Office for Nuclear Regulation

ONR Assessment Report

Generic Design Assessment of the BWRX-300 – Step 2 assessment of Fuel and Core Design

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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 Fuel and Core Design 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 2 of the Design Reference Report (DRR) 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, benchmarking my regulatory judgements against the expectations of ONR's Safety Assessment Principles (SAPs), Technical Assessment Guides (TAGs) and other guidance which ONR regards as relevant good practice, such as International Atomic Energy Agency (IAEA) safety, security and safeguards standards. Where appropriate, I have also considered how I could use relevant learning and regulatory conclusions from the UK ABWR GDA to inform my assessment of the BWRX-300.

My assessment focused primarily on Chapter 4 of the RP's Preliminary Safety Report (PSR) and its associated references relevant to Fuel and Core. The RP's safety case outlines the components of the nuclear reactor core, including the fuel assemblies, reactivity control systems, and core monitoring system. It also provides a detailed description of the nuclear and thermal-hydraulic design aspects related to the reactor core.

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

- Fuel design and performance, design criteria, Fuel Operational Experience (OPEX) and fuel performance codes and methodologies.
- Nuclear design, including reactivity control, equilibrium cycle analysis, and reactor physics codes.
- Thermal-hydraulic design, with emphasis on natural circulation, flow impairment resilience, transient behaviour, and core stability.
- Shutdown systems.
- Core monitoring system.

Based upon my assessment, I have concluded the following:

- The RP's documentation in the Fuel and Core Design area is of good quality and appropriate for the scope and maturity expected at GDA Step 2. The submissions are well-structured and provided sufficient detail to support a fundamental assessment of the design.
- The RP has presented a coherent and structured safety case for the Fuel and Core Design, supported by mature methodologies and analytical tools, and OPEX from similar BWR designs.
- The GNF2 fuel assembly design reflects an extended and systematic design evolution that is likely to effectively reduce operational risk to a level that is as low as reasonably practicable.
- The nuclear design is based on a conventional equilibrium cycle approach which is consistent with RGP and demonstrates adequate shutdown margin, reactivity control, and margin to thermal limits throughout the fuel cycle.
- The thermal-hydraulic design of the BWRX-300 is well-understood and supported by conservative modelling, although further substantiation will be required in future BWRX-300 safety cases. The RP has provided an adequate demonstration for GDA Step 2 that the reactor will remain stable under operational and fault conditions. The passive features implemented in the design combined with the reactor's low power density, should reduce susceptibility to thermal-hydraulic instabilities.
- From a fuel and core perspective, I confirmed that the mechanical shutdown system described in the submission does not introduce novel features or significant differences compared to the system previously assessed in the UK ABWR. I judge that it is likely to meet its safety functional requirements, noting that aspects related to system diversity have been addressed by the Control & Instrumentation (C&I) and Fault Studies disciplines in their respective reports.
- The core monitoring system is similar to previous designs already assessed by ONR, with differences that are consistent with established BWR practices. I am satisfied that the system is capable of providing real-time, spatially resolved data to support safe operation and compliance with operational limits.

I have identified areas which will require further substantiation in future BWRX-300 safety cases and are detailed in this report. However, I do not consider these as fundamental safety shortfalls.

Overall, based on my assessment to date I have not identified any fundamental safety shortfalls that could prevent ONR permissioning the construction of a power station based on the generic BWRX-300 design; noting that any decision to permission a BWRX-300 will require further assessment (in either a future Step 3 GDA or during site specific activities) of suitable and sufficient supporting evidence that can substantiate the claims and proposals made in the GDA Step 2 submissions.

List of abbreviations

ABWR	Advanced Boiler Water Reactor
ALARP	As Low As is Reasonably Practicable
AOO	Anticipated Operational Occurrence
APS	Anticipatory Protection System
BL	Baseline Level
BOC	Beginning Of Cycle
BT	Boiling Transition
BTC	Basic Technical Characteristics
BWR	Boiling Water Reactor(s)
CAE	Claim, Argument and Evidence
CCFL	Counter-Current Flow Limitation
C&I	Control and Instrumentation
CPR	Critical Power Ratio
CR	Control Rod(s)
CNSC	Canadian Nuclear Safety Commission
CRD	Control Rod Drive(s)
CRDA	Control Rod Drop Accident
DAC	Design Acceptance Confirmation
DEC	Design Extension Conditions
DESNZ	Department of Energy Security and Net Zero
DPS	Diverse Protection System
DNB	Departure from Nucleate Boiling
DR	Design Reference
DRC	Doppler Reactivity Coefficient
DRP	Design Reference Point
DRR	Design Reference Report
EA	Environment Agency
EOC	End of Cycle
ESBWR	Economic Simplified Boiling Water Reactor
FAP	Forward Action Plan
FHA	Fuel Handling Accident
FLE	Fuel Loading Error
FMCRD	Fine-Motion Control Rod Drive
GB	Great Britain
GDA	Generic Design Assessment
GVHA	GE Vernova Hitachi Nuclear Energy Americas LLC
GNF	Global Nuclear Fuel
GSE	Generic Site Envelope
GSR	Generic Security Report
GT	Gamma Thermometer(s)
HCU	Hydraulic Control Unit
IAEA	International Atomic Energy Agency
I&C	Instrumentation and Control
ICSO	Isolation Condenser System
KKM	Kern Kraftwerk Müehleberg

LOCA	Loss of Coolant Accident
LPRM	Local Power Range Monitors
LTP	Lower Tie Plate
LTR	Licensing Topical Report
MCPR	Minimum Critical Power Ratio
MDSL	Master Document Submission List
MFLPD	Maximum Fractional Linear Power Density
MOC	Middle Of Cycle
MTC	Moderator Temperature Coefficient
MVC	Moderator Void Coefficient
NPP	Nuclear Power Plant
NRW	Natural Resources Wales
NSF	Niobium (Nb) Tin (Sn) Iron (Fe)
OLMCPR	Operating Limit Minimum Critical Power Ratio
ONR	Office for Nuclear Regulation
OPEX	Operational Experience
PA	Protected Area
PCI	Pellet-Cladding Interaction
PCMI	Pellet-Cladding Mechanical Interaction
PCSR	Pre-construction Safety Report
PER	Preliminary Environment Report
PLR(s)	Partial Length Rod(s)
PPS	Primary Protection System
PRNM	Power Range Neutron Monitoring
PSAR	Preliminary Safety Analysis Report
PSR	Preliminary Safety Report
RGP	Relevant Good Practice
RAPFE	Radial Average Peak Fuel Enthalpy
RIA	Reactivity Initiated Accidents
RITE	Risk Informed, Targeted Engagements
RO	Regulatory Observation
RP	Requesting Party
RQ	Regulatory Query
RPV	Reactor Pressure Vessel
SAP	Safety Assessment Principle(s)
SMR	Small Modular Reactor
SBWR	Simplified Boiling Water Reactor
SSC	Structure, System and Component
SSSE	Safety, Security, Safeguards and Environment Cases
TAG	Technical Assessment Guide(s) (ONR)
TIP	Traversing In-core Probe
TMOL	Thermal-Mechanical Operating Limit
TOP	Thermal Over-Power
TSC	Technical Support Contractor
UK	United Kingdom
US	United States of America
US NRC	United States Nuclear Regulatory Commission
WENRA	Western European Nuclear Regulators’ Association

WRNM	Wide Range Neutron Monitor

Contents

1. Introduction

1. This report presents the outcome of my Fuel and Core Design assessment of the BWRX-300 design as part of Step 2 of the Office for Nuclear Regulation (ONR) Generic Design Assessment (GDA). My assessment is based upon the information presented in the Safety, security, safeguards and environment cases (SSSE) head document (ref. [1]), specifically chapters 4, 5, 7, 15.5 (refs. [2], [3], [4], [5]), the associated revision of the Design Reference Report (DRR) (ref. [6]) and supporting documentation.

2. Assessment was undertaken in accordance with the requirements of the ONR's Management System and follows ONR's guidance on the mechanics of assessment, NS-TAST-GD-096 (ref. [7]) and ONR's risk informed, targeted engagements (RITE) guidance (ref. [8]. The ONR Safety Assessment Principles (SAPs) (ref. [9]), together with supporting Technical Assessment Guides (TAGs) (ref. [10]), 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. [11]).

1.1. Background

4. The ONR's GDA process (ref. [12]) calls for an assessment of the Requesting Party's (RP) submissions with the assessments increasing in detail as the project progresses. This GDA will be finishing at Step 2 of the GDA process. For the purposes of the GDA, GE Vernova Hitachi Nuclear Energy International LLC, United Kingdom (UK) Branch, is the RP. GE Vernova Hitachi Nuclear Energy Americas LLC (GVHA) is a provider of advanced reactors and nuclear services and is the designer of the BWRX-300. GVHA is headquartered in Wilmington, North Carolina, United States of America (US).

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

6. In the UK, the RP has been supported by its supply chain partner Amentum who has assisted the RP in the development of the UK-specific chapters of the 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 (NRW) 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 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 Fuel and Core Design (ref. [13]). 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. [14]) and published on the regulators' website.

1.2. Scope

14. The assessment documented in this report is based upon the SSSE for the BWRX-300 (refs. [1], [15], [16], [17], [2], [3], [18], [4], [19], [20], [21], [22], [5], [23], [24], [25], [26], [27], [28], [29], [30], [31], [32], [33], [34], [35], [36], [37], [38], [39], [40], [41], [42], [43], [44], [45], [46], [47]).

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. [48]). This is further supported by its DRR (ref. [6]) and the Master Document Submission List (MDSL) (ref. [49]). 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 Structures, Systems and Components (SSCs) which are included in the scope of the GDA, and their relevant GDA reference design documents.

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

17. The GDA scope includes the Power Block (comprising the Reactor Building, Turbine Building, Control Building, Radwaste Building, Service Building, Reactor Auxiliary Structures) and Protected Areas (PA) as well as 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 [7] and ONR's guidance on Risk Informed, Targeted Engagements [8].

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:

- The initial core design, including the nuclear design of individual fuel bundles and the core loading pattern for the first operational cycle, is excluded. This exclusion is of low significance to the current assessment, which relies on suitably conservative bounding assumptions for the equilibrium core. It is standard industry practice to develop the detailed first core design at a later stage, and it is expected this information should be provided in future regulatory submissions;

- The specific features and capabilities of the actual commercial core monitoring system are excluded. However, the requirements and a description of the basic functionality of a generic core monitoring system are included within Chapter 4 and Chapter 7 of the Preliminary Safety Report (PSR) and are within the scope of this assessment;

- Detailed safety functional claims applicable to the fuel and associated reactor core systems are excluded. While the design bases are included in Chapter 4 of the PSR, a comprehensive tabulation of all safety functional claims—covering components such as fuel, control rods, and instrumentation—will be required in future (post-GDA) submissions to support deployment of the BWRX-300 in Great Britain;

- The detailed design of interim fuel storage facilities is excluded from this Step 2 GDA assessment. This exclusion is justified by the fact that such facilities will not be required until approximately ten years after the station begins operation. Furthermore, there are currently no known fuel design features that would present compatibility issues with existing interim storage solutions.

20. My assessment has considered the following fundamental aspects:

- Control of core reactivity, ensuring the reactor can be safely shut down under all operating conditions and fault scenarios;

- Removal of heat generated in the fuel via the coolant, which must be effective during both normal operation and fault conditions;

- Containment of radioactive substances within the fuel cladding, ensuring that fuel integrity is maintained during frequent faults and that the fuel retains a coolable geometry under all reasonably foreseeable fault conditions.

