has captured this as a commitment in FAP item PSR15.4-192.

132. The RP considered Type-C human errors explicitly in the fault trees of the BWRX-300 PSA and used the EPRI HRA calculator which is based on the ASEP method to derive probability numbers. One Type-C human error was found to be risk important – HEP-BORON-INJ, or the failure of an operator to initiate alternate boration when called to do so. I sampled the analysis of this operator action and found it was assigned a typical screening value. It assumes that the reactor is in an emergency situation wherein SCRAM has failed and ATWS has occurred. In this emergency situation, the consequence of failure is core damage. The RP completed importance analysis and found this basic event to be one of the most risk-important basic events in the PSA. I sampled the calculation of the HEP for this human error and found that it was assigned a number. I was not provided with justification for how this figure was attained, however this would be further than expected for Step 2 in GDA. Overall, I am content with how the RP has assigned a value for this human error and calculated risk importance analysis to highlight this human error as particularly risk important.

133. I noted that in the list of Type-C operator errors found in the BWRX-300 PSA Summary Report (ref. [28]), a few of the HEPs were of very low probability (1x10-05). I asked the RP to provide further justification for why these values are appropriate to be used. In the response to RQ-01964, the RP stated that although the PSA documentation does report that these HEPs are of very low probability, in the PSA these HEPs were assigned a more conservative probability, and that the misreporting of the very low probability was an oversight. I reviewed the PSA model and confirmed that the HEPs of very low probability were in fact assigned a less reliable value, as stated by the RP. I am content that the RP has used appropriate PSA methods to calculate Type-C errors and has also recognised that using these methods arrived at HEPs that were optimistic for this stage of the design and thus went further to modify

them conservatively. I judge that the Type-C errors in the PSA are modelled as per typical PSA methods.

134. The BWRX-300 PSA does not contain many operator errors. I have addressed the lack of Type-A and Type-B errors, and the RP has included the requirement to include Type-A and Type B human errors in the FAP to record the issue for further action. Whilst the RP's approach does not fully meet my expectations as set out in SAP FA.12 and/or IAEA SSG-3 (ref. [2]) paragraphs 5.103 and 5.122, the information shared with me in response to RQ-01964 and RQ-01998 demonstrates to me that the RP has acknowledged that more on Type-A and Type-B human error modelling needs to be provided and has captured this as a commitment in FAP item PSR15.4-191. In addition, I compared the RP's consideration of Type-C error modelling against IAEA SSG-3 (ref. [2]) paragraphs 5.106 and 5.19, SAP FA.12 and the PSA TAG-030 (ref. [3]) and am content with the modelling.

4.11. Level 1 PSA – Consideration of I&C

135. I&C, computers and software can be a significant source of risk in the design of a nuclear power plant. As per ref. [3], and IAEA SSG-3 (ref. [2]) paragraph 5.27, I expected that this be considered in the safety case, and explicitly in PSA models. For the BWRX-300 PSA, the RP has considered I&C both at a high level and sometimes in detail.

136. The BWRX-300 digital control and instrumentation system (DCIS) is arranged in three Safety Classified (SC) DCIS segments and a Non-Safety Class segment. The design documentation contained in Chapter 7 of the PSR (ref. [6]), calls the segments Primary Protection System - SC1, Diverse Protection System (DPS) - SC2, Nuclear Controllers - SC3, and Balance of Plant (BOP) Controllers - SCN.

137. I sampled the consideration of the DCIS in the PSA and noted that SC1 and SC3 are only modelled by a simplified single basic event which represents a combined 'supercomponent', with assigned values. SC2 is modelled as an analogue hardware platform, with explicitly modelled circuit failures, pressure sensors/transmitters, level sensor/transmitter and relay failures as well as appropriate CCFs.

138. I noted that the probability of failure on demand (pfd) was within the expected range for SC1, however, the RP has used a lower than expected pfd for the SC3 system. Thus, I addressed this with the RP in RQ-01988 and the RP explained that the pfd assigned for the SC3 supercomponent was based on guidance from NUREG-0463 (ref. [78]). Other reports discussed by the RP in the response to RQ-01988 recommend a slightly higher or slightly lower value for systems containing software, but that the value assigned in the PSA for SC3 was considered by the RP to be best estimate. However, the RP also recognised that there is a need to bring the modelling of both SC1 and SC3

up to modern standards in future versions of the PSA, and this was documented in FAP PSR15.6-59.

139. I am content with the RP's response regarding SC3's value. The value may be slightly more reliable than I expected, but it is recognised that using an overly conservative figure would result in a PSA dominated by SC3, which is also not appropriate.

140. I am content with the RP's preliminary modelling of SC1 as a supercomponent. The pfd meets with my regulatory expectations for a Class 1 system, and the RP also noted that the system will require full modelling when the design is complete. This is included in FAP PSR15.6-59.

