GE Hitachi Nuclear Energy

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BWRX-300 UK Generic Design Assessment (GDA) Safety, Security, Safeguards and Environment Summary

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EXECUTIVE SUMMARY

GEH's BWRX-300 is a 300 MWe water-cooled natural circulation Small Modular Reactor utilising simple natural phenomena-driven safety systems. It is the tenth generation of the Boiling Water Reactor (BWR) and represents the simplest BWR design since General Electric, GEH's predecessor in the nuclear business, began developing nuclear reactors in 1955. Target applications include base load electricity generation, load following electrical generation generally within a range of 50 to 100% power, hydrogen production, district heating, and other process heat applications.

GEH entered the BWRX-300 design into Generic Design Assessment (GDA) with the United Kingdom (UK) Office for Nuclear Regulation (ONR), the Environment Agency (EA) and Natural Resources Wales (NRW) with the objective of gaining regulatory confidence on the acceptability of a conceptual full plant design for deployment in England and Wales. GEH, referred to in the GDA process as the 'Requesting Party', will exit GDA at the end of Step 2 before progressing directly to site licensing, and therefore a further objective is to gain regulatory confidence in the strategy to develop from the Step 2 GDA submissions to a future site-specific project. GEH intends to complete GDA Steps 1 and 2 without additional design changes beyond the Design Reference. Instead, any design changes beyond the GDA will be tracked for implementation using Forward Action Plans in the UK as part of future site-specific licensing.

The purpose of this document is to summarise GEH's main submission as part of this two-step GDA process. This main submission is made up of four volumes that address the topics of safety, security, safeguards, and the environment across the lifecycle of the BWRX-300. The submission, and its supporting references, is designed to be broad enough in scope and detail to enable a meaningful assessment by the ONR, EA and NRW.

GEH believes that this submission and its supporting references provide confidence that there is a viable path towards substantiation for all claims made related to safety, security, safeguards and the environment. Given this path to substantiation, GEH believes the submission shows the feasibility of constructing, commissioning, operating, and decommissioning the BWRX-300 design at a site in England or Wales. The document also contains a summary of the Forward Action Plan process that is used to capture commitments for future work in the development of the BWRX-300 in the UK (Section 12).

ACRONYMS AND ABBREVIATIONS

Acronym	Explanation
ABWR	Advanced Boiling Water Reactor
AC	Alternating Current
ALARA	As Low As Reasonably Achievable
ALARP	As Low As Reasonably Practicable
AOO	Anticipated Operational Occurrence
ASME	American Society of Mechanical Engineers
BAT	Best Available Techniques
BDBA	Beyond Design Basis Accident
BIS	Boron Injection System
BL	Baseline
BPVC	Boiler and Pressure Vessel Code
BSI	British Standards Institution
BWR	Boiling Water Reactor
C&S	Codes and Standards
CAE	Claims, Arguments and Evidence
CB	Control Building
CCF	Common Cause Failure
CCS	Containment Cooling System
CDF	Core Damage Frequency
CEAP	Continuous Exhaust Air Plenum
CF	Conventional Fire
CFD	Condensate Filters and Demineralisers System
CFR	Code of Federal Regulations
CFS	Condensate and Feedwater Heating System
CHS	Conventional Health and Safety
CIS	Containment Inerting System
CIV	Containment Isolation Valve
CNSC	Canadian Nuclear Safety Commission
CRD	Control Rod Drive System
CUW	Reactor Water Cleanup System
CWE	Chilled Water Equipment
CWS	Circulating Water System
CySSP	Cyber Security System Plan

Acronym	Explanation
D-in-D	Defence-in-Depth
DBA	Design Basis Accident
DBT	Design Basis Threat
DEC	Design Extension Condition
DL	Defense Line
DSA	Deterministic Safety Analysis
DSILW	Dry Solid Intermediate Level Waste
EA	Environment Agency
ECCS	Emergency Core Cooling System
EFS	Equipment and Floor Drain System
EHE	External Hazard Evaluation
EKP	Engineering Key Principle
EME	Emergency Mitigation Equipment
ESBWR	Economic Simplified Boiling Water Reactor
ESF	Engineered Safety Feature
EUST	End User Source Term
FAP	Forward Action Plan
FFA	Functional Failure Analysis
FFHE	Functional Failure Hazard Evaluation
FMCRD	Fine Motion Control Rod Drive
FMEA	Failure Modes and Effect Analysis
FP	Fission Product
FPC	Fuel Pool Cooling and Cleanup System
FSF	Fundamental Safety Functions
FW	Feedwater
GDA	Generic Design Assessment
GDC	General Design Criteria
GDF	Geological Disposal Facility
GEH	GE-Hitachi Nuclear Energy
HAW	Higher Activity Waste
HEPA	High Efficiency Particulate Air
HFE	Human Factors Engineering
HFEA	Human Failure Event Analysis
HLW	High Level Waste