21. In addition to the fundamental principles, the following specific considerations were included in the assessment:

- The BWRX-300 features a smaller core and lower power output compared to the already assessed Advanced Boiling Water Reactor (ABWR), while utilising a fuel design (GNF2) that is a closely related and improved version of the GE14 fuel used in the UK ABWR. This significantly simplified the scope of my assessment, as many design aspects of the fuel had already been thoroughly reviewed during the previous ONR assessment of the UK ABWR. To identify and address any differences I conducted a gap analysis;

- The codes, methods, and their verification and validation are consistent with those used for the UK ABWR. As such, I adopted a proportionate assessment approach, leveraging prior regulatory insights;

- The plant design places a fundamental reliance on natural circulation and passive safety systems. Consequently, core stability and cooling performance under natural circulation conditions were a key focus of the assessment, in line with the design's intent to simplify active systems while maintaining robust thermal-hydraulic performance.

2. Assessment standards and interfaces

22. The ONR's 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.

23. 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

24. The ONR SAPs (ref. [9]) 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.

25. The International Atomic Energy Agency (IAEA) safety standards (ref. [50]) and nuclear security series (ref. [51]) 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.

26. Furthermore, ONR is a member of the Western European Nuclear Regulators Association (WENRA). WENRA has developed Reference Levels (ref. [52]), which represent good practices for existing nuclear power plants, and Safety Objectives for new reactors (ref. [53]).

27. The relevant SAPs, IAEA standards are embodied and expanded on in the TAGs (ref. [10]). The TAGs provide the principal means for assessing the Fuel and Core Design aspects in practice.

28. The relevant SAPs and TAGs have been benchmarked against IAEA and WENRA guidance available at the time of publication. Throughout most of this report I have taken credit for this and referred primarily to the SAPs and TAGs rather than directly to international guidance. I have referred to IAEA guidance where it provides directly relevant advice on some topics.

29. 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)

30. The key SAPs applied within my assessment are ERC SAPs, EAD.2, FA.7, AV SAPs.

31. The ERC SAPs set engineering principles that are fundamental to the design of the reactor core and fuel.

32. SAP EAD.2 sets an expectation that adequate margins exist throughout the life of a facility to allow for the effects of degradation. This is important because of the significant way in which fuel and core performance evolve during operation as fuel is depleted, as well as the degradation mechanisms that affect fuel assemblies under irradiation.

33. SAP FA.7 is important because it sets an expectation that fault consequences be predicted conservatively, which has implications for all of the fuel and core data and methods used in fault analyses.

34. The AV SAPs are important because the highly complex nature of neutronic, thermal-hydraulic and thermo-mechanical phenomena occurring in the core mean that the design and safety case are reliant on outputs from analyses using computer codes. The AV SAPs set expectations for assuring the validity of data and models used within, or outputted by, such codes.

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

2.1.2. Technical Assessment Guides (TAGs)

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

- NS-TAST-GD-005 Revision 12 - Regulating duties to reduce risks ALARP (ref. [54])
- NS-TAST-GD-051 Revision 7.1 - The purpose, scope and content of safety cases (ref. [55])
- NS-TAST-GD-075 – Safety of Nuclear Fuel in Power Reactors (ref. [56])

2.1.3. National and international standards and guidance

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

- IAEA SSR-2/1 – Safety of Nuclear Power Plants: Design
- IAEA SSG-52 – Design of the Reactor Core for Nuclear Power Plants
- IAEA SSG-61 – Format and Content of the Safety Analysis Report for Nuclear Power Plants

2.2. Integration with other assessment topics

38. 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.

39. The key interactions with other topic areas were Fault Analysis, Chemistry and Control and Instrumentation (C&I).

40. The Fuel and Core Design assessment provides input to the core performance and design criteria aspects of the Fault Analysis assessment. The assessment of faults is led by Fault Analysis.

41. Reactor Chemistry provides input to the assessment of crud, activation and corrosion aspects of the Fuel and Core Design assessment.

42. I have collaborated with the C&I inspector in our assessments of the core monitoring system. I have also collaborated with the C&I inspector and Fault Studies inspector in our respective assessments of shutdown system adequacy.

43. In addition to the above, there have been interactions between Fuel and Core Design and the rest of the technical areas such as Radiation Protection, Structural Integrity and Human Factors.

2.3. Use of technical support contractors

44. During Step 2 I have not engaged Technical Support Contractors (TSCs) to support my assessment of the Fuel and Core Design aspects of the BWRX-300 GDA.

3. Requesting Party's submission

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

46. 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.

47. This section presents a summary of the RP's safety case for Fuel and Core Design. 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

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

49. The BWRX-300 is the tenth generation of the 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.

50. The BWRX-300 reactor core houses 240 fuel assemblies and 57 control rods inside a steel Reactor Pressure Vessel (RPV). It uses fuel assemblies (GNF2) that are already currently widely used globally (ref. [6]).

51. 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.

3.1.1. Specific Design aspects relevant to the Fuel and Core Assessment

52. The BWRX-300 RPV is a vertical, cylindrical vessel constructed from forged rings, featuring a removable top head secured by a flange, seals, and bolting. It includes penetrations, nozzles, and supports for reactor internals.

53. The RPV is taller than that of a typical forced circulation BWR, due to the inclusion of a "chimney" structure. This chimney extends from the top of the core to the steam separator assembly. According to the RP, this design enhances natural circulation, providing adequate thermal margins during both normal and off-normal conditions.

54. The fuel assemblies are comprised of hermetically sealed fuel rods in a square array along with upper and lower tie plates, water rods, fuel rod spacers, fuel channel and connecting components. The fuel assemblies are supported by the reactor internals. Each core cell consists of a control rod and four fuel assemblies that immediately surround it. Each core cell is associated with a four–lobed fuel support piece. Around the outer edge of the core, certain fuel assemblies are not immediately adjacent to a control rod and are supported by individual peripheral fuel support pieces.

55. The Global Nuclear Fuels GNF2 fuel assembly consists of 92 fuel rods and two large central water rods that occupy eight (8) fuel rod locations contained in a 10x10 array (i.e., 100 lattice locations). Fourteen fuel rod locations are occupied by part length fuel rods.

56. The fuel rod consists of uranium dioxide in the form of cylindrical pellets contained in Zircaloy tubing. The tubing is plugged, sealed, and welded at the ends to encapsulate the fuel. Fuel rods are pressurised internally with helium during fabrication to reduce clad creepdown and promote heat transfer.

57. Core reactivity is managed through a combination of:

- Movable control rods;

- Burnable neutron absorbers (Gadolinia doped fuel rods);
- Natural circulation coolant flow.

58. The BWRX-300 uses the Ultra-HD™ control rod, derived from designs used in BWR/2 through BWR/6. It features a cruciform structure with stainless steel tubes filled with boron carbide powder and, in some locations, hafnium rods for enhanced performance in high-depletion areas.

59. Control rod positioning is achieved using Fine Motion Control Rod Drives (FMCRDs), which are electro-hydraulic mechanisms capable of both fine and rapid movement. Reactor scram is executed via high-pressure hydraulic insertion, powered by nitrogen-water accumulators in Hydraulic Control Units (HCUs), each serving two FMCRDs.

60. The performance of the core is monitored by fixed neutron detectors located within the reactor core. The in-core nuclear instrumentation provides input to automatic reactor core control and protection functions. The BWRX-300 nuclear instrumentation consists of Power Range Neutron Monitoring (PRNM), Gamma Thermometers (GT), and Wide Range Neutron Monitor (WRNM) systems.

61. The PRNM provides neutron monitoring power signals to the Safety Class 1 I&C protection systems. The PRNM also provides signals for post-accident monitoring purposes and, through isolated one-way optical data, links to the core monitoring three-dimensional power distribution program and to the control rod blocking systems.

62. GTs are in-core devices that convert local gamma flux to an electrical signal that supplies information required to calibrate the Local Power Range Monitors (LPRM) in the PRNM system.

63. The WRNM system is a pair of industrial computers which provide redundancy with a real-time operating system that monitors a sub-set of the fixed neutron detectors in the core. The detectors are distributed radially in the core at fixed heights. Each detector is sensitive to neutrons from below criticality to approximately 20% of rated power.

64. The BWRX-300 reactor core operates via natural circulation flow, and according to the RP, is considered a composite design comprised of the most desirable features developed and previously applied to the BWR fleet. The key features are:

    - There are 240 fuel assemblies arranged identically to the Kernkraftwerk Mühleberg (KKM) reactor core.
    - The lattice type is the N-lattice that originated with the ABWR. The N-lattice provides additional moderator volume in the intra-assembly bypass gap as compared to earlier lattice types.

- The core flow results from natural circulation and the nominal bundle flow during power operation is lower than forced circulation reactors.

- The core average power density is low compared to most forced circulation BWRs and approximately 20% lower than KKM (i.e., 870 MWth vs. 1097 MWth).

- The Control Rod Drives (CRDs) are the same fine motion that were developed for the ABWR.

- GTs are used in-lieu of the Traversing In-core Probe (TIPs) system used in the ABWR for LPRM instrument calibration.

- The reference control rod type is the most modern commercially available control rod– the Ultra-HD™.

65. The RP claims that while the exact configuration of the BWRX-300 reactor core is new, the configuration is similar to the BWR operating fleet, and the performance of all principal aspects have been proven in fleet application.

66. The RP claims that any differences from KKM are encompassed within the current approved nuclear methods.

3.2. BWRX-300 Case Approach and Structure

67. The RP has submitted information on its strategy and intentions regarding the development of the SSSE (refs [57], [58], [59], [60]). This was submitted to ONR during Step 1.

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

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

3.3. Summary of the RP’s case for Fuel and Core Design

70. The RP case for Fuel and Core Design is based on the overall claim that:

- The BWRX-300 is capable of being constructed, operated and decommissioned in accordance with the standards of environmental, safety, security and safeguard protection required in the UK.

71. This overall claim is broken down into a set of Level 1 claims relating to environment, safety, security, and safeguards, which are then broken down again into Level 2 area related sub-claims and then finally into Level 3 (chapter level) sub-claims.

72. The level 2 claim related to Fuel and Core design is that:

- Claim 2.1: The design and operation of the fuel and core has been derived and substantiated taking into account Relevant Good Practice (RGP) and Operational Experience (OPEX).

73. This is further broken down in the following subclaims:

- Claim 2.1.1: The safety functions (Design Basis) and integrity claims have been derived for the fuel and associated reactor core systems through a robust analysis, based upon RGP.
- Claim: 2.1.2: The design of the fuel and associated reactor core systems has been substantiated to achieve the required safety functions in all relevant operating modes.
- Claim 2.1.3: The design of the fuel and associated reactor core systems has been undertaken in accordance with proven methodologies, analysis tools and design safety principles and taking account of OPEX to support reducing risks to ALARP.
- Claim 2.1.4: The performance of the fuel and associated reactor core systems are validated by suitable testing, inspection and monitoring throughout manufacturing, operation, and site-based storage.
- Claim 2.1.5: Ageing and degradation mechanisms applicable to the fuel and associated reactor core systems are identified and assessed as an integral part of the design process. Suitable examination, inspection and monitoring are specified to ensure the integrity of fuel remains fit-for-purpose through-life.

74. The aspects covered by the BWRX-300 safety case in the area of Fuel and Core Design can be broadly grouped under three headings which are summarised as follows:

3.3.1. Fuel Design

75. This area of the safety case addresses:

- The description of the fuel design,
- The identification and definition of fuel safety functional requirements (Design Bases),
- The arguments and preliminary evidence supporting the fuel's performance against these requirements.

76. The safety case provides details of the applicable thermal-mechanical criteria and methodologies, analysis tools and design safety principles which are claimed to consider relevant OPEX and contribute to reduce the risk ALARP.

77. Relevant Claims: 2.1.1, 2.1.2, 2.1.3, 2.1.4, 2.1.5.

3.3.2. Nuclear Design

78. The nuclear design case describes the reference equilibrium fuel cycle, expected operating conditions, and how the fuel bundles are designed to perform safely within the core. It includes:

- Analysis methods and computer codes used,
- Main core design limits and conditions of operation,
- Planned core layout and fuel movement,
- Control rod strategy for reactivity control.

79. The BWRX-300 GDA safety case focuses on a 12-month equilibrium cycle, though the RP claims the design can support cycles of up to 18 or 24 months. The 18 or 24-months equilibrium cycle is outside the scope of GDA as it is not part of the BWRX-300 reference design.