141. Whilst the RP's approach does not fully meet my expectations as set out in TAG-030 (ref. [3]), the information shared with me in response to RQ-01998 demonstrates to me that the RP has acknowledged that more SC1 and SC3 design and modelling needs to be provided and has captured this as a commitment in FAP item PSR15.6-59. Overall, I am content with the I&C modelling in the BWRX-300 PSA for the level of maturity in the design.

4.12. Level 1 PSA – Data Analysis

142. PSA Data includes IEFs, individual component failure probabilities, unavailabilities due to test and maintenance and CCFs. ONR SAP FA.13 and the PSA TAG (ref. [3]) outline the expectation that PSA data should be best-estimate as far as possible, and where this is not practicable, conservative assumptions may be used with the sensitivity to the PSA results of these assumptions being established. IAEA SSG-3 (ref. [2]) paragraphs 5.143 to 5.148 and SAP FA.13 describe the preferred quality order of PSA data derivation to be (in descending quality): facility specific data, generic data, expert judgement data.

143. My assessment of the IEFs is presented in this report (section 4.7). My assessment of the RP's approach for assigning individual component failure probabilities and CCFs is presented in the following paragraphs.

144. I reviewed the RP's PSA Methodology report (ref. [57]) to understand how the PSA data was assigned in the models. As the BWRX-300 design contains more reliance on passive systems than traditional BWRs, the RP found that there tended to be less available PSA data that was directly applicable, and so instead used failure data for components that are most similar to those that would be used in a passive plant design. This approach required the RP to adjust generic failure data when necessary after analysing EIMT intervals and environmental factors.

145. The data are based on generic values and, where available, technical specification allowable outage times (AOTs). As mentioned previously, there is limited component OPEX for passive plants, such as the BWRX-300, and

this has been taken account in the RP's associated uncertainty that is assigned with some data.

146. The RP used NUREG-6928 (ref. [76]) for the majority of both standby and running components. When this generic data was unavailable, the RP used previously performed PSA for similar BWR technology. In addition, when necessary, other generic data from refs. [79], [80] and [78] were used.

147. I assessed the RP's approach for assigning failure rate data in the PSA model and found that it met my expectations. As the BWRX-300 is not a plant in operation it would be inappropriate to expect the plant to use actual plant data. The RP has used various generic sources and in the sample, I examined, they have assigned appropriate levels of uncertainty data to account for the fact that this is not actual plant data from a BWRX-300 plant.

148. The RP used the alpha factor approach in calculating common cause failures (CCFs) in the PSA model. This is a commonly used approach for modelling CCFs and it meets my expectation compared with IAEA SSG-3 (ref. [2]) paragraphs 5.93 to 5.96 and the PSA TAG-030 (ref. [3]).

149. I assessed a sample of the most risk important components in the PSA (CCF of control rods failure to insert and failure of the LGA and LGB electrical boards for the UPS to operate) and was able to successfully trace the assigned failure rates, CCF values and uncertainty values for these two basic events in the generic reference data.

150. I am content with the RP's use of data analysis in the PSA. Although IAEA SSG-3 (ref. [2]) paragraphs 5.143 to 5.148 and the PSA TAG-030 (ref. [3]) discusses the expectation that plant data be used before generic data is used, for a first-of-a-kind reactor, it would not be proportionate to expect this data to be used at this stage in GDA without a plant built. See Section 4.23 of this report for my assessment of data assigned to some of the passive systems.

4.13. Level 1 PSA Results, Uncertainty, Quantification and Interpretation

151. The BWRX-300 internal events Level 1 PSA results estimate a core damage frequency (CDF) that is less than the BWRX-300 target reliability goal from Chapter 15.3 of the PSR (ref. [13]) by close to three orders of magnitude. The RP's probabilistic safety goal is for all event sequences, which includes not just internal events, but also hazards and low power/shutdown PSA. Although hazards PSA and low power/shutdown PSA was outside of my sample for Step 2 of the GDA, these are all estimated to have a lower CDF than the internal events Level 1 PSA results (except for seismic Level 1 PSA, which is slightly greater).

152. IAEA-SSG3 (ref. [2]) paragraph 12.2 to 12.7, SAPs FA.13, FA.14 and TAG-30 (ref. [3]) expect that the results of the PSA be used to understand the level of

risk arising from the design (see also TAG-116 on the Use of PSA (ref. [47]). This can be performed by uncertainty analysis, sensitivity analysis and case studies.

153. In the following sub-sections I present my assessment of the use of the results of the PSA against the expectations outlined in the relevant SAPs and the PSA TAG.

154. The RP quantified the internal events Level 1 PSA to provide an estimate of the CDF using applicable accident sequences which resulted in core damage. The RP used a cut-of value of 1x10-15 /year so that minimal cutsets with a frequency of less than 1x10-15 /year are not included in the quantification calculations. The RP found that the results converged after this level of truncation.

155. The method for quantification that the RP used is consistent with my regulatory expectations. The quantification results for the CDF show that the level of risk arising from the design is low, as is expected for a modern NPP design. ONR does not have a CDF target for reactor designs, however when compared with ref. [81], the CDF compares favourably.