Acronym	Explanation
HOHE	Human Operation Hazard Evaluation
HVAC	Heating, Ventilation, and Air Conditioning
HVS	Heating, Ventilation, and Cooling System
I&C	Instrumentation and Control
IAEA	International Atomic Energy Agency
IC	Isolation Condenser
ICC	Isolation Condenser Pools Cooling and Cleanup System
ICS	Isolation Condenser System
IEC	International Electrotechnical Commission
IHE	Internal Hazard Evaluation
ILW	Intermediate Level Waste
INSAG	International Nuclear Safety Group
ISOE	Information System on Occupational Exposure
LAW	Lower Activity Waste
LLW	Low Level Waste
LLWR	Low Level Waste Repository
LOCA	Loss of Coolant Accident
LPRM	Local Power Range Monitor
LRF	Large Release Frequency
LWM	Liquid Waste Management System
MCA	Main Condenser and Auxiliaries
MCPR	Minimum Critical Power Ratio
MCR	Main Control Room
MSL	Main Steam Line
MSR	Moisture Separator Reheater System
MTE	Main Turbine Equipment
NBS	Nuclear Boiler System
NRW	Natural Resources Wales
OGS	Offgas System
ONR	Office for Nuclear Regulation
OPEX	Operational Experience
PA	Protected Area
PCCS	Passive Containment Cooling System
PCS	Primary Containment System

Acronym	Explanation
PCW	Plant Cooling Water System
PER	Preliminary Environmental Report
PIE	Postulated Initiating Event
POCO	Post Operational Clean Out
PSA	Probabilistic Safety Analysis
PSfR	Preliminary Safeguards Report
PSR	Preliminary Safety Report
PST	Primary Source Term
PSyR	Preliminary Security Report
PWR	Pressurized Water Reactor
RB	Reactor Building
RCPB	Reactor Coolant Pressure Boundary
RGP	Relevant Good Practice
RIV	Reactor Isolation Valve
RPV	Reactor Pressure Vessel
RWB	Radwaste Building
SAA	Severe Accident Analysis
SAP	Safety Assessment Principle
SB	Service Building
SBD	Safeguards by Design
SBWR	Simplified Boiling Water Reactor
SC	Safety Class
SCCV	Steel-Plate Composite Containment Vessel
SCN	Non-Safety Class
SDC	Shutdown Cooling System
SDD	System Design Description
SF	Spent Fuel
SJAE	Steam Jet Air Ejector
SMR	Small Modular Reactor
SNI	Sensitive Nuclear Information
SRV	Safety Release Valve
SSCs	Structures, Systems and Components
SWM	Solid Waste Management System
SyBD	Security by Design

Acronym	Explanation
TAG	Technical Assessment Guide
TB	Turbine Building
TGSS	Turbine Gland Seal System
TIG	Technical Inspection Guide
UK	United Kingdom
UPR	Ultimate Pressure Regulation
UPS	Unifying Purpose Statement
U.S.	United States
USNRC	U.S. Nuclear Regulatory Commission
VA	Vital Area
WAC	Waste Acceptance Criteria
WENRA	Western European Nuclear Regulators’ Association
WRNM	Wide Range Neutron Monitor

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REVISION SUMMARY

Revision #	Section Modified	Revision Summary
A	All	Initial Issuance
B	All	Tranche 5 review
C	All	Update for minor typographical errors

1 INTRODUCTION

1.1 Background

GEH's BWRX-300 is a 300 MWe water-cooled natural circulation Small Modular Reactor (SMR) utilising simple natural phenomena-driven safety systems (Figure 0-1). It is the tenth generation of the Boiling Water Reactor (BWR) and represents the simplest BWR design since General Electric (GE), GEH's predecessor in the nuclear business, began developing nuclear reactors in 1955. The BWRX-300 is an evolution of the U.S. Nuclear Regulatory Commission (USNRC) licensed 1,520 MWe Economic Simplified Boiling Water Reactor (ESBWR), as set out in 26A6642AD, "ESBWR Design Control Document, Chapter 1: Introduction and General Description of Plant, GE-Hitachi Nuclear Energy" (Reference 0-1). It is designed to provide clean, flexible energy generation that is cost-competitive with natural gas fired plants. Target applications include base load electricity generation, load following electrical generation generally within a range of 50 to 100% power, hydrogen production, district heating, and other process heat applications.