80. Thermal-hydraulic stability is also addressed, with predictions for key safety and performance parameters (e.g., reactivity margins, shutdown capability, etc.) across the fuel cycle.

81. Relevant Claims: 2.1.1, 2.1.2, 2.1.3.

3.3.3. Thermal-hydraulic Design

82. This section presents the RP's claim that the reactor core's thermal-hydraulic behaviour is well understood under both normal and fault conditions. The claim is supported by:

- A range of thermal analyses using validated computer codes from previous BWR designs,
- Experimental data, cross-code comparisons, and OPEX.

83. The RP asserts that the analyses demonstrate adequate safety margins and acceptable performance across a variety of scenarios. According to the RP, the methods and codes used have been justified and validated, and the assessment is claimed to address both reactor and non-reactor faults across all operating modes, in accordance with ONR's Safety Assessment Principles.

84. Relevant Claims: 2.1.1, 2.1.2, 2.1.3.

3.4. Basis of assessment: RP's documentation

85. The principal documents that have formed the basis of my Fuel and Core Design assessment of the SSSE are:

- Preliminary Safety Report Chapter 4 (Reactor) (ref. [2])
- Fuel Summary Report (ref. [62])
- Core Nuclear Design Report (ref. [63])
- Thermal-Hydraulics Summary Report (ref. [64])

I have sampled other documents as appropriate and those are referenced in the body of the report.

3.5. Design Maturity

86. My assessment is based on revision 3 of the DRR (ref. [6]). The DRR presents the baseline design for GDA Step 2, outlining the physical system descriptions and requirements that form the design at that point in time.

87. The reactor building and the turbine building, along with the majority of the significant SSCs are housed with the 'power block'. The power block also includes the radwaste building, the control building and a plant services building. For security, this also includes the PA boundary and the PA access building.

88. The GDA Scope Report (ref. [48]) describes the RP's design process that extends from Base Line 0 (BL 0) (where functional requirements are defined) up to BL 3 (where the design is ready for construction).

89. In the March 2024 design reference, SSCs in the power block are stated to be at Base Line 1 (BL1). BL1 is defined as:

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

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

91. The BWRX-300 Fuel and Core Design, as presented at this stage of the GDA, is considered to be at least at BL1. However, I note that, the proposed fuel design, control rods, core monitoring system, and supporting codes and methodologies are not novel; they are well-established and widely used across the nuclear industry. The RP’s safety case provides sufficient detail on the design features of key core components to enable a meaningful assessment at this stage.

92. While detailed substantiation is not expected at Step 2 of the GDA, the submission outlines the fundamental design intent. This provides a sound basis for future development. Further substantiation and supporting evidence are likely to be required in subsequent regulatory submissions to fully support the safety case.

4. ONR assessment

4.1. Assessment strategy

93. 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 Fuel and Core Design 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. [65]).

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

95. Whilst there is no operating BWR plant in the UK, ONR has previously performed a four-step GDA on the Hitachi-GE UK ABWR (ref. [66]). 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.

96. This assessment has been conducted in accordance with the Fuel and Core Design Topic Assessment Plan developed at the start of GDA Step 2 (ref. [13])

97. The following criteria have been used to prioritise areas for detailed scrutiny:

- Safety significance and potential impact on fundamental nuclear safety functions,
- Novelty or uncertainty in the design or supporting safety case,
- Departure from established RGP,
- Areas where the RP's justification appears limited or incomplete.

98. My assessment strategy has been informed by the objectives of Step 2 of the GDA and guided by ONR's expectations for a proportionate, risk-informed, and targeted approach. The strategy has focused on identifying and examining the fundamental claims made by the RP in relation to the Fuel and Core Design, and on sampling supporting arguments and evidence where appropriate.

99. A deviation from the original assessment plan was the decision not to proceed with a planned face-to-face meeting with the RP and international regulators in the United States. Following a detailed review of the RP's safety case, I concluded that the information provided was sufficiently robust to support progression to GDA Step 2 without requiring in-person international engagement.

4.2. Assessment Scope

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

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

102. 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 Fuel and Core Design. 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.

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

- Fuel Design: Including the GNF2 fuel assembly design, thermal-mechanical performance criteria, the derivation of safety functional requirements (Design Bases) and limits and conditions of operation, fuel performance code and methodology.
- Nuclear Design: Including the reference equilibrium cycle, reactivity control strategy, core layout, and the use of validated analysis methods and codes.
- Thermal-Hydraulic Design: Focusing on natural circulation performance, thermal margins, and core stability under normal and fault conditions.
- Instrumentation and Monitoring: Including the proposed in-core instrumentation and core monitoring systems, with emphasis on their role in supporting safe operation and protection functions.
- Shutdown Systems: Including the control rods insertion mechanisms utilised for reactor scram and control rods design

104. Informed by the characteristics of the BWRX-300 design, my assessment has focused on areas where I thought the design either introduces novel features, diverges from established Relevant Good Practice (RGP), or relies on safety arguments that require further scrutiny. I have prioritised:

- Natural circulation and passive safety features, given the design's reliance on these for core cooling and stability.
- Core stability and thermal-hydraulic performance, especially under low-flow conditions, which are central to the reactor's safety case.
- Reactivity control strategy, including the use of Ultra-HD™ control rods and the integration of burnable poisons, to ensure shutdown capability and manage reactivity throughout the cycle.
- Use of established fuel designs (GNF2) and supporting methodologies, where I have applied a gap analysis approach to assess differences from the UK ABWR and determine the extent to which previous regulatory conclusions remain valid.

105. I have reviewed the Forward Action Plans (FAPs) provided in Appendix B of (ref. [5]) and acknowledge the future work commitments and recommendations identified by the RP. While the FAPs have not been considered as part of my assessment for GDA Step 2, I recognise that some of the identified areas—particularly load following capabilities of the BWRX-300 and the management of failed fuel—may warrant further attention in

future BWRX-300 safety cases. I have not sampled these areas in my current assessment, as there do not appear to be any existing design features of the fuel that would suggest a potential weakness in load following performance or indicate specific arrangements for the management of failed fuel at this stage.

106. My assessment has also considered the maturity of the design information provided, recognising that detailed substantiation is not expected at this stage. Where appropriate, I have identified areas where further evidence will be required in future BWRX-300 safety cases to support the RP's claims.

4.3. Assessment

4.3.1. Structure and content of the PSR

107. My expectations on the structure and content of the BWRX-300 Fuel and Core Design safety case are based upon IAEA SSG-61-Format and Content of the Safety Analysis Report for Nuclear Power Plants.

108. To confirm that the RP's submission meets the expected standard I reviewed the consolidated version of the PSR against paragraphs 3.4.1 to 3.4.10 of IAEA SSG-61 which are specific to the Reactor Core. I also conducted a proportionate review against the expectation of the ONR NS-TAST-GD-051 para 5.16 to 5.39 which looks at generic aspects of the safety case.

109. I judge that the PSR Chapter 4 and supporting references follow the structure and the content outlined in the IAEA guidance, and ONR NS-TAST-GD-051, commensurate with the current stage of the BWRX-300 GDA Step 2.

4.3.2. Fuel design

110. The fuel design is a fundamental aspect of nuclear reactor safety and performance. The fuel system must demonstrate robust performance under both normal operating conditions and a range of fault scenarios.

111. While the BWRX-300 builds upon the established design principles of BWRs and the previously assessed UK ABWR, it incorporates optimisations tailored to its smaller core size, reliance on natural circulation and passive safety features.

112. I derive my assessment expectations from the following guidance:

- IAEA SSG-52 – Design of the Reactor Core for Nuclear Power Plants.
- NS-TAST-GD-075 – Safety of Nuclear Fuel in Power Reactors.
- ONR SAPs.

113. My expectation, based upon ONR SAP ERC.1 is that the design and operation of the reactor should ensure the fundamental safety functions are delivered with an appropriate degree of confidence for permitted operating modes of the reactor.

114. Based on ONR SAP EAD.2, I expect a safety case for fuel and core to demonstrate an understanding of the degradation mechanisms which potentially impair the performance of fuel and core components and their ability to deliver their safety functions.

115. This section of the assessment focuses on evaluating the adequacy of the BWRX-300 fuel design, safety functional requirements, design criteria, operational limits and OPEX.

4.3.2.1. Differences between GNF2 and GE14 designs

116. A description of the GNF2 fuel design is provided in the PSR Chapter 4 (ref. [2]) and the supporting Fuel Summary Report (ref. [62]). Overall, the GNF2 design is very similar to the GE14 design proposed for the UK ABWR. Both have been manufactured by Global Nuclear Fuel (GNF) and have extensive operational history, which forms part of this assessment.

117. To confirm the fundamental adequacy of the GNF2 fuel design, I compared it with the previously assessed GE14 design. This comparison focused on identifying the main physical differences between the two designs and determining whether any of these differences would warrant a more detailed assessment. A summary of all the differences is provided in Table 2-1 of the Fuel Summary Report (ref. [62]), however here are reported only those that in my opinion are more significant.

Fuel cladding

118. Since the primary safety function of nuclear fuel is to retain fission products, I reviewed the cladding dimensions of GNF2 and observed that the cladding is slightly thinner than that used in GE14. Thinner cladding may present a slightly increased risk of failure due to reduced mechanical strength, particularly if corrosion or oxidation leads to metal thinning. To confirm whether these effects had been properly accounted for, I reviewed the design criteria for the cladding in more detail.

119. My review found that the design criteria appropriately account for metal thinning and the related temperature increase caused by oxidation and corrosion buildup. These factors are addressed in the GNF's assessment of material properties, the cladding's structural integrity, and in the thermal-mechanical analysis (ref. [62]).

120. To my knowledge, crud and corrosion have been effectively eliminated in the BWR fleet, with no such issues reported in 20 years. To confirm this, I consulted the Chemistry inspector who concluded that the buildup of corrosion products and crud is unlikely to be more onerous than existing

BWRs (ref. [67]). This is due to better water chemistry control, use of additives, improved treatment systems, and more corrosion-resistant cladding materials. On this basis, I am satisfied that the reduced cladding thickness in the GNF2 design presents no fundamental shortfalls and did not sample this area further in Step 2.

121. The GNF2 design retains the same zirconium liner on the inside of the Zr-2 cladding as the GE14 design. This liner serves to protect against Pellet-Cladding Interaction (PCI). OPEX in (ref. [62]) confirms that this feature is highly effective in preventing PCI-related failures. The effect of the zirconium liner was also assessed by ONR in (ref. [68]) and considered a good example of continuous improvement. Therefore, I considered adequate the fact that this feature was retained in the GNF2 design.

122. Another design improvement in GNF2 is the use of a larger plenum in the UO_2 fuel rods. This change helps accommodate fission gases and reduces internal pin pressure, and in my view, this increases the margin to cladding lift-off when compared to GE14 fuel, especially at higher burnups.

123. On this basis, I judge that the expectation of ONR SAP ERC.1 and EAD.2 are likely to be met.

Part-Length Rods (PLRs)

124. A more significant design difference in GNF2 fuel is the introduction of short Partial-Length Rods (PLRs) in two distinct lengths. While PLRs were already present in the GE14 design, GNF2 includes 14 PLRs per lattice: eight approximately two-thirds the length of full-length rods (terminating just above the sixth spacer), and six approximately three-eighths the length.

125. Since the other safety functions of the fuel are core cooling and control of reactivity, I expect these to be delivered with an appropriate degree of confidence for permitted operating modes of the reactor, as per ONR SAP ERC.1.

126. The RP claims that the inclusion of PLRs helps reducing the two-phase pressure drop, which is particularly beneficial in natural circulation systems where adequate cooling depends on maintaining flow. This contributes to improved thermal-hydraulic performance and an enhanced cold shutdown margin.

127. The RP also states that the use of multiple PLR lengths also improves the fuel-to-moderator ratio by creating axial regions within the assembly that contain less fissile material and more moderator.