156. The RP considered uncertainty in the quantification process. The RP included consideration of uncertainty throughout the PSA model in that each individual basic event would have its own uncertainty, which accompanies the failure rates obtained from OPEX. This uncertainty from each basic event was systematically carried through in the PSA software package used ‘computer aided fault tree analysis’ (CAFTA) when quantifying all of the different accident scenarios, such that any result that was provided from the PSA contained uncertainty values.

157. I consider that the RP’s approach for uncertainty analysis meets expectations as compared to the PSA TAG (ref. [3]) and that the results demonstrate a fairly low level of uncertainty and thus high level of accuracy in the Level 1 PSA results.

158. To provide insight and interpretation, the RP has provided importance analysis for the Level 1 PSA results. Importance analysis is used to identify and verify the major contributors to the CDF, namely, component failure and human errors. Information given by importance analysis is significant for providing insights for plant safety and indicating some measures to reduce plant risk.

159. The RP presented detailed importance analysis in the BWRX-300 PSA Summary Report (ref. [28]) for top events, systems, components and human failure events. The most important findings were:

- RVR-05 – Excessive LOCA below the top of active fuel is the most risk important top event;

- The control rod drive system is the most risk important system and is highly sensitive to its reliability data. This indicates that overall reliability could be improved by increasing the reliability of this system;
- The most risk important basic event is a CCF of 1/3 of the control rods to insert; and,
- The most risk important human error is a failure to manually actuate the boron injection system (BIS) for ATWS accident scenarios to inject boron into the RPV.

160. The RP considered several sensitivity analysis cases to understand the model's resiliency to changing specific parts of the model on its own. The RP's most risk important findings of the sensitivity analysis were:

- Increasing human error values to the 95th percentile value only raises the overall CDF by a trivial amount;
- The model is very sensitive to CCF values, and using the 95th percentile for CCF values will increase the overall CDF by a large percentage; and
- Increasing or decreasing the testing and maintenance of components does not significantly alter the CDF.

161. The RP also ran several sensitivity cases to understand if specific modelling assumptions of various systems was risk important.

- The base model of the PSA only includes component failures for the ICS and disregards consideration of physical phenomena on which the system is based such as the reliability of natural circulation. When an assumption is added to the model considering the failure of natural circulation, based on OPEX, the CDF increases by 22.7%; and
- Heating Ventilation and Air Conditioning (HVAC) is not modelled in the PSA as there was no detailed design available when the model was created. If HVAC is introduced to the model based on a supercomponent with a typical failure probability for an HVAC system, the CDF increases by between 22.5% - 23.6%.

162. The sensitivity cases that the RP modelled show that the overall PSA results would be expected to increase if these changes were made. However, even with this increase in the CDF, the overall PSA results are still expected to be lower than the project reliability targets.

163. I am content that the RP has identified several design modifications and modelling approaches and has demonstrated the effect these would have on the PSA results. This modelling helps to demonstrate the use of PSA for the project and that the high reliability claims in the safety case are supported by the PSA.

4.14. Low Power and Shutdown PSA

164. I did not sample the methodology and modelling of the low power and shutdown (LPSD) PSA during Step 2 of the GDA. The findings of my detailed internal events Level 1 PSA assessment are applicable to the LPSD PSA and the learning from that assessment provides confidence in the consideration of the risk arising from the design during LPSD operating states.

165. I did review the results of the LPSD PSA and my conclusions of these is in the following paragraphs.

166. The RP created a LPSD PSA to understand sources of risk arising from the design during low power and shutdown operating modes. This is expected in IAEA SSG-3 (ref. [2]) Chapter 9, the SAPs and TAG-30 (ref. [3]). The operating modes included in the LPSD PSA were:

- Hot shutdown;
- Stable shutdown;
- Cold shutdown;
- Refuelling (reactor cavity drained);
- Refuelling (reactor flooded to normal with the spent fuel pool (SFP) gate installed); and
- Refuelling (reactor flooded to normal with SFP gate removed).

167. The RP claims to have followed the ASME LPSD standard (ref. [69]) for creating the PSA model and analysing the results for each of the operating modes. The results of the LPSD PSA CDF and LRF are reported in ref. [16]. It is noted that the results for LPSD CDF and LRF are less than the BWRX-300 quantitative PSA safety goals as written in ref. [15].

168. I found the LPSD PSA conclusions to be logical and thorough and the ultimate results to be in line with expectations given the reported result of the internal events Level 1 PSA. It was clear that the RP has performed analysis to understand the uncertainty and sensitivity of the results. Overall, I am content with the results and learning from the LPSD PSA, and although there is more work to be done as the design progresses, for Step 2 of the GDA I am of the opinion that the results of the LPSD PSA meet my expectations compared with IAEA SSG-3 (ref. [2]) paragraphs 9.72 and 9.73, the PSA TAG-30 (ref. [3]) and SAP FA.12.

4.15. Internal Fire PSA

169. I did not sample the methodology and modelling of the internal fire during Step 2 of the GDA. The findings of my detailed internal events Level 1 PSA

assessment are applicable to the internal fire PSA and the learning from that assessment provides confidence in the consideration of the risk arising from the design during internal fire accident scenarios.