GEH entered the BWRX-300 design into Generic Design Assessment (GDA) with the United Kingdom (UK) Office for Nuclear Regulation (ONR), the Environment Agency (EA) and Natural Resources Wales (NRW) with the objective of gaining regulatory confidence on the acceptability of a conceptual full plant design for deployment in England and Wales[1]. GEH, referred to in the GDA process as the 'Requesting Party', will exit GDA at the end of Step 2.

[1] Note that, in places, this document makes reference to the UK or "UK requirements," for example when discussing certain nationally applicable law. However, only sites in England and Wales are being considered for the BWRX-300.

1.2 Purpose

The purpose of this document is to summarise GEH's main submission as part of this two-step GDA process. This main submission is made up of four volumes that address the topics of safety, security, safeguards, and the environment across the lifecycle of the BWRX-300. The submission, and its supporting references, is designed to be broad enough in scope and detail to enable a meaningful assessment by the ONR, EA and NRW.

1.3 Objective

GEH has structured its submission using the 'Claims, Arguments and Evidence' (CAE) approach that has been widely used in the licensing of recent nuclear power projects in the UK. The top-level claim, referred to as the Fundamental Objective, is provided below:

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

The Fundamental Objective is supported by four second-tier claims; one for each of the four volumes summarised in this document. These second-tier claims are then further subdivided to address discrete topic areas and aspects of the design. The claims are supported by arguments and evidence, although aspects of both of these will only become available in later licensing phases. An overview of the CAE structure is provided in Appendix A.

This submission does not intend to demonstrate that all claims are substantiated. Instead, it aims to demonstrate that there is a *viable path towards* substantiation for all claims, and that GEH's current and planned activities are consistent with achieving such substantiation. Some activities needed to support substantiation have been captured in a Forward Action Plan (FAP) (see Section 12), but others will be addressed as a matter of course as the design matures (i.e., as part of 'normal business').

1.4 Document Route Map

The remainder of this document is structured as follows:

- An overview of GEH's approach to GDA for the BWRX-300 (Section 2)
- The chapter structure of the submission, consisting of four volumes (Section 3)
- A summary of the design evolution and operating experience of BWRs (Section 4)
- A plant overview, including the BWRX-300's main features, buildings and systems (Section 5)
- A summary of the four volumes of main submission (Sections 6 to 9)
- Discussion of how the BWRX-300 is being designed for decommissioning and end-of-life (Section 10)
- Discussion of how risks are being reduced As Low As Reasonably Practicable (ALARP) and that Best Available Techniques (BAT) are being followed for environmental protection systems (Section 11)
- A summary of the Forward Actions (Section 12)
- Conclusions (Section 13)
- References (Section 14) and
- An expanded version of the CAE matrix (Appendix A).

2 APPROACH TO THE GENERIC DESIGN ASSESSMENT OF THE BWRX-300

2.1 Scope of the Assessment

The scope of the GDA was defined in the NEDC-34148P, “BWRX-300 UK GDA Scope of Generic Design Assessment” (Reference 0-2). The scope is limited to the first two steps of the GDA process but addresses all significant aspects of safety, security, safeguards, and environmental protection such that a meaningful assessment can be carried out by the regulatory bodies. The NEDC-34154P, “BWRX-300 UK GDA Design Reference Report” (Reference 0-3) defined the:

a. **Design Reference**: The final plant design reference by which the Safety, Security, Safeguards, and Environmental cases are based upon for the completion of the GDA. It lists the documents that comprise the design of the nuclear power plant that are applicable to GDA submissions

b. **Design Reference Point**: This is a plant configuration control point chosen by the Requesting Party upon which the Safety, Security, Safeguards, and Environmental cases are based. It is an interim plant design freeze representing a snapshot in time for the purposes of generating the four volumes summarised in this document. The Design Reference Report (Reference 0-3) outlines measures in place to manage design changes as the BWRX-300 progresses through each licensing phase.