128. Based on my own experience and understanding of thermal-hydraulic principles, I concur with the RP’s position regarding the reduced pressure drop. While I am not aware of published RGP specifically addressing this

configuration, the claim is consistent with established principles of two-phase flow in vertically oriented fuel assemblies.

129. Regarding the improved fuel-to-moderator ratio, I agree with the RP’s position. It is evident that the termination of PLRs creates water-filled regions that enhance local moderation. This effect is particularly beneficial in the upper part of the core, where steam formation reduces moderator density. By tailoring the axial distribution of fissile material through varied PLR lengths, the design compensates for steam voiding, maintaining a more favourable neutron spectrum and supporting reactivity control and axial power shaping.

130. On the claim of improved cold shutdown margin, I am satisfied with the RP’s argument. For a given lattice enrichment, this is achieved by reducing excess reactivity in selected axial regions, thereby improving the effectiveness of control rods during shutdown. The moderated regions introduced by PLRs are more responsive to absorber insertion, especially under cold, fully flooded conditions when moderator density is highest. This results in a more favourable axial reactivity profile, enhancing subcriticality margins.

131. While detailed substantiation of these effects is expected in future submissions, I judge—based on first principles and alignment with known thermal-hydraulic and neutronic behaviour—that the introduction of PLRs is a positive development.

132. On this basis, I judge that the expectations of ONR SAP ERC.1 are likely to be met.

Channel box

133. The GNF2 design introduces a new material for the channel boxes compared to the Zircaloy Channel box of the GE14 design. The GNF2 fuel channel box is made from a Niobium (Nb)-Tin (Sn)-Iron (Fe) alloy (NSF), and it performs several key functions:

- Forms the outer boundary for coolant flow through the fuel bundle.
- Provides a guiding surface for control rods within the reactor core.
- Offers structural lateral stiffness to maintain the shape and position of the fuel bundle.
- Works together with the Lower Tie Plate (LTP) to control coolant bypass flow at the channel–LTP interface.
- Acts as a heat sink during a Loss-of-Coolant Accident (LOCA); and
- Creates a stagnant volume to support in-core sipping inspections.

134. The stiffness of the fuel assembly is primarily determined by the stiffness of the channel box. As reported in (ref. [62]), testing shows that the channel stiffness in the GNF2 design is essentially the same as that of the GE14 design. Therefore, I judged that the overall fuel assembly stiffness of the GNF2 should remain very similar to that of the GE14 design.

135. In line with ONR SAP ERC.1, I considered this aspect important as fuel assembly stiffness has an impact on fuel assembly bowing, which can affect the ability to insert control rods and/or influence the safety margins on the Critical Power Ratio (CPR) by perturbing the neutron spectrum. I note that fuel assembly bowing was considered in detail during the previous ONR assessment (ref. [68]) of GE14 and judged to be acceptable on the basis that the core design process would minimise channel bowing by constraining the loading of fresh fuel assemblies, periodic monitoring, and control rod insertion tests. I did not find evidence in the RP submission that suggests any changes to the core design approach and therefore I reached the same judgment.

136. In addition, there is indication (ref. [62]) that the use of the new NSF channel material improves resistance to shadow corrosion, a well-documented phenomenon in BWRs, that affects zirconium alloys, which represents a positive development in terms of material durability and long-term performance.

137. Based on the similarity in structural performance, the outcome of the previous ONR assessment, and considering that the operating conditions of the GNF2 in the BWRX-300 are expected to be less onerous compared to the UK ABWR, I concluded that bowing is not expected to pose a concern for the GNF2 design therefore I did not sample this area in more detail for the purpose of GDA Step 2.

Debris filter

138. Another evolution in GNF2 design is the standard inclusion of a LTP debris filter called “Defender™”. This was already an option for the GE14 design, and an improved version is now incorporated as a standard feature in GNF2. The Defender™ filter went through many iterations, optioneering design and testing which are detailed in (ref. [62]) and reflect a thorough process of fuel design improvement.

139. Based on OPEX (ref. [62]), a significant reduction in fuel failures demonstrates the effectiveness of the inclusion of the debris filter in the LTP. Therefore, I judged this to be an appropriate and positive enhancement to the design.

Spacers

140. Guided by ONR NS-TAST-GD-075 I considered whether the fuel design has made adequate provision against failure by debris-fretting caused by foreign

material becoming trapped between fuel pins and spacer grids (or similar structures).

141. Another design change in GNF2, compared to GE14, is a different spacer design including the use of a different spacer material. Spacers are important because they maintain proper spacing between the fuel pins along the axial length of the bundle as well as influencing critical power performance.

142. GNF2 spacers are made completely of Alloy X-750. According to the RP, the new spacers provide a reduced pressure drop and improved resistance to boiling transition as compared to the Zircaloy ferrule spacer design in GE14. Geometrical changes were introduced also to reduce debris capture points, especially in the corners, which further reduce the risk of fretting failure from caught debris.

143. A detailed description of the evolution and the rationale behind the spacer modification is provided in (ref. [62]). The detailed substantiation of the thermal-hydraulic behaviour for this design feature goes beyond the scope of GDA Step 2. However, I acknowledge that this outcome stems from a thorough design development and testing process, and that the effectiveness of the design changes is evidenced by the available OPEX (ref. [62]), which demonstrates that these features are contributing to enhanced overall fuel safety performance.

OPEX

144. I have reviewed the OPEX data provided in Appendix A of (ref. [62]) and I have compared it with publicly available information. The RP's report provides an overview of the fuel reliability assessment, a summary of the OPEX for 10x10 fuel designs and descriptions of historical design updates and improvements.

145. Over 37,000 GE14 fuel bundles have been used in 204 fuel cycles (each lasting 1–2 years) over a period exceeding 20 years. GNF2 fuel OPEX counts over 26,000 bundles and 130 fuel cycles (ref. [62]). Besides the US, GNF2 is licensed to operate in Switzerland, Germany, Spain, Sweden, and Finland.

146. Manufacturing plays a vital role in fuel reliability, as high-quality, proven processes help prevent failures during normal operation. Therefore, it is my expectation that GNF2 fuel assemblies are built to strict specifications for dimensions, materials, and processes, all under a strong quality assurance program.

147. ONR has previously visited the GNF manufacturing facility in Wilmington, North Carolina, and found the quality control aspects of fuel manufacturing be of a high standard, pointing out that the manufacturing defect rates for

GNF fuel are well within benchmark expectations set out in TAG NS-TAST-GD-075. (refs. [68], [69]).

148. Also, I have consulted the ONR Quality Assurance Inspector who made me aware that the RP has a Nuclear Energy Quality Assurance program which has been endorsed by the US NRC ("NEDO-11209-A, Nuclear Energy Quality Assurance Program Description," GE-Hitachi Nuclear Energy, Americas, LLC) .

149. On this basis, I did not consider to be proportionate to visit the Wilmington site or sample manufacturing quality in detail as part of this GDA Step 2 assessment. I remain satisfied that manufacturing quality is adequately controlled under a mature and well-established quality assurance regime.

4.3.2.2. Safety functional requirements and design criteria

150. My expectations for Safety Functional Requirements and design criteria are primarily based on IAEA SSG-52, which states that fuel design limits should be established considering all relevant physical, chemical, and mechanical phenomena affecting fuel rod and assembly performance across all applicable plant states.

151. ONR SAP ERC.1 further requires that the design and operation of the reactor ensure the fundamental safety functions are delivered with an appropriate degree of confidence for all permitted operating modes.

152. In line with ONR SAP ERC.1, I therefore expect a demonstration that the BWRX-300 fuel will fulfil key safety functions during normal operation and frequent design basis faults. These functional requirements are essential for maintaining core integrity and ensuring safe reactor operation. They include:

- Containment of radioactive materials – preventing breach of the fuel cladding and release of fission products into the reactor coolant or containment.
- Reactivity control – maintaining safe and stable reactivity levels during all operational states and ensuring reliable reactor shutdown capability.
- Preservation of geometry for heat removal – maintaining the physical configuration of the fuel to support effective coolant flow and heat transfer.

153. According to IAEA SSG-52 para 3.66, the fuel design should conform to established design limits aimed at preventing known failure mechanisms. These criteria ensure the fuel maintains its mechanical integrity under a range of operational and fault conditions and specifically seek to prevent:

- Excessive cladding creep-out due to high internal pin pressure

- Fuel melting that could compromise structural integrity or reactivity control
- Cladding fatigue and excessive mechanical stress leading to cracking or rupture
- Cladding collapse into axial gaps within the fuel column, causing localised damage
- Loss of cladding ductility from oxidation and hydriding, particularly at high burnup
- Fretting wear due to rod vibration, especially at spacer grid contact points
- Pellet-Cladding Interaction (PCI) failures from mechanical stress and chemical attack (e.g. iodine)

154. It is RGP that a defined set of design limits is applied within thermal-mechanical analyses to ensure these failure modes are avoided and that mechanical integrity is maintained throughout the fuel's design life. These criteria are conservatively based and prioritise the most critical parameters affecting fuel performance, under both normal and Anticipated Operational Occurrence (AOO) conditions.

155. The thermal mechanical design criteria applicable to GNF2 fuel are summarised in Table 3-1 of the Fuel Summary Report (ref. [62]). I reviewed the identified fuel design criteria and confirmed that they remain consistent with those applied to the GE14 fuel design.

156. These criteria and their technical bases were assessed by ONR during the UK ABWR GDA (ref. [68]), therefore I judged to be effective and proportionate not to re-examine their underlying justification in detail for this Step 2 assessment. In my view, the identified GNF2 design criteria continue to reflect RGP, and I am confident that the RP will be able to substantiate them in future regulatory submissions, where required.

157. I initially noted that the Radial Average Peak Fuel Enthalpy (RAPFE) criterion, which is relevant for assessing fuel behaviour during Reactivity-Initiated Accidents (RIA), was not explicitly defined in the RP's submission. This criterion is important because cladding failure can result from stresses induced by Pellet-Cladding Mechanical Interaction (PCMI) during rapid reactivity transients. The RAPFE limit is intended to mitigate this risk by ensuring that fuel enthalpy remains within bounds that preserve cladding integrity.

158. However, the RP has since clarified that the Control Rod Drop Accident (CRDA)—a key RIA scenario—has been eliminated by design through the mechanical coupling of the control blades to the drive mechanism.

159. This justification has not been targeted by Fault Studies or Mechanical Engineering inspectors during Step 2 and may be subject to further regulatory attention in later safety cases. I do not consider this to be a fundamental shortfall from a fuel and core design perspective.

160. For the remaining credible reactivity events—principally startup rod withdrawal errors—the RP applies enthalpy-based criteria from NEDO-33885 Rev.0 “GNF CRDA Application Methodology”, consistent with operating BWR practice. The criterion applied depends on the dominant cladding failure mechanism: brittle fracture via PCMI for rapid insertions, or creep rupture for slower insertions where cladding temperature and differential pressure are the controlling factors.

161. This provides a reasonable basis for Step 2 assessment, but in my opinion, future BWRX-300 safety cases should explicitly demonstrate how the selected criteria meet IAEA SSG-52, Requirement 3.71(d).

4.3.2.3. Fuel design - Conclusion

162. Based on the information provided in the PSR and supporting documentation, I conclude that the GNF2 fuel assembly design reflects an extended and systematic design evolution that is likely to effectively reduce operational risk to a level that is ALARP.

163. This aligns with ONR’s expectations as set out in NS-TAST-GD-005, which emphasises the importance of applying RGP at the design stage, supporting my judgement that risks are likely to be reduced to ALARP.

164. The information that has been submitted is consistent with my expectations, as set out in in IAEA SSG-52, ONR SAP ERC.1, ONR SAP EAD.2, NS-TAST-GD-005 and ONR NS-TAST-GD-075. I am content that there are no fundamental shortfalls in the fuel design which would prevent the RP from further developing the generic BWRX-300 design and associated SSSE evidence to support any future permissioning activities for the construction of a power station based on the design.