170. I did review the results of the internal fire PSA and my conclusions of these are in the following paragraphs. The internal fire PSA was performed by the RP for both Level 1 and Level 2 PSA.

171. The results of internal fire PSA are found in ref. [28]. I noted that they are higher than the internal events Level 1 PSA CDF and Level 2 PSA LRF reported in ref. [28] whilst still being lower than the BWRX-300 quantitative PSA safety goals as written in ref. [15].

172. It is noted that the majority of the risk contribution arises from fires that produce a general transient. The RP also performed uncertainty analysis for internal fire PSA and found that the use of certain conservative assumptions was judged to be the source of the majority of uncertainty in the results.

173. I found the internal fire PSA conclusions to be logical and thorough and the ultimate results to be in line with expectations given the reported result of the internal events Level 1 PSA and Level 2 PSA. It was clear that the RP has performed analysis to understand the uncertainty and sensitivity of the results. Overall, I am content with the results and learning from the internal fire PSA, and although there is more work to be done as the design progresses, for Step 2 of the GDA I am of the opinion that the results of the internal fire PSA meet my expectations compared with IAEA SSG-3 (ref. [2]) paragraphs 7.68 to 7.70, TAG-030 (ref. [3]) and SAPs FA.12.

4.16. External Hazard PSA

174. To account for external hazards, the RP performed a high wind PSA and a seismic PSA. I did not sample the methodology and modelling of the external hazards PSA during Step 2 of the GDA. The findings of my detailed internal events Level 1 PSA assessment are applicable to the external hazards PSA and the learning from that assessment provides confidence in the consideration of the risk arising from the design during external hazard accident scenarios.

175. The high wind PSA was performed for three scenarios: straight wind, tornadoes and hurricanes. The results for CDF and LRF were below the internal events PSA and also below the BWRX-300 quantitative safety goals.

176. The RP found several areas of the plant to be more risk important to high wind hazards, including the equipment pool and accident sequences containing failures in the ICS isolation valves. Uncertainty analysis was also performed to understand how the results are affected by assumptions and conservative modelling.

177. I found the high wind PSA conclusions to be logical and thorough and the ultimate results to be in line with expectations given the reported result of the internal events Level 1 PSA and Level 2 PSA. It was clear that the RP has performed analysis to understand the uncertainty and sensitivity of the results.

178. The RP completed a seismic PSA to consider risks from the design from seismic activity. As seismic PSA is very location dependent, the RP based its location on a geologically conservative location in the US. This does not bound all UK locations where a BWRX-300 plant may be built, but it is thought to be somewhat typical for the purposes of GDA.

179. I am content with the selection of this set of parameters for the GDA seismic PSA. In my opinion it is better to use a conservative set of parameters and demonstrate the risks arising from the design for GDA, which is what the RP has reportedly used in its analysis.

180. The RP reported the CDF and LRF arising from the seismic PSA in ref. [28]. The results for CDF and LRF were greater than the internal events PSA whilst the CDF was still below the BWRX-300 quantitative safety goals as written in ref. [15]. I noted that the seismic PSA LRF was slightly above the LRF BWRX plant safety goal, identical to the seismic PSA CDF. The RP comments on this in ref. [15] by noting that there is a modelling assumption whereby all core damage sequences are treated as going straight to a large release. The Level 2 PSA part of the seismic PSA has not been developed in detail for GDA. I am content with this as I would not have expected this for the stage of design.

181. The RP reported that there are considerable conservatisms and uncertainty built into the PSA model at the present time. One of the key sources of uncertainty is in seismic failure of the polar crane. The failure of the polar crane is modelled to fail both the RPV and lead to containment bypass, however no structural analysis has yet been completed. Thus, the modelling introduces a significant source of uncertainty into the results. I am content that the RP understands that this uncertainty exists (see ref. [28]) and are aware of it for further consideration in future seismic PSA modelling.

182. I found the seismic PSA conclusions to be logical and thorough and the ultimate results to be in line with expectations given the reported result of the internal events Level 1 PSA and Level 2 PSA. It was clear that the RP has performed analysis to understand the uncertainty of the results. Overall, I am content with the results and learning from the seismic PSA, and although there is more work to be done as the design progresses, for Step 2 of the GDA In my judgement, the results of the seismic PSA meet my expectations.

4.17. Other Assessed Hazards

183. The RP also assessed the risk from the spent fuel pool (SFP) and dropped loads, via two separate PSA models: the SFP Level 1 PSA and the Fuel and Heavy Load Movements Level 1 PSA.

184. The SFP PSA considers risk outside the reactor core for accidents in which the fuel is damaged. Rather than measuring this risk in CDF or LRF, the SFP PSA uses a metric called ‘spent fuel damage frequency (SFDF)’. The SFP PSA covers all operating modes and does not credit any FLEX-type response (such as external mobile supplies of electricity or water).