2.2 Licensing Strategy

Although the first BWRX-300 projects to be announced were in Canada and the United States, the intention since its conception has been for a ‘standard plant design’ for the BWRX-300 that is suitable for deployment in any market internationally.[2] Therefore, the BWRX-300 standard plant design is based wherever possible on meeting the requirements and guidance of the International Atomic Energy Agency (IAEA) Safety Standards, recognising that there are country-specific requirements and regulatory expectations to be met that go beyond those in the IAEA Safety Standards. In England and Wales, this means, for example, meeting the legal requirements of the “Nuclear Installations Act 1965” (Reference 0-4), the “Ionising Radiation Regulations 2017” (Reference 0-5) and the “Environmental Permitting (England and Wales) Regulations 2016” (Reference 0-6). It also means addressing regulatory expectations, for example those set out by the ONR in its “Safety Assessment Principles for Nuclear Facilities” (SAPs) (Reference 0-7), Technical Assessment Guides (TAGs) and Technical Inspection Guides (TIGs). Demonstration of how the design meets such requirements and regulatory expectations is provided in the relevant chapters of the four volumes.

[2] On 4 April 2025, the CNSC issued a construction licence for the BWRX-300 at Darlington, Canada.

2.3 Generic Site

As the GDA is not for a specific site, a so-called 'generic site' has been defined in a GEH Topical Report, NEDC-34138P, "BWRX-300 UK GDA Generic Site Envelope and External Hazards Identification" (Reference 0-8). This report presents the set of generic site characteristics that are applicable to potentially suitable sites[3] for new nuclear power plants in the UK. These generic site characteristics are used as the basis for the safety analysis to be assessed during GDA. As set out in the GDA Scope Report (Reference 0-2), GDA will consider a single unit operating on a generic UK coastal site using a once-through cooling system with seawater as the normal heat sink.

The site characteristics have been further developed as part of NEDC-34219P, "BWRX-300 UK GDA Preliminary Environment Report Chapter E2: Generic Site Description" (Reference 0-10) and NEDC-34164P, "BWRX-300 UK GDA Preliminary Safety Report Chapter 2: Site Characteristics" (Reference 0-11). The generic site characteristics account for geology, hydrology, meteorology, marine and land-based biological receptors, demography and a range of other characteristics. In line with UK policy and regulatory expectations, the generic site has been defined accounting for potential changes to site characteristics due to the effects of climate change (Reference 0-8).

[3] 'Potentially suitable sites' has been taken to be those listed in the UK "National Policy Statement for Nuclear Power Generation (EN-6)" (Reference 0-9).

2.4 Structures, Systems, and Components in Scope of GDA

The scope of GDA includes all Structures, Systems, and Components (SSCs) on a generic site important for safety, security, safeguards, and environmental protection throughout the lifecycle of the BWRX-300. The Design Reference for GDA is detailed in Reference 0-3 which defines the SSCs within the scope of GDA and reflects the BWRX-300 standard plant design maturity nominally as of March 2024, although certain aspects are beyond this timeline as discussed in Reference 14-3. It also lists the system design documents, drawings and other design information relevant to each SSC. A representative site layout is shown in Figure 0-2.

There is a different level of maturity for the Power Block (comprising the Reactor Building (RB), Turbine Building (TB), Control Building (CB), Radwaste Building (RWB), Service Building (SB), Reactor Auxiliary Structures and the corresponding systems and components) than there is for the balance of plant.

GEH adopts a four-phase design process that extends from Baseline 0 (where functional requirements are defined) up to Baseline 3 (where the design is ready for construction).

- Baseline 0 (BL0): plant requirements established; high-level conceptual SSC design developed; corresponding requirements identified
- Baseline 1 (BL1): system interfaces established; integrated 3D model, Instrumentation and Control (I&C) Simulation Assisted Engineering (SAE) model, baseline Deterministic Safety Analysis (DSA), and Probabilistic Safety Analysis (PSA) to support License to Construct developed; System Design Descriptions (SDDs) developed for primary systems
- Baseline 2 (BL2): standard design completed; ready for construction planning, detailed component design, and support for equipment procurement/fabrication
- Baseline 3 (BL3): the standard design applied to specific project; all remaining system and component design completed in preparation for construction activities

Generally, the systems and structures within the Power Block are at least at BL1 whilst the Balance of Plant are at BL0.

The Power Block contains the safety classified SSCs that are within the scope of this GDA and the majority of SSCs important to environmental protection that are within the scope of this GDA. The Power Block, plus the Protected Area (PA) security fences constitute the SSCs important for security and safeguards within the scope of this GDA.