4.3.3. Fuel Performance Code and Methodology

165. A general expectation is that analytical models will employ methods that have been appropriately verified and validated. The ONR SAPs AV.1 to AV.8 provide general principles for assessing the verification and validation of models and the associated data used in safety submissions.

166. I selected the fuel rod thermal-mechanical design methodology presented in (ref. [62]) for sampling because this analysis underpins fundamental safety margins by addressing fuel integrity under normal operation and anticipated transients. Given that cladding failure can have direct implications for radiological risk and safety case margins, I judged this to be a proportionate and risk-informed area to sample during Step 2 of the GDA.

167. My expectation, based upon ONR SAP AV.2 is that calculation methods used for the analyses should adequately represent the physical and chemical processes taking place.

168. The fuel rod thermal mechanical analysis is broad in scope, covering a range of complex and interdependent physical phenomena that span the full operational lifetime of the fuel rod. Furthermore, this type of analysis often involves multiple assumptions and simplifications, making it important to confirm that the models used are appropriately verified, validated, and conservative. Therefore, its adequacy supports the overall credibility of the RP's and GNF's approach to fuel performance.

169. The full analysis is covered in six different design reports, three related to the thermal-mechanical performance under steady-state and slow-transient conditions, one related to thermal-mechanical performance under fast-transient conditions, and two for evaluating the potential for creep collapse. In addition to the design methodologies, two additional reports define the material properties used to design fuel rods with an alternate fuel pellet containing GNF's alumino-silicate fuel additive and an alternate cladding material called Ziron.

170. However, for the purpose of GDA Step 2, I limited my review to the fundamental aspects of the methodology, relying primarily on the summary information provided in Section 3.2 of (ref. [62]). In doing so, I also considered the applicability of previous ONR assessments of fuel performance codes and the similarities between the GNF2 and the GE14 fuel designs.

171. I note that GNF has employed the PRIME code for the thermal-mechanical analysis of GNF2 fuel rods. PRIME is a finite element/finite difference code incorporating phenomenological material models for $UO_2/(Gd_2O_3)$ fuel pellets and Zircaloy-2 cladding, accounting for dependencies on temperature, time, neutron flux, and burnup. The code is used to predict temperature distributions within the fuel and cladding as a function of power, and to calculate cladding strain under normal operating and slow-transient conditions.

172. PRIME was also used for the UK ABWR's GE14 fuel. ONR previously assessed the code in detail—including its modelling approach, treatment of uncertainties, and validation—and concluded that it was fit for purpose (ref. [68]). Therefore, I considered it effective and proportionate not to re-sample in detail the PRIME code's modelling or its verification and validation for the purpose of GDA Step 2. Instead, I took confidence from the previous ONR assessment that the code is adequate for this application.

173. In my view, the high-level description of the fuel rod design methodology, outlines a technically sound and coherent approach consistent with current industry practice.

174. From a fundamentals perspective, I am satisfied that:

- The methodology is structured around establishing conservative power-versus-exposure limits and applying these as design and operational constraints, which in my experience aligns with accepted nuclear fuel design practices.
- The treatment of both steady-state and AOO conditions, including the use of bounding assumptions, demonstrate the intent to maintain margin and ensure compliance with thermal-mechanical design criteria. This is in line with the expectations set out in ONR SAP AV.3 which states that where uncertainty in the data exists, an appropriate safety margin should be provided.
- The use of finite element modelling to assess cladding creep collapse is well established and was recently updated to reflect modern fuel characteristics (e.g. smaller axial gaps between the pellets due to higher density of the fuel and reduced in-reactor densification). This is in line ONR SAP AV.2, which states that calculation methods used for the analyses should adequately represent the physical and chemical processes taking place.

175. I noted that the PRIME fuel rod design methodology is currently applicable to Zircaloy-2 cladding with UO_2 or $(U,Gd)O_2$ fuel pellets. However, the RP may seek to use Ziron cladding—a high-iron variant of Zircaloy-2 which is claimed to have improved hydrogen pickup characteristics—and additive fuel pellets, which incorporate dopants to mitigate PCI failures. These materials have been approved for use with PRIME in the US by the US NRC, however their suitability with PRIME has not been assessed during the UK ABWR GDA nor the BWRX-300 GDA.

176. Ziron cladding and Additive fuel are not part of the reference design for the BWRX-300 GDA. Hence, should the RP intend to use Ziron cladding and additive fuel in the UK context, it is my opinion that further substantiation would be required in future BWRX-300 safety cases to confirm the suitability and applicability of the PRIME modelling approach to this material, particularly in relation to cladding behaviour and associated performance models. This substantiation work may be an opportunity for future collaboration with the US NRC.

177. To confirm that there are no novel aspects, I have also sampled the summary of the stress analysis of the fuel rods presented in section 3.3 of (ref. [62]).

178. Consistent with expectations, the analysis used conservative (bounding) power-versus-exposures values for determination of parameters such as fuel rod internal pressure, fuel centerline temperature, the Thermal Over-Power limit (TOP) and the Thermal-Mechanical Operating Limit (TMOL). I noted that the TMOL, based on fuel ramp test data, is higher for GNF2 compared

to GE14, indicating a greater margin in terms of allowable linear heat generation rate (compared at the same exposure points).

179. The summary of the analysis shows that a Monte Carlo statistical approach was used to assess the impact of various factors, including pressure differential, cladding ovality, internal temperature gradients, spacer contact, thermal bow, and circumferential thermal gradients. The resulting stresses were combined using Von Mises theory and compared against material limits to produce design ratios at key cladding locations. Analyses were performed for a range of operating conditions and burnup stages and covered both UO_2 and gadolinia fuel rods.

180. The results, as presented in Table 3-2 of (ref. [62]), show that GNF2 fuel rod stresses remain within acceptable limits. Overall, I was able to confirm that the approach used is not novel and reflects normal fuel design practices.

4.3.3.1. Fuel performance codes and methodologies - Conclusion

181. The information that has been submitted is consistent with my expectations, derived from the AV series of SAPs. I am content that there are no fundamental shortfalls in fuel performance codes and methodologies which would prevent the RP from further developing the generic BWRX-300 design and associated SSSE evidence to support any future permissioning activities for the construction of a power station based on the design.

4.3.4. Nuclear design

182. A description of the BWRX-300 nuclear design is provided in the PSR Chapter 4 (ref. [2]) and the supporting Nuclear Core Design Report (ref. [63]).

183. The BWRX-300 reactor core arrangement is that of KKM, a GE BWR/4 class of reactor, that recently completed service and has been retired. The core is made of 240 fuel assemblies and 57 control rods/drives. The core is significantly smaller compared to the UK ABWR with a rated thermal power of 870 MWth and a power density of approximatively 40 kW/L. The reactor rated conditions are summarised in table 3-1 of (ref. [63]).

184. My expectations on the BWRX-300 nuclear design for this Step 2 GDA are informed by IAEA SSR 2/1 Requirements 43 to 46, and the relevant ONR SAPs ERC.1 to ERC.3.

185. The safety functional requirements of the core design are:

- Control of core reactivity to enable the reactor to be safely shut down under all circumstances.
- Removal of heat produced in the fuel via the coolant fluid.
- Containment of radioactive substances (actinides and fission products).

186. For this Step 2 GDA I sought to confirm the adequacy of:

- safety margins for controlling reactivity during both normal and fault conditions,
- the proposed reactivity control mechanisms,
- the design features intended to mitigate the insurgence of instabilities, or coolant flow impairment considering the dynamic behaviour of natural circulation,
- reactivity coefficients, in particular how the BWRX-300 manages the Moderator Temperature Coefficient (MTC) and the negative void coefficient to ensure stable and predictable power regulation under both normal and fault conditions,
- codes and methods to predict margins to nuclear design basis requirements, control of power distribution and core stability,
- the maturity of verification & validation evidence for the nuclear design codes and methods applied comparing it with the codes and methods utilised for the UK ABWR design.

4.3.4.1. Nuclear design approach

187. The placement of fuel assemblies within the core is a key factor influencing overall core performance. I expect this to result from a systematic and well-justified design process that ensures the core maintains sufficient margins to safety limits throughout the fuel irradiation cycle.

188. I therefore sampled the RP's nuclear design approach for the BWRX-300 as presented in (ref. [63]) and judged it to be adequate.

189. In my opinion, the design methodology follows a conventional and well-understood equilibrium cycle approach, consistent with standard BWR practice and similar to that used in the UK ABWR GDA. This provides confidence that the core design is based on proven principles and is capable of delivering predictable and safe performance across cycles. The basis of my judgement is provided below.

190. The reference cycle is based on a small reload batch fraction and assumes annual refuelling, with the objective of achieving high discharge exposure. I confirmed that the average bundle exposure is within industry experience. While this supports efficient fuel use, more importantly, it ensures that the core operates within established thermal and mechanical limits over the fuel lifetime.

191. The fuel loading pattern of the BWRX-300 is informed by the core geometry of KKM, which shares the same core layout. I note that fresh fuel is placed in locations that support reactivity control and power shaping, with enrichment

tailored to manage local power peaking and maintain adequate margins to thermal limits such as Minimum Critical Power Ratio (MCPR) and Maximum Fractional Linear Power Density (MFLPD). In my opinion this strategy reflects a standard approach and should prevent fuel cladding damage by ensuring reliable heat removal under all operating conditions, if thermal limits are met (see subsection 4.3.4.3.).

192. The RP claims that the use of gadolinia burnable absorbers is optimised to manage hot excess reactivity and shape the axial power profile. This is important for controlling reactivity at startup and for avoiding excessive axial peaking, which could challenge thermal limits. The optimisation also accounts for the number of inserted control rods during steady-state operation, ensuring that the design remains conservative under normal and off-normal conditions. Based on my experience, I concur with the RP's argument and I consider the use of gadolinia absorbers to be an efficient method to manage hot excess reactivity and axial power shaping.

193. The RP claims that non-fresh fuel is placed in peripheral and control cell locations (control cells are areas of the core made by four fuel assemblies and a control rod) to flatten the radial power distribution and reduce local reactivity. I agree with the RP's argument as, in my experience, using non-fresh fuel in the peripheral regions of the core, contributes to a more uniform power profile, which helps maintain thermal margins and reduces the risk of localised overheating.

194. In my opinion, the approach adopted for the BWRX-300 is consistent with standard BWR nuclear design practice and reflects a mature, safety-focused methodology that supports the overall adequacy of the design.

4.3.4.2. Codes and methods

195. The purposes of the reactor physics methods are:

- to confirm that the core design will conform to the operating limits assumed in performance and fault analysis,
- to provide data to describe core performance in faults.

196. In addition, the reactor physics codes are used to set the protection limits on MCPR by providing pin power factors for use in the thermal-hydraulic assessment. The adequate representation of the core performance is therefore an important part of the reactor design.

197. I reviewed the suite of nuclear analysis codes and methods used to support the BWRX-300 core design and judged them to be adequate. I confirmed that these tools—TGBLA, PANACEA, and TRACG—are the same as those used during the UK ABWR GDA, where they were assessed in detail by ONR (Ref. [68]).

198. I considered whether there were novel features in the BWRX-300 fuel design that would warrant further assessment of the identified codes. Based on my experience and considering the similarities of the GNF2 and GE14 design, I judge that those design features should be straightforwardly represented by the same codes. However, I did not consider it proportionate to sample the evidence at a further level of depth, as this is beyond the scope of GDA Step 2.

199. Regarding the TRACG code, the fault studies inspector gained sufficient confidence through ONR's previous assessment of the codes and the US NRC and CNSC reviews of the application of TRACG that there would be no significant shortfalls against the expectations for validation of computer codes, as set out in AV.1 to 6 of the SAPs (ref. [70]).

200. In my opinion, the validation evidence for TRACG in natural circulation applications may warrant further scrutiny in future BWRX-300 safety cases, as ONR did not examine this aspect in depth during the previous UK ABWR GDA. Nonetheless, based on the discussion above, I judge the use of TRACG to be adequate for the purposes of GDA Step 2.