185. I did not sample the methodology and modelling of SFP PSA during Step 2 of the GDA, however I did review the results of the SFP PSA as reported in the BWRX-300 PSA Summary Report (ref. [28]) as well as any learning.

186. The results of the SFDF were reported as less than the internal events PSA CDF and also below the BWRX-300 quantitative safety goals. The most risk-dominant component was reported to be failure of the SFP injection valve and failure to recover the valve after the PIE whereby all SFP cooling is lost. The SFP injection valve appears to be a single point of vulnerability in the design, however the overall reliability of the SFP appears to be very high. The RP includes consideration of an optioneering PSA modelling case whereby an additional SFP injection valve was added to the design. When this is added to the design the results of the SFDF reduce by almost an order of magnitude.

187. I found the SFP PSA conclusions to be logical and thorough and the ultimate results to be in line with expectations given the reported result of the internal events Level 1 PSA.

188. The RP also completed a Heavy Load Movements Level 1 PSA. This PSA considered accidents that resulted in either fuel damage, or reactor core damage from dropped loads. All plant operating modes were included. The RP considered accidents where heavy dropped loads were over: the SFP, the RPV, fuel, and over an SSC excluding the SFP, RPV or fuel.

189. I did not sample the methodology and modelling of Heavy Load Movements Level 1 PSA during Step 2 of the GDA, however I did review the results of the Heavy Loads PSA as reported in the BWRX-300 PSA Summary Report (ref. [28]) as well as any learning. I noted that the frequency of dropped loads was very low. I discussed this with the ONR mechanical engineering inspector for this GDA and was told that this crane is planned to be built to very high standards, as per the ASME standard for nuclear cranes. The SFP PSA is a preliminary PSA with only a few accident scenarios included, and structural analysis has not yet been completed.

190. The results of this PSA were below the Level 1 internal events CDF and LRF as reported in ref. [28] and lower than the BWRX-300 quantitative PSA safety goals as written in ref. [15].

191. Although the SFP PSA and Heavy Load Movements PSA are preliminary and both use significant assumptions, I found the conclusions to be logical and thorough and the ultimate results to be in line with expectations given the reported result of the internal events Level 1 PSA. Overall, I am content with the results from these PSAs, and although there is more work to be done as

the design progresses, for Step 2 of the GDA I am of the opinion that the results of these PSAs meet my expectations.

4.18. Level 2 PSA

192. I did not sample the methodology and modelling of Level 2 PSA during Step 2 of the GDA. The findings of my detailed internal events Level 1 PSA assessment are applicable to the Level 2 PSA and the learning from that assessment provides confidence in the consideration of the risk arising from the design during the phase of accidents after a core has been damaged.

193. I did review the results of the Level 2 PSA as reported in ref. [16] and my conclusions of these is noted below in the following paragraphs.

194. The RP has not completed a full scope Level 2 PSA yet, and for Step 2 of GDA the RP has only included full power internal events, wind and internal fire. The RP also stated in Chapter 15.9 of the PSR (ref. [16]) that the GDA Level 2 PSA is based on conservative assumptions and expert judgement given the unavailability of supporting containment analysis and severe accident analysis at the time of GDA.

195. The results for the Level 2 PSA highlight the areas for which additional analysis is a priority and also which assumptions need to be validated by further design development. For example, the top sequence is an ATWS sequence where containment fails due to overpressure. Second to this is a RPV rupture sequence where containment fails due to overpressure cause by a rupture-induced pressure surge which over-pressurises containment. The RP notes in ref. [15] that this indicates the need to assess the ultimate containment fragility in order to replace the assumptions being used. I noted that even with the conservative assumptions and expert judgement the RP noted, the LRF is less than the BWRX-300 quantitative PSA safety goals as written in ref. [15].

196. The RP stated in the BWRX-300 PSA Summary Report (ref. [28]) that the systems and components that are important in the Level 1 PSA are also important in the Level 2 PSA, and so the Level 1 insights generally apply to the Level 2 PSA.

197. The RP also stated in the BWRX-300 PSA Summary Report (ref. [28]) that the uncertainty assessment of the Level 2 PSA showed that there is a high level of uncertainty for the parameters in the containment event trees (CETs) due to the fact that they were not supported by analysis, and due to the uncertainty at this point in the design and plant operation. The RP performed sensitivity analysis on these key aspects and found that two areas included in the Level 2 PSA were particularly affecting the final results including: failure of the pedestal water injection system, and assumptions relating to steam explosions. The RP noted that the LRF remained within an order of magnitude of the base Level 2 PSA results even with significant changes to these two areas of modelling, which made the modelling more conservative.

198. The RP also stated in the BWRX-300 PSA Summary Report (ref. [28]) that a significant percentage of the total release frequency is from filtered events. This shows the importance of the containment vent design in terms of capacity and capability to ensure it is reliable and effective, as assumed in the Level 2 PSA.