3 STRUCTURE OF THE SUBMISSION

The GDA submission is made up of four volumes that address the topics of safety, security, safeguards, and the environment across the lifecycle of the BWRX-300, as summarised below. Each volume is underpinned by supporting design information and analyses. The NEDO-34087, "BWRX-300 UK GDA Master Document Submission List" (Reference 0-12) presents the full list of documents submitted for the GDA.

3.1 Volume 1 – Preliminary Safety Report

Volume 1 is the Preliminary Safety Report (PSR). The PSR presents the 'safety case' for the BWRX-300, i.e., how safety is being addressed in the design, construction, commissioning, operation, and decommissioning of the plant. Section 6 of this document provides further detail on the objectives and contents of the PSR. The PSR chapter structure is provided below.

The structure and contents are based on that recommended in IAEA Safety Standards Series No. SSG-61, IAEA SSG-61, "Format and Content of the Safety Analysis Report for Nuclear Power Plants" (Reference 0-13), with some additional chapters to address topics typically treated in greater detail for UK nuclear power plant projects:

PSR Ch. 1: Introduction and Overview	PSR Ch. 15.4: Safety Analysis - Human Actions
PSR Ch. 2: Site Characteristics	PSR Ch. 15.5: Safety Analysis - Deterministic Safety Analyses
PSR Ch. 3: Safety Objectives and Design Rules for SSCs	PSR Ch. 15.6: Safety Analysis - Probabilistic Safety Assessment
PSR Ch. 4: Reactor (Fuel and Core)	PSR Ch. 15.7: Safety Analysis - Internal Hazards
PSR Ch. 5: Reactor Coolant System and Associated Systems	PSR Ch. 15.8: Safety Analysis - External Hazards
PSR Ch. 6: Engineered Safety Features	PSR Ch. 15.9: Safety Analysis - Summary of Results of the Safety Analyses Including Fault Schedule)
PSR Ch. 7: Instrumentation and Control	PSR Ch. 16: Operational Limits and Conditions of Safe Operation
PSR Ch. 8: Electrical Power	PSR Ch. 17: Management for Safety and Quality Assurance
PSR Ch. 9A: Auxiliary Systems	PSR Ch. 18: Human Factors Engineering
PSR Ch. 9B: Civil Structures	PSR Ch. 19: Emergency Preparedness and Response
PSR Ch. 10: Steam and Power Conversion Systems	PSR Ch. 20: Environmental Aspects
PSR Ch. 11: Management of Radioactive Waste	PSR Ch. 21: Decommissioning and End of Life Aspects
PSR Ch. 12: Radiation Protection	PSR Ch. 22: Structural Integrity
PSR Ch. 13: Conduct of Operations	PSR Ch. 23: Reactor Chemistry
PSR Ch. 14: Plant Construction and Commissioning	PSR Ch. 24: Conventional Safety and Fire Safety
PSR Ch. 15: Safety Analysis (Including Fault Studies, PSA and Hazard Assessment)	PSR Ch. 25: Security
PSR Ch. 15.1: Safety Analysis - General Considerations	PSR Ch. 26: Interim Storage of Spent Fuel
PSR Ch. 15.2: Safety Analysis - ID, Categorisation and Grouping of PIEs and Accident Scenarios	PSR Ch. 27: ALARP Evaluation
PSR Ch. 15.3: Safety Analysis - Safety Objectives and Acceptance Criteria	PSR Ch. 28: Safeguards Annex

3.2 Volume 2 – Preliminary Security Report

GEH has submitted the 006N6248, “BWRX-300 Security Assessment” (Reference 0-14). This document is similar in scope and content to a Preliminary Security Report (PSyR) but has been developed for the US and contains US Safeguarded Information that cannot be shared publicly. NEDC-34197P, “BWRX-300 UK GDA Preliminary Safety Report Chapter 25: Security” (Reference 0-15) provides a summary of the approach to security for the BWRX-300 at a level of detail appropriate for a public document. This summary will be developed into the generic security report and site-specific security report in future licensing stages. An overview of the security case is provided in Section 7of this document.

3.3 Volume 3 – Preliminary Safeguards Report

A summary of how safeguards is being addressed in the design, commissioning, construction and decommissioning of the BWRX-300 is presented in NEDC-34200P, "BWRX-300 UK GDA Preliminary Safety Report Chapter 28: Safeguards" (Reference 0-16). This Preliminary Safeguards Report (PSfR) will be developed into a site-specific safeguards report at a future licensing stage. An overview of the safeguards case is provided in Section 8 of this document.