4.3.4.3. Equilibrium cycle analysis

201. The BWRX-300 equilibrium cycle analysis is based on a 12-month fuel cycle equilibrium core made of four different fuel bundle types. Details of the core configuration are presented in (ref. [63]).

202. I reviewed the analysis, and I judge it to be adequate for GDA Step 2 on the following basis:

- The analysis addresses the key critical aspects of core behaviour, including neutronics, thermal performance, reactivity feedbacks, shutdown capability, and thermal-hydraulic stability.
- The RP employs established simulation tools and applies conservative bounding assumptions.
- The results demonstrate compliance with operational and safety limits, with substantial margins in key areas such as MCPR, MFLPD, shutdown margin, and stability.
- The trends and behaviours observed are consistent with established expectations for BWR equilibrium cycles, reinforcing confidence in the design approach.

203. In my assessment of the RP's equilibrium cycle analysis, I focused on evaluating whether the reactor core design ensures safe and stable operation throughout the fuel cycle. The areas I reviewed are directly tied to fundamental nuclear safety principles, including reactivity control, thermal margin assurance, shutdown capability, and system stability. These

analyses are essential to demonstrate that the core will behave predictably under normal operation, AOOs, and postulated accident conditions.

204. In summary I found the following, which meets my expectations derived from IAEA SSG-52, ONR SAP ERC.1 and ONR NS-TAST-GD-075:

- Hot excess reactivity remains modest throughout the cycle and decreases more rapidly toward end-of-cycle. This is important as it defines the reactivity margin that must be controlled during startup, informs the required worth and configuration of control rods and burnable absorbers, and serves as a key input to the shutdown margin calculation, ensuring the core can be rendered subcritical under the most reactive conditions.

- The average fuel bundle exposure is in line with industry experience.

- The axial power shape transitions from bottom-peaked at Beginning Of Cycle (BOC) to a cosine shape at End Of Cycle (EOC), reflecting expected voiding and control rod withdrawal effects, typical of BWRs.

- The Minimum Critical Power Ratio (MCPR) remains above the likely range of Operating Limit MCPR (OLMCPR) throughout the cycle. The reported minimum MCPR in my view provides a strong margin against boiling transition. The OLMCPR is made up of two components: MCPR99.9%, which ensures 99.9% of fuel rods avoid boiling transition under normal conditions, and the transient delta MCPR, which accounts for the reduction in margin during the most limiting AOO. I confirmed that the methodology for the determination of CPR is not different from that assessed and judged to be adequate during the UK ABWR GDA (ref. [68]), and therefore I have not sampled it in detail.

- I reviewed the deterministic analysis presented in Chapter 15.5 (ref. [5]) and verified that fuel limits are not challenged for bounding faults scenarios. I gained sufficient confidence that the BWRX-300 is likely to maintain sufficient margin against boiling transition under both steady-state and transient conditions.

- The Doppler Reactivity Coefficient (DRC) is negative under all conditions

- The MTC is negative for all temperatures at or above hot standby. I note that the RP used a bounding low-reactivity core—characterised by minimal control rod insertion and elevated xenon concentration—to ensure conservative evaluation of the MTC. In my view this approach is adequate because it is based on the understanding that the intra-assembly water gap, where cruciform control rods reside, is a region of thermal flux peaking, and contributes positively to the MTC due to its high moderating effect. When control rods are inserted, they displace this water, thereby reducing the positive contribution to reactivity from moderator temperature changes. By modelling a low-reactivity core with minimal rod insertion—achieved through low enrichment, short cycle length, extended coastdown, and high xenon—the design maximises the water-filled gap volume and thus the potential for a positive MTC. This results in a conservative bounding case that envelopes normal operating conditions and ensures that the MTC remains negative at or above hot standby.

- The Moderator Void Coefficient (MVC) is negative across all exposures and operating conditions.
- The Power Coefficient, being the combination of DRC, MTC, and MVC, is also negative above hot standby, ensuring stable reactivity feedback during power manoeuvres.
- The Cold Shutdown Margin is maintained throughout the cycle.
- The backup boron injection system is capable, under conservative assumptions (all rods out, zero xenon), to provide shutdown margin above the required limit.
- The Decay Ratio remains below the limit in all evaluated scenarios, confirming that power oscillations are effectively damped. Regional instability was assessed using asymmetric flow perturbations. The results show rapid convergence to in-phase oscillations, which would be detected and mitigated by the reactor protection system. The RP claims that design features such as natural circulation, low power density, a tall chimney, and high inlet orifice pressure drop contribute to enhanced stability and reduced susceptibility to instabilities compared to other BWRs. Based on first principles I concur with the view that those design feature should provide improved stability performance, however further scrutiny is recommended in this area for future submissions, particularly on the methodology for the determination of the pressure drops in the chimney (see section 4.3.5 on core stability for more details).

205. Based on the above, I judge that the equilibrium cycle analysis presented in the PSR and supporting references reflects RGP and that the proposed core design is adequate for the purpose of GDA Step 2.

4.3.4.4. Core misloading/ Fuel Loading Error (FLE)

206. The regulatory expectations for assessing fuel loading errors are set out in ONR TAG NS-TAST-GD-075, which advises inspectors to consider whether adequate controls are in place to prevent inadvertent criticality during core loading, including the potential for multiple errors.

207. ONR SAP FA.7 further requires that fault analyses be conservative and use appropriate methods to demonstrate that risks are reduced to ALARP.

These principles emphasise the need for robust analysis and effective procedural controls to manage fuel misloading risks.

208. I reviewed how the RP has addressed the potential for Fuel Loading Errors (FLE), also known as core misloading, as part of my assessment. I selected this fault for sampling because it was previously identified in the ONR's assessment of the UK ABWR. I wanted to determine whether similar concerns apply to the BWRX-300, particularly in light of its smaller core, tall chimney, and different refuelling configuration—all of which could influence the likelihood and detectability of such faults.

209. Upon reviewing the PSR and supporting documentation, I found no explicit discussion or analysis of FLE scenarios. To clarify the RP's position, I raised Regulatory Query RQ-02130 requesting information on:

- Preventive and mitigative measures in place.
- Justification for not performing a cycle-specific analysis.
- Consequences of misloading events from both criticality and radiological perspectives.
- Whether the Fuel Handling Accident (FHA) described in PSR Chapter 15.5 (ref. [13]) is considered bounding in terms of operator dose.

210. In response (ref. [71]), the RP stated that the BWRX-300 adopts established BWR refuelling practices, including scripted procedures, modern fuel handling equipment, and independent verification through video surveillance. These measures are intended to ensure that FLEs remain infrequent events. The RP also referenced U.S. regulatory precedent to justify the absence of a cycle-specific Critical Power Ratio (CPR) analysis, arguing that bounding evaluations demonstrate that fuel damage is not expected, even in undetected misloading scenarios.

211. From a criticality safety perspective, the RP referenced evaluations previously carried out for the UK ABWR, which were assessed by ONR and deemed adequate (ref. [68]). These evaluations demonstrate that even in the event of significant misloading configurations, the reactor remains subcritical with all control rods inserted. I was able to confirm that these analysis are applicable to the BWRX-300 because both the ABWR & the BWRX-300 are of the same lattice type (i.e., N-lattice) with control rods having the same reactivity worth requirements. The reactivity of the core is dominated by the "error region" and the size of the core is a second order effect, although I note that the specific reactivity characteristics of the fuel will define the maximum subcritical array size.

212. The RP also provided results from a mislocated bundle study for the BWRX-300 reference core, which indicated that cladding failure is not expected and

that thermal margins remain adequate. I confirmed that the analysis presented is conservative and that adequate margins exist.

213. Regarding radiological consequences, the RP confirmed that the FHA analysis—assuming 129 failed rods and resulting in a dose of 1.7 mSv to control room operators—is bounding. In contrast, the FLE scenarios evaluated for the BWRX-300 predict zero failed rods, reinforcing the conclusion that dose consequences from FLE are significantly lower, and therefore I did not sample further in this area for the purpose of GDA Step 2.

214. I judge the response provided to RQ-02130 to be adequate for the purposes of GDA Step 2. However, while the RP's justification relies heavily on U.S. precedent and bounding arguments, future BWRX-300 safety cases will need to substantiate the arguments in the UK regulatory context, particularly in relation to human factors, operator training, and demonstration that the approach reduces the risks to ALARP.

4.3.4.5. Nuclear design - Conclusion

215. The information that has been submitted is consistent with my expectations, as set out in in IAEA SSR 2/1, IAEA SSG-52, ONR SAP ERC.1 to ERC.3, ONR SAPs FA.7, and ONR TAG NS-TAST-GD-075. I am content that there are no fundamental shortfalls in the BWRX-300 Nuclear Design which would prevent the RP from further developing the generic BWRX-300 design and associated SSSE evidence to support any future permissioning activities for the construction of a power station based on the design

4.3.5. Thermal-hydraulic design

216. My expectation based on ONR SAP ERC.1 is that the design and operation of the reactor should ensure the heat is removed from the core and that there are suitable and sufficient margins between the normal operational values of safety-related parameters and the values at which the physical barriers to release of radioactive materials are challenged.

217. The BWRX-300 Thermal-hydraulic design is described in section 4.4 of the PSR Chapter 4 (ref. [2]) and the supporting Thermal-Hydraulic Summary Report (ref. [64]). The reactor coolant system description is provided in PSR Chapter 5 (ref. [3])

218. I reviewed the above documents and relevant analyses, and I judged that the core thermal-hydraulic design of the BWRX-300 is adequate for the purpose of GDA Step 2. A summary of my findings is provided in the following sections.

4.3.5.1. Natural circulation core cooling

219. The thermal-hydraulic configuration of the BWRX-300 closely resembles that of conventional BWRs, with the notable exception that it operates without

recirculation pumps or associated coolant piping. Instead, core coolant circulation is achieved entirely through natural circulation mechanisms.

220. While the BWRX-300's reliance on natural circulation is a key design feature that is relatively uncommon in the UK nuclear industry, natural circulation itself is not a novel concept in BWRs. Given its limited application in the UK, I considered it important to assess this aspect of the BWRX-300 design at a fundamental level to better understand its operational principles and safety implications.

221. For instance, the KKM reactor in Switzerland, although primarily operating with forced circulation, demonstrated the ability to sustain natural circulation under certain conditions, such as during shutdown or transient scenarios. This highlights the viability of natural circulation in BWRs. Notably, commercial reactors like Dodewaard NPP in the Netherlands and Humboldt Bay NPP in the USA operated for several decades using natural circulation, providing valuable operational data that informed modern passive safety designs.

222. Passive coolant flow in BWRs is primarily driven by the density difference between the heated fluid in the core and chimney region and the cooler fluid in the downcomer—a well-understood principle in natural circulation systems. The RP claims that in the BWRX-300 design, this effect is enhanced by a tall chimney structure located between the top guide plate and the steam separator assembly, which increases the natural circulation head. Based on this understanding, I judge to be credible that the resulting flow rate is inherently self-regulating and varies with reactor power, as fluid density in the core and chimney region changes with thermal output.

223. Fuel bundle flow in the BWRX-300 is regulated using orificed fuel supports, strategically distributed throughout the core. The orifices are designed to preferentially direct flow to higher-power channels, helping to balance thermal margins across the core. Accurate prediction of individual bundle flow and power distribution is essential for determining margins to thermal limits, such as the MCPR. Together with the Fault Studies Inspector (Ref. [70]) I have assessed the codes used to determine margins to thermal limits and found that they contain no fundamental shortfalls, as discussed in subsection 4.3.4.2.

224. Flow distribution across fuel assemblies and bypass paths is calculated using detailed pressure drop models, which include friction and local loss coefficients, two-phase multipliers, and void-quality correlations. The GNF2 fuel assembly features unique hydraulic characteristics—including inlet orifices, tie plates, spacers, and water rods—which are incorporated into the relevant analysis models.