199. I found the Level 2 PSA conclusions to be logical and thorough and the ultimate results to be in line with expectations given the reported result of the internal events Level 1 PSA. It was clear that the RP has performed analysis to understand the uncertainty and sensitivity of the results. Overall, I am content with the results and learning from the Level 2 PSA, and although there is more work to be done as the design progresses, for Step 2 of the GDA I am of the opinion that the results of the Level 2 PSA meet my expectations.

4.19. Overall Conclusions from the PSA

200. This section presents my conclusions of the GDA review of the BWRX-300 PSA when compared against relevant expectations of the ONR's PSA TAG (ref. [3]). My judgement is based on the significance of the outcomes of my review and its potential impact on the risk profile of the BWRX-300 design. I have considered the following:

- the adequacy of the PSA documentation;
- whether the PSA has a credible and defensible basis (i.e. whether the underpinning data and modelling has credibility and can be shown to represent the plant);
- whether the PSA reflects the design of the BWRX-300 submitted for GDA;
- whether the PSA has enabled a judgement to be made as to the acceptability of the overall risk of the facility against ONR's SAPs numerical targets; and
- whether the PSA has been effectively used to demonstrate that a balanced design has been achieved and that the risk associated with the design and operation of the BWRX-300 is ALARP

201. The BWRX-300 PSA meets the expectations of ONR's PSA TAG (ref. [3]), given the objectives of Step 2, and the design and PSA modelling work still to be done.

202. The BWRX-300 has a credible and defensible basis. The submitted BWRX-300 design is reflected in the PSA submitted for GDA.

203. The BWRX-300 PSA has completed a Level 1 and Level 2 PSA but has not provided a comparison against the ONR SAPs targets 7, 8 and 9. The RP

discusses this shortfall in ref. [15] by referring to FAP item PSR15.6-39 to 41 which states that a Level 3 PSA will be developed in the future to allow direct comparison with the ONR SAP target 9. In addition, for the GDA, the RP justifies the absence of a Level 3 PSA by stating that 'the PSA demonstrates low risk when compared to the plant safety goals for CDF and LRF, especially considering the level of conservatisms at this stage in development'. The RP claims in Chapter 3 of the PSR (ref. [82]) that the application of its stringent intermediate CDF and LRF targets, combined with the radiological protection principles and safety strategy for the BWRX-300 will ensure that Legal Limits are met and that risks can be demonstrated to be tolerable and ALARP.

204. The various PSA results are all very low and although the case has not been made to directly compare the PSA results with the SAPs targets 7, 8 and 9, I am satisfied that the risk from the design has been assessed in detail and determined to be very low.

205. The internal events Level 1 PSA shows a risk profile of the plant where it is significantly dominated by the probability of the RPV to rupture and the consequences and plant response to this initiating event. Ref. [3] expects that a PSA for a modern power plant would show a well-balanced risk profile, where a single area of the plant does not reflect a dominant risk position. However, the BWRX-300 has been specifically designed with the PSA in mind and the design has consciously been adapted to remove former high risk areas of traditional BWR designs. Thus, the traditional areas of risk in a BWR design do not appear as risk-dominant, and the remaining large area of risk is the RPV itself.

206. I am content that although the risk profile is not well balanced, there are valid reasons for this and in fact, it is expected that this will be normal with modern plant design which tend to improve the design to this point.

4.20. PSR Chapter 15.6

207. PSR Chapter 15.6 (ref. [15]) describes the development of the PSA that has been undertaken to analyse the risk profile of the BWRX-300. I have assessed this report and my conclusions about it are found in the following paragraphs.

208. Ref. [15] sets out a high level general approach to PSA for the BWRX-300. It also establishes the BWRX-300 plant safety goals and quantitative PSA safety goals for Level 1 and Level 2 PSA in terms of CDF and LRF. The report sets out a high level PSA scope and a simplistic overview for the methods used in each of the various PSAs. Results for the various PSAs are included with a small discussion surrounding the top risk contributor.

209. In the appendix, ref. [15] states that nuclear safety risks cannot be demonstrated to have been reduced ALARP within the scope of a two-step GDA, but rather points to future work that will be required to 'contribute to the

development of a future ALARP statement'. An appendix table lists the forward action plans (FAPs) that were known at the time the PSR was written.

210. Overall I found Chapter 15.6 of the PSR to be lacking in enough detailed information to judge whether or not the methods, modelling, and results analysis met my regulatory expectations. However, the RP was able to provide me with additional information over the course of Step 2 (notably refs. [28], [60], [59], and [57]) to give me confidence in the adequacy of its approach. Any future BWRX-300 safety case chapter of PSA should be strengthened, in part by summarising and referencing the information the RP already has available.

4.21. Overall View of the Safety Case

211. I have assessed the PSA-related aspects of the safety case that was submitted or requested. I found that there are aspects of the safety case that were well written and very detailed, and I am content with their quality and the information contained. These would be the PSA specific reports such as refs. [28], [60], [59], and [57]. The initial PSA submissions such as refs. [15] and [16], were high level but once I received the PSA specific reports I was able to make a meaningful assessment.

4.22. Defence in Depth

212. The PSA models and reports are useful to judge whether or not the design has sufficiently considered defence in depth, because of the very nature of PSA. PSA is a structured, methodical analysis of accident scenarios looking at thousands of unique combinations of events. In Chapter 3 of the PSR (ref. [82]), the RP claims that the BWRX-300 safety objective is to achieve a design with a very high level of safety with safety and design principles based on a defence in depth approach consisting of five levels of defence. Safety is enhanced by deliberate design decisions informed by deterministic and probabilistic safety analyses, through an iterative safety framework wherein the design is implemented to meet defined safety objectives, which are confirmed via safety assessments. Results of safety assessments then provide feedback regarding the design and the process is repeated as required. Thus, the results of the PSA are a useful tool to judge the adequacy of the design's defence in depth.

213. The CDF and LRF results of the PSAs are very low and generally meet the project safety goals as set out in ref. [15]. There are no identified accident scenarios in any of the PSA models whereby a lack of defence in depth is identified. If there was such a lack, it would be apparent by the PSA results highlighting accident scenarios that had a higher than expected CDF/LRF, which could mean a lack of barriers or defence in depth in the design. This does not appear in any of the PSA results that I reviewed or assessed.

214. The RP also explained in ref. [57] that PSA has been used in the early design to highlight design weaknesses early, and if found the design was modified, if

appropriate, to remove those weaknesses. Examples are the number of ICS trains, having a secondary shutdown system, and the decision to move the primary penetration into the RPV up so high. Although other design engineering areas would have a primary reason to affect these areas of the design, the RP claimed that the PSA provided early insight into these areas and they were modified accordingly early in the design.

215. Overall, I am content with the BWRX-300's consideration of the concept of design in depth, from a PSA perspective, and it is clear from the results and modelling of the PSA that the principle of defence in depth has been in the design since the beginning.

4.23. Passive Safety Systems

216. The BWRX-300 design includes a significant reliance on passive safety systems, such as the ICS system, which is credited in most accident scenarios to both depressurise the reactor and remove decay heat.

217. In the PSA, the ICS system is assumed to always succeed with its mission when called upon to do so, as there is no failure of the ICS due to passive reasons, i.e., physics, etc, modelled explicitly. The RP completed a sensitivity study examining the internal events Level 1 PSA and adding a new basic event which represented failure of the passive function of the ICS, and assigning it a probability based on a similar ICS design using the CAREM reactor in Argentina (see IAEA TECDOC-1752 - ref. [83]).

218. As reported in the BWRX-300 PSA Model Results and Insights Report (ref. [59]) it was found that about a quarter of cutsets credit reliance on the ICS to perform (which is a passive operation). General transient sequences were affected the most. Inclusion of a failure of the ICS due to passive failure was found to increase the CDF of the internal events Level 1 PSA by nearly 22%. The RP referenced deterministic studies that are ongoing to verify that the ICS is designed and sized appropriately for various accident scenarios.

219. I found that the RP has considered the reliability of the ICS passive safety system appropriately in GDA. It is apparent that it is a significantly risk important system, but that is expected from the design of the plant. I judge that safety claims on the ICS have been supported by appropriate safety analysis, design substantiation and experimental work.

220. The RP stated that the passive containment cooling system (PCCS) (another passive system) is not credited in the PSA as it was designed for small LOCAs, and thus does not appear in PSA cutsets. Therefore, the current PSA modelling cannot provide any insights into the adequacy of the PCCS. I judge that this is acceptable as the RP did not identify PCCS relevant accident sequences in the PSA which are of high frequency and for which a theoretical inclusion of PCCS would lower them to acceptable risks.

4.24. Containment

221. The Level 2 PSA relies on the design and assumptions/claims regarding the BWRX-300 containment structure. I did not sample the Level 2 PSA during Step 2 of GDA, however I did review the Level 2 PSA results and learning.

222. In ref. [28], the RP claims that the following containment failure modes are considered explicitly in the Level 2 PSA:

- hydrogen deflagration;
- failure of containment isolation/containment bypass;
- direct containment heating;
- containment basemat melt-through;
- steam explosion in containment;
- failure of containment venting; and
- containment overpressure

223. The RP states in ref. [28] that containment performance analysis was not available for GDA, and thus assumptions are used together with expert judgement to consider containment performance. In the Level 2 PSA, generally conservative assumptions have been used to understand the plant response to various containment failure types. For example, in the hydrogen deflagration accident, it is assumed that hydrogen enters containment and it is assumed that the containment will fail if hydrogen deflagration occurs. Direct containment heating assumes a direct failure of containment. Containment overpressure assumes failure of containment as well.

224. Although containment performance analysis was not completed, and thus failure of the containment in the Level 2 PSA was modelled conservatively, the results of the Level 2 PSA are still very low, and generally below the plant safety goals.

225. In my opinion, the low results of the Level 2 PSA (even with the identified conservative assumptions regarding the containment performance) provides me with confidence that the risk associated with the containment is low.