225. Accurate determination of pressure drops is essential for ensuring safe reactor operation. It helps verify that coolant adequately flows through all fuel assemblies, supports calculation of MCPR, and informs the design of

core components and recirculation systems. Pressure drop data also plays a key role in validating thermal-hydraulic models and ensuring the reactor can respond safely during transients or accidents.

226. The submission (ref. [64]) provides several references to OPEX, and relevant testing and experiments which have been conducted on fuel bundles to characterise their behaviour in a wide range of conditions. The outcomes of these tests are incorporated into GNF’s standard, industry-wide accepted analytical codes and methods.

227. While I did not undertake a detailed review of the thermal-hydraulic modelling and validation data for GDA Step 2, I consider that the RP has provided a reasonable basis to support their claims at this stage and I am confident that further substantiation can be provided in future BWRX-300 safety cases.

228. Given the importance of accurate pressure drop determination in underpinning the safety analysis, the future BWRX-300 safety case should substantiate the uncertainties associated with pressure drop calculations in the chimney region. This is important because pressure losses significantly influence natural circulation flow rates and thermal-hydraulic stability. However, I do not consider the absence of this evidence to be a fundamental shortfall for GDA Step 2 as, in my view, it would be disproportionate to expect a detailed quantification of these uncertainties at this stage.

4.3.5.2. Core flow impairment

229. My expectation based upon ONR SAP ERC.3 is that the core should not undergo sudden changes of conditions when operating parameters go outside the permitted range, and that an increase in reactivity or reduction in coolant flow caused by unplanned movement within the core, loss from or addition to the core of any component, object or substance should be prevented.

230. I assessed core blockages because I was concerned about the long-term effectiveness of natural circulation cooling in the BWRX-300, particularly in low power/shutdown scenarios where debris could obstruct fuel inlet flow. The submission did not initially provide a detailed justification, so I raised RQ-01873 and RQ-02110.

231. The RP’s responses (refs. [72], [73]) addressed both LOCA-induced and operational sources of flow impairment. They confirmed that fibrous debris—previously a concern in suppression pool designs like the UK ABWR—is not a credible issue in the BWRX-300 due to the absence of a suppression pool and improved insulation practices. For operational debris, the RP described robust foreign material exclusion (FME) practices and referenced industry OPEX showing that even in high crud environments, flow blockage at the lower tie plate is not observed.

232. The RP also provided a bounding thermal-hydraulic analysis for the BWRX-300, assuming total blockage of the fuel assembly inlet. This analysis evaluated both Counter-Current Flow Limitation (CCFL) and pool boiling Departure from Nucleate Boiling (DNB) under conservative decay heat conditions. The analysis was deliberately conservative: it assumed complete inlet blockage, saturated coolant conditions, and high-power peaking without credit for nuclear feedback or power reduction. In addition to these pessimistic assumptions, a sensitivity analysis using multiple critical heat flux correlations has been carried out. The results presented in (ref. [73]) show that CCFL and pool boiling DNB will not occur with considerable margin. This demonstrates that adequate cooling would still be maintained in the unlikely event of significant flow blockage.

233. The RP also highlighted specific design advantages that reduce the likelihood of foreign material ingress. These include the absence of recirculation and jet pumps (common sources of component failure in legacy BWRs), a simplified primary system with fewer moving parts, and a natural circulation flow regime that is less capable of lifting large or heavy debris into the core. Additionally, the feedwater system includes multiple barriers—such as strainers, impellers, and spargers—that act as filters to prevent debris from reaching the reactor. In my view, this highlights an adequate RP's consideration of defence-in-depth against fuel failures.

234. Based on this, for the purpose of GDA Step 2, I judge that the RP's response is adequate, and I have sufficient confidence that the BWRX-300 can maintain fuel cooling even in the event of significant core flow blockage.

4.3.5.3. Transient analysis

235. I have examined the results of the BWRX-300 transient analysis provided in Section 6 of (ref. [64]) and I found that across all fault scenarios considered, there are adequate margins to safety limits, with no predicted fuel failures under normal and anticipated fault conditions.

236. I note the following exceptions:

- A fuel misloading fault, where a small number of fuel failures are conservatively assumed in absence of further analysis. In (ref. [66]), no dedicated thermal-hydraulic analysis was performed for this fault scenario. Instead, a conservative assumption is made that fuel rod failures may occur due to a fuel bundle operating at higher power than intended. Upon detection of fuel failure via offgas system activity, operators are required to identify the failed rods and insert a control rod to suppress power. If power suppression is unsuccessful, the reactor is shut down. This is consistent with BWRs operation practice worldwide.

- Sensitivity analyses following a single rod withdrawal Design Extension Condition (DEC), where a limited number of fuel failures may occur.

237. In general, I confirmed that:

- There are large margins to peak cladding temperature limits, supporting fuel integrity.
- The fuel remains wetted in all scenarios, indicating effective core cooling is maintained.

238. In conclusion, the transient analysis summary provides sufficient confidence that the BWRX-300 design maintains appropriate safety margins under a range of fault conditions. I consider the approach and results to be adequate, and I judge that the expectations set out in ONR SAP ERC.1 are met for the purpose of GDA Step 2.

4.3.5.4. Core stability

239. My expectation based upon ONR SAP ERC.3 is that the core should be stable in normal operation and should not undergo sudden changes of condition when operating parameters go outside their permitted range.

240. BWRs, including the BWRX-300, can exhibit coupled neutronic and thermal-hydraulic instabilities under certain conditions. These instabilities manifest as periodic oscillations in power and coolant flow, driven by density waves—regions of high void fraction coolant moving through the core. If severe, such oscillations could challenge fuel cladding integrity. Xenon-induced oscillations are not considered a concern for the BWRX-300, as OPEX across the BWR fleet has shown these to be highly damped due to strong negative void reactivity feedback.

241. Two types of thermal-hydraulic instabilities are recognised in commercial BWRs:

- Type I Instabilities: These occur during startup and are hydraulic in nature, without a reactivity or power response. They result from vapour formation in the chimney, reducing hydrostatic head and causing flow oscillations. These are inherent to natural circulation reactors like the BWRX-300 and are typically of low magnitude, posing no threat to cladding integrity.
- Type II Instabilities: These involve coupled power and flow oscillations due to density waves. If these oscillations grow large and void fractions are high, they may compromise safety margins. These are the primary concern for BWR stability.

242. Three modes of density-wave instability are identified:

- Core-wide – all channels oscillate in phase.
- Regional – half the core oscillates out of phase with the other half.

- Single-channel – localised flow oscillations with minor power fluctuations.

243. The RP defines the decay ratio as the ratio of the peak amplitude of one oscillation to the next. A stable system must have a decay ratio less than 1, and the BWRX-300 design sets a limit of 0.8. Stability to core-wide oscillations is assessed by applying pressure perturbations under a range of operating conditions (BOC, Middle Of Cycle (MOC), EOC, and AOOs with reduced feedwater heating) using the TRACG code. The results indicate that the decay ratio remains below 0.8 in all cases, with a peak of 0.71 at MOC. In faulted conditions, such as after a loss of feedwater heating, the decay ratio is further reduced to below 0.6.

244. An extreme sensitivity case was analysed by the RP to assess the potential for regional instabilities. In this scenario, inlet flow velocities were varied by ±20% across the two halves of the core, with the simulation initiated from a pessimistic 115% rated power condition. The results showed that initial antiphase oscillations rapidly condensed into core-wide oscillations within a few seconds. Although the decay ratio exceeded 0.8 in this case, this was attributed to the artificially high initial power level. Even under such conditions, the oscillations would be detected by the core monitoring system and scram initiated on a high flux signal, mitigating the risks of fuel failure.

245. Feedback from a joint NRC-CNSC seminar (ref. [74]) highlighted that reliance solely on the core-average decay ratio may obscure localised instability risks. A channel-level decay ratio could be higher, indicating less effective damping in specific regions. In my opinion, while this does not undermine the current conclusions for GDA Step 2, I would expect the future BWRX-300 safety case to include further substantiation, providing spatially resolved decay ratio data to strengthen the case for stability.

246. The RP's comparison of the BWRX-300 with the Swiss KKM plant, which has a similar core layout and natural circulation capability for decay heat removal in AOOs and shutdown conditions, supports the claim of enhanced stability. This is because the key parameters such as power/flow ratio, core inlet subcooling, and orifice design are within the tested range of OPEX. However, while the core design is comparable, the surrounding thermal-hydraulic conditions may differ. A detailed comparison of these conditions would be necessary to fully validate the applicability of KKM experience, though in my opinion this is beyond the scope of GDA Step 2.

247. The RP uses the TRACG code and PRIME model for stability analysis, consistent with the methodology used for the UK ABWR. While ONR only assessed TRACG's qualification for forced circulation in the ABWR GDA, the RP asserts that a substantial code qualification basis exists for natural circulation. This forms part of detailed design assessment and therefore falls outside the scope of GDA Step 2. Based on the information provided, I have confidence that the RP's future detailed design can substantiate the

TRACG's modelling capabilities under natural circulation conditions (see section 4.3.4.2.).

248. I note that the Selected Control Rod Rapid Insertion (SCRRI) system, similar to that used in the UK ABWR, is incorporated in the BWRX-300. Its primary function is to reduce core power in response to overpower events, such as those following a loss of feedwater heating. It is not intended as a primary means of suppressing instabilities but contributes to maintaining stability margins during transients.

249. I observed that the modelling of pressure drops, particularly in the chimney region, is a critical factor in the stability analysis. Uncertainties in geometry, flow regime transitions, and boundary conditions could significantly affect the accuracy of fault analysis. However, this forms part of detailed design assessment and therefore falls outside the scope of GDA Step 2.

250. Given the importance of accurate pressure drop determination in underpinning the safety analysis, the future BWRX-300 safety case should substantiate the uncertainties associated with pressure drop calculations in the chimney region. This is important because pressure losses significantly influence natural circulation flow rates and thermal-hydraulic stability.

251. Based on the information provided, I consider the following stability-related claims made by the RP to be credible:

- Power oscillations that result in conditions exceeding specified acceptable fuel design limits are not possible.
- Regional instability is not possible under credible operating conditions.
- Design features prevent the loss of stability margin during upset events.

252. In conclusion, the information provided gives sufficient confidence that the BWRX-300 design is likely to maintain appropriate margins towards the insurgence of instabilities. I consider the approach and results to be adequate for the purpose of GDA Step 2, and I judge that the expectations set out in ONR SAP ERC.3 are likely to be met.

Core stability - Conclusion

253. In conclusion, the RP's stability analysis using TRACG and PANAC11, shows that the reactor exhibits decay ratios well below the design limit of 0.8 under a range of normal and fault conditions. The analysis also demonstrates that regional instabilities, when artificially induced, rapidly evolve into core-wide oscillations that are benign and detectable by the protection system.

254. The design incorporates several passive features that enhance stability, including a tall chimney, large downcomer, and high inlet orifice pressure

drops. These features, combined with the reactor’s low power density, reduce susceptibility to thermal-hydraulic instabilities. The SCRRI system provides additional margin by reducing power following events such as loss of feedwater heating.

255. While some modelling assumptions—particularly in the chimney region and pressure drop determination—require further substantiation in future BWRX-300 safety cases, the current evidence supports the conclusion that the BWRX-300 meets the expectations of SAP ERC.3 for the purpose of GDA Step 2.

4.3.5.5. Thermal-hydraulic design - Conclusion

256. The information provided in the safety case gives sufficient confidence that the thermal-hydraulic design of the BWRX-300 is well-understood and supported by conservative modelling, although further substantiation will be required in future BWRX-300 safety cases.

257. In my opinion, the RP has provided an adequate demonstration for GDA Step 2 that the reactor will remain stable under operational and fault conditions. The passive features implemented in the design combined with the reactor’s low power density, should reduce susceptibility to thermal-hydraulic instabilities.

258. The information that has been submitted is consistent with my expectations, as set out in in ONR SAP ERC.1 and ERC.3. I am content that there are no fundamental shortfalls in the BWRX-300 thermal-hydraulic design which would prevent the RP from further developing the generic BWRX-300 design and associated SSSE evidence to support any future permissioning activities for the construction of a power station based on the design

4.3.6. Shutdown systems

259. The description of the BWRX-300 reactivity control system design is provided in Section 4.5 of the PSR Chapter 4 (ref. [2])

260. Reactivity Control within the BWRX-300 design consists of:

- Control rods and Control Rod Drive Systems (discussed in this section).
- Supplementary reactivity control in the form of fuel rods containing gadolinia (burnable neutron poison).
- Backup Boron Injection System.

261. My expectation, based on ONR SAP ERC.2, is that at least two diverse systems should be provided for shutting down a civil nuclear reactor.

4.3.6.1. Reactor scram

262. My objective was to identify any major differences or novel aspects of the BWRX-300 control rod system compared to the UK ABWR design. I confirmed that the BWRX-300 system is fundamentally identical to that of the UK ABWR, which was previously assessed and considered adequate by ONR (ref. [68]). It features bottom-mounted FMCRDs capable of both fast electric motor-driven positioning and hydraulic scram insertion via nitrogen accumulators.

263. In the BWRX-300, a scram is initiated by opening the scram valve in the Hydraulic Control Unit (HCU), allowing high-pressure water to lift-off the hollow piston from the ball nut and drive the control rod into the core. A spring washer buffer assembly halts the piston at the end of its stroke. As the piston separates from the ball nut, spring-loaded latches engage with the guide tube to hold the control rod in the fully inserted position.

264. Simultaneously, a scram follow signal activates the FMCRD electric motor, which moves the ball nut upward to just below the latched piston. This provides a diverse backup insertion mechanism and ensures continued support of the control rod. If the hydraulic scram does not occur, the motor-driven run-in alone can insert the rod, with the piston remaining in contact with the ball nut and the latches staying retracted.

265. I have considered the possibility of failure to scram due to both I&C faults and mechanical obstruction, and I have engaged with the relevant C&I and Fault Studies inspectors. These potential failure modes are addressed in the respective C&I and Fault Analysis reports (refs. [75], [70]).

266. I note that the Fault Studies inspector has identified a potential shortfall in the diversity of means of shutdown, specifically that both the hydraulic scram and motorised control rod run-in rely on the same physical mechanism — control rod insertion — to achieve shutdown. This might introduce a common cause failure vulnerability and challenges the expectations of ONR's SAPs ERC.2 and IAEA SSR 2/1 Requirement 46, which call for two independent and diverse means of shutdown. The Fault Analysis report further discusses the reliability claims associated with these systems and the potential role of the back-up shutdown system, currently a manually actuated Class 3 system, as an ALARP measure. These considerations fall outside the scope of this Fuel and Core assessment but are being addressed by the Fault Studies discipline.

267. Regarding the I&C matter, the reactor shutdown safety functions are initiated by the SC1 Primary Protection System (PPS), SC2 Diverse Protection System (DPS), and SC3 Anticipatory Protection System (APS). The PPS is claimed to be independent and diverse from the DPS and APS to meet overall plant safety targets. This independence claim has been assessed by ONR's C&I inspector (ref. [75]), resulting in Regulatory Observation RO-

BWRX300-001 (Ref. [76]) and the RP's resolution plan (ref. [77]) concerning the demonstration of independence and diversity in the I&C architecture.

268. From a fuel and core perspective, I confirmed that the shutdown system described in the submission does not introduce novel features or significant differences compared to the system previously assessed in the UK ABWR. I judge that it is likely to meet its safety functional requirements, noting that aspects related to system diversity have been addressed by the C&I and Fault Studies disciplines in their respective reports.

4.3.6.2. Control rods

269. The BWRX-300 uses the Ultra-HD™ control rod design, which represents an evolution of the control rod technology used in earlier BWRs, including those proposed for the UK ABWR.

270. The Ultra-HD™ control rods retain the proven cruciform geometry but incorporate a refined absorber configuration, combining boron carbide powder capsules with hafnium rods, particularly in the outer, high-depletion regions. According to the RP, this hybrid absorber arrangement improves neutron absorption characteristics over the fuel cycle, especially in areas subject to prolonged exposure, thereby enhancing shutdown reliability and extending component life.

271. In line with the objectives of GDA Step 2, I have not reviewed the detailed substantiation of the Ultra-HD™ control rod design. However, based on my understanding and experience, this design is expected to offer improved resistance to swelling and embrittlement, better mechanical stability under irradiation, and more uniform depletion behaviour compared to earlier control rod designs. These enhancements should ultimately contribute to improved nuclear safety and reduced radioactive waste.

272. In addition to the hydraulic and electrically driven shutdown systems, the BWRX-300 includes a boron injection system as a backup means of reactivity suppression. In my opinion, this system should provide an additional layer of defence-in-depth, ensuring that sufficient negative reactivity can be introduced to shut down and maintain the reactor in a subcritical state in the unlikely event of control rod insertion failure. Additional considerations on systems diversity are addressed in the ONR Fault Studies report and C&I report (ref. [70], [75]) as explained in section 4.3.6.1.

273. In my view, the adoption of the Ultra-HD™ control rod design—combined with the proven FMCRD system and the backup boron injection capability—represents a robust and well-balanced approach to reactivity control.

4.3.6.3. Shutdown systems - Conclusion

274. The information that has been submitted is consistent with my expectations, as set out in in ONR SAP ERC.2. Recognising the outstanding

considerations on systems diversity addressed in the ONR Fault Studies report and C&I report, I am content that there are no fundamental shortfalls in the BWRX-300 shutdown system design which would prevent the RP from further developing the generic BWRX-300 design and associated SSSE evidence to support any future permissioning activities for the construction of a power station based on the design.

4.3.7. Core monitoring system

275. My expectation based on ONR SAP ERC.4 is that the core should be designed so that parameters and conditions important to safety can be monitored in all operational and design basis fault conditions and appropriate recovery actions taken in the event of adverse conditions being detected.

276. The description of the core monitoring system is provided in Section 4.6 of PSR Chapter 4 (ref. [2]) and PSR Chapter 7 (ref. [4]). An overview is provided is Section 3 of this report.

277. Based on my review, I consider the BWRX-300 core monitoring system to be adequate for the purposes of GDA Step 2. The system provides comprehensive three-dimensional core power monitoring using live reactor data, processed through a coupled nuclear thermal-hydraulic diffusion model. It integrates signals from in-core instrumentation, including Local Power Range Monitors (LPRMs) and Gamma Thermometers (GTs), to adapt power distribution calculations and assess thermal margins.

278. I confirmed that the LPRM system detectors used in the BWRX-300 are similar as those proposed for the UK ABWR and assessed by ONR in (ref. [68]). In the BWRX-300 there are 13 vertical “strings” of LPRMs arranged radially throughout the BWRX-300 core. Each string has four LPRMs equally spaced vertically and equally along height of the core. The LPRM strings are located between the corners of the fuel channels of the adjacent four fuel bundles. The total number of 52 LPRMs are divided into the three divisions such that each division gets an approximately equal number of vertical and radial LPRM core locations. This configuration provides detailed axial and radial resolution of local power conditions and supports accurate core monitoring and protection functions.

279. Gamma Thermometers (GTs), which replace the Transverse In-Core Probes (TIPs) used in the UK ABWR design, are passive devices that measure gamma heating in the core. They are used to support LPRM calibration and improve the accuracy of power distribution predictions.

280. I queried the safety classification of the GTs, and the RP confirmed they are classified as Safety Class 3. This is consistent with the classification of the UK ABWR TIPs. ONR previously judged that a Class 3 designation does not present a fundamental shortfall, provided that LPRM calibration is supported by additional administrative measures to confirm its validity (ref. [68]). I found

no evidence to suggest that this approach would not be applicable to the BWRX-300. Therefore, I judge the classification to be acceptable for the purposes of GDA Step 2, with full justification to be provided in future BWRX-300 safety cases. I confirmed that my judgement is aligned with the C&I assessment (ref. [75]).

281. It is the responsibility of the RP (or future licensee) to demonstrate that calibration practices are adequate to support the safety case, particularly where errors could affect the operation of systems important to safety. Therefore, I expect that calibration or testing activities where error could result in the wrong operation of systems important to safety will need to be substantiated in future safety cases.

282. For the purpose of GDA Step 2, while I have not reviewed the detailed implementation at this stage, the described architecture and functionality of the proposed core monitoring system are consistent with established BWR core monitoring practices and I am satisfied that the system is capable to provide real-time, spatially resolved data to support safe operation and compliance with operational limits.

4.3.7.1. Core monitoring system - Conclusion

283. The information that has been submitted is consistent with my expectations, as set out in in ONR SAP ERC.4. I am content that there are no fundamental shortfalls in the BWRX-300 core monitoring system design which would prevent the RP from further developing the generic BWRX-300 design and associated SSSE evidence to support any future permissioning activities for the construction of a power station based on the design.

5. Conclusions

284. This report presents the Step 2 Fuel and Core Design 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. [13]), at the content of most relevance to Fuel and Core Design against the expectations of ONR's SAPs, TAGs and other guidance which ONR regards as relevant good practice, such as IAEA SSG-52 – Design of the Reactor Core for Nuclear Power Plants.

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

- The RP's documentation in the Fuel and Core Design area is of good quality and appropriate for the scope and maturity expected at GDA

Step 2. The submissions are well-structured and provided sufficient detail to support a fundamental assessment of the design.

- The RP has presented a coherent and structured safety case for the Fuel and Core Design, supported by mature methodologies and analytical tools, and OPEX from similar BWR designs.
- The GNF2 fuel assembly design reflects an extended and systematic design evolution that is likely to effectively reduce operational risk to a level that is as low as reasonably practicable.
- The nuclear design is based on a conventional equilibrium cycle approach which is consistent with RGP and demonstrates adequate shutdown margin, reactivity control, and margin to thermal limits throughout the fuel cycle.
- The thermal-hydraulic design of the BWRX-300 is well-understood and supported by conservative modelling, although further substantiation will be required in future safety cases. The RP has provided an adequate demonstration for GDA Step 2 that the reactor will remain stable under operational and fault conditions. The passive features implemented in the design combined with the reactor’s low power density, should reduce susceptibility to thermal-hydraulic instabilities.
- From a fuel and core perspective, I confirmed that the mechanical shutdown system described in the submission does not introduce novel features or significant differences compared to the system previously assessed in the UK ABWR. I judge that it is likely to meet its safety functional requirements, noting that aspects related to system diversity have been addressed by the C&I and Fault Studies disciplines in their respective reports.
- The core monitoring system is similar to previous designs already assessed by ONR, with differences that do not present any fundamental shortfalls. I am satisfied that the system is capable of providing real-time, spatially resolved data to support safe operation and compliance with operational limits.

286. I have identified the following areas which in my opinion will require further substantiation in future safety cases, although I do not consider these as fundamental safety shortfalls:

- Treatment of fuel loading errors and demonstration that the overall approach to core loading reduces the risks to ALARP.
- Modelling of chimney pressure drops, and spatially resolved decay ratio and their impact on natural circulation and stability.

- Calibration substantiation for core monitoring systems, particularly where Class 3 instrumentation supports Class 1 systems.

287. Overall, during GDA Step 2 I have not identified any fundamental safety shortfalls in the generic BWRX-300 design. Subject to the provision and assessment of suitable and sufficient supporting evidence in either a future GDA Step 3 or during site specific activities, I have found nothing that would prevent ONR permissioning the construction of a power station based on this design.

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Appendix 1 – Relevant SAPs considered during the assessment

SAP reference	SAP title
EAD.2	Lifetime Margins
FA.7	Consequences
ERC.1	Design and operation of Reactors
ERC.2	Shutdown Systems
ERC.3	Stability in normal operation
ERC.4	Monitoring of parameters important to safety
AV.1	Theoretical models
AV.2	Calculation methods
AV.3	Use of data
AV.4	Computer models
AV.5	Documentation
AV.6	Sensitivity studies
AV.7	Data collection
AV.8	Update and review