4.25. Redundancy of Safety Systems

226. The BWRX-300 PSA models all the trains of the various systems and components of the reactor along with appropriate consideration of CCFs.

227. In the PSA Model Results (ref. [59]), the RP discusses more detailed learning from the PSA modelling and results related to the risk important safety systems and redundancy. One of the most risk-important systems in which

redundant claims are made are the shutdown rods. The PSA considers that failure of more than 1/3 of the shutdown rods to insert represents a failure of the reactor to shutdown. Ref. [59] discusses how the project believes that this is somewhat conservative and that in reality it may be possible to achieve full shutdown of the reactor with more than 1/3 of the rods failed. This is kept in the safety case for GDA because there are fewer rods in the BWRX-300 than traditional BWRs and it’s not clear that the CCF multiplier ought to scale for a smaller common cause component grouping. This CCF is identified by the RP to be a major source of uncertainty in the risk profile of the reactor.

228. The ICS system for the BWRX-300 is highly reliable as modelled. This is due, in large part, to having three redundant I&C systems controlling two mechanically diverse valves on three 100 percent capacity ICS trains. Because the overall ultimate safety strategy is to isolate the RPV, reliability of ICS is key to the overall risk of the plant. The results of the PSA shows this design to be highly reliable and is not a large source of risk.

229. Aside from these two systems, the redundancy of other safety systems does not arise in the PSA results to be a major risk contributor. This implies that the redundancy of other safety systems is acceptable for GDA, or it would appear as a risk outlier in the PSA results.

230. I judge that based on the information that I assessed in the PSA, the generic design has incorporated sufficient redundancy in the design of safety systems.

4.26. Challenges of BWR Technology

231. A unique aspect of BWR technology that is considered in the PSA is accidents releasing coolant offsite, from active steam going to the turbine and condensate returning to the RPV. This type of initiating event is considered in PWR PSA models and is generally collected and analysed in the Level 3 PSA to understand the off-site consequences due to PWR secondary coolant. With PWR technology, the secondary coolant is not very active and thus the effect on the general public is not generally very significant from secondary coolant leaks on their own, unless they lead to a greater accident. For BWR technology, the coolant is more active, especially during at-power operation and shortly after shutdown, and thus off-site leaks are considerably more hazardous.

232. These types of accidents are considered in the PSA as various LOCA scenarios outside containment, such as: main steam line break (MSLB), feedwater line break (FWB) and large LOCA and small LOCA outside containment. These accidents are modelled as described previously in this report and my general assessment of the internal events Level 1 PSA can be found earlier. As there is no Level 3 PSA for the BWRX-300 for GDA, these accidents are modelled to understand the plant response, leading to potential

core damage, rather than understanding the effects of an off-site release to the general public.

233. In the assessment of the PSA topic area for the ABWR Step 4 GDA, the RP for that reactor project included a detailed Level 3 PSA which included faults leading to off-site consequences. The ABWR Level 3 PSA results were not very significant for these types of faults, and this is noted in the PSA assessment.

234. I judge, based on this submission and leveraging the positive conclusions of adequacy reached in the ABWR GDA, that the BWR design of having active steam going to the turbine and condensate returning to the RPV is acceptable for UK application from a PSA perspective. This BWR unique design aspect has been considered in the BWRX-300 PSA and based on the ABWR GDA is not likely to be risk-important.

235. In addition, the risks associated with location of the spent fuel pool (required to be above the RPV) and the fact that any benefits of building below grade as opposed to earlier designs, are not modelled in the PSA so no insights can be gained on the benefits or risks of this change.

5. Conclusions

236. This report presents the Step 2 PSA 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. [27]), at the content of most relevance to PSA against the expectations of ONR's SAPs, TAGs and other guidance which ONR regards as relevant good practice, such as IAEA standards SSG-3 and SSG-4.

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

- The PSA models and reports that were submitted are of high quality and demonstrate that the risk arising from the design of the BWRX-300 is understood, and that PSA has been used to inform the design;
- As the design is still in an early stage, there are certain areas of the PSA that are not mature, namely human reliability analysis and I&C. I have identified these areas as shortfalls against IAEA SSG-3 (ref. [2]) and TAG-030 (ref. [3]) and the RP has agreed and has identified these on a forward action plan. I am content with these actions and consider that it is appropriate to bring this part of the model up to maturity when the design has also matured;
- I have assessed the areas of the design which are represented in the PSA and found that the RP has appropriate levels of redundancy and defence in depth with respect to a PSA perspective; and
- I have assessed the unique design aspect of BWR technology whereby radioactive steam exists outside the containment vessel, from a PSA perspective, and find that this aspect is acceptable. I have used learning from the advanced boiling water reactor (ABWR) GDA, and am content that the likelihood of this type of an accident associated with active steam outside containment is expected to be acceptably low.

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

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Appendix 1 – Relevant SAPs considered during the assessment

SAP reference	SAP title
FA.10	Need for PSA
FA.11	Validity of PSA
FA.12	Scope and Extent of PSA
FA.13	Adequate Representation of PSA
FA.14	Use of PSA