3.4 Volume 4 – Preliminary Environmental Report

Volume 4 is the Preliminary Environmental Report (PER). The PER presents the measures being taken to protect the environment in the design, construction, commissioning, operation, and decommissioning of the BWRX-300. Section 9 of this document provides further detail on the objectives and contents of the PER. The chapter structure is as follows:

PER Ch. E1: Introduction

PER Ch. E2: Generic Site Description

PER Ch. E3: Management Arrangements and Responsibilities

PER Ch. E4: Information about the Design

PER Ch. E5: Radioactive Waste Management Arrangements

PER Ch. E6: Demonstration of BAT Approach

PER Ch. E7: Radioactive Discharges

PER Ch. E8: Approach to Sampling and Monitoring

PER Ch. E9: Prospective Radiological Assessment

PER Ch. E10: Other Environmental Regulations

4 DEVELOPMENT OF THE BWRX-300

4.1 Evolution of Boiling Water Reactor Design

The BWR has its origins in technology developed in the 1950s by Argonne National Laboratory and GE. The first BWR plant built by GE was the 5 MWe Vallecitos plant (1957) located near San Jose, California. The Vallecitos plant confirmed the ability of the BWR concept to produce electricity successfully and safely for a grid. The Vallecitos test facility success led to building the Dresden 1 plant located near Morris, Illinois. Construction of this 180 MWe plant began in 1956 with commercial power production achieved in 1960. The BWR design has subsequently undergone a series of evolutionary changes. Early changes focused on increasing power density and overall power output whilst later changes focused on simplification.

While the BWR had its genesis in the United States, it also led to developments in other countries under the Eisenhower administration's “Atoms for Peace” initiative. As examples, the Japan Power Demonstration Reactor BWR introduced nuclear power to Japan, and the KAHL Nuclear Power Plant, which employed an indirect-cycle BWR using steam generators to produce turbine steam, was the first commercial nuclear power plant built in Germany.

The advantage of the indirect-cycle was to avoid radioactive carryover from steam to the turbine, but operations from single (direct) and dual-cycle plants demonstrated that the process of boiling itself provided a natural separation process that tended to leave activated radioactive impurities in the vessel water, and this process was further enhanced with the introduction of in-vessel steam separation and a reactor water cleanup system.

During the 1950s and into the 1960s, when the BWR was in its infancy, it grew by exploring different options and design configurations and feeding back operating experience gained. As a result, standardisation did not exist to the extent that it would for future product lines of the Generation II reactors. While the various Generation I BWR configurations successfully operated for many years, operating experience along with improvements in manufacturing capabilities eventually drove toward forced circulation, internal vessel steam separation, and direct-cycle BWRs as a standard configuration.

The ever-evolving BWR design has been simplified in two key areas – the reactor systems and the containment design. describes the development of the BWR.

Dresden 1 was based on a dual steam cycle rather than the direct steam cycle that characterises later generation BWRs. Steam was generated in the reactor and flowed to an elevated steam drum and secondary steam generator before reaching the turbine. The first step in BWR simplification was elimination of the external steam drum. This was achieved by two technical innovations, the internal steam separator, and the steam dryer at Gundremmingen (KRB-A, 1967). The simplification of the design using technical innovations has been repeated with each new iteration.

The first large direct cycle BWRs (e.g., Oyster Creek) appeared in the late-1960s and were characterised by elimination of steam generators and the use of five external recirculation loops. Later plants were simplified by the introduction of internal jet pumps. These pumps boosted recirculation flow so that only two external recirculation loops were needed. Jet pumps first appeared in the Dresden-2 BWR/3 plant. BWR/4, BWR/5, and BWR/6 designs continued the path to simplification with the changes shown in Table 0-1.

The use of reactor internal pumps in the Advanced Boiling Water Reactor (ABWR) design represents another simplification. By using pumps attached directly to the vessel, jet pumps and external recirculation systems (with associated pumps, valves, piping, and snubbers) were eliminated. The ESBWR and its smaller predecessor the Simplified Boiling Water Reactor (SBWR) are a result of the logical simplification of using a taller vessel and shorter core to achieve natural recirculation flow without the use of any pumps.

BWRX-300 continues the cost-saving advances of the SBWR and ESBWR with a tall vessel design to achieve natural circulation but without the need for a shorter core. This allows the BWRX-300 to use the same fuel bundle designs in use in the operating BWR fleet. Challenges to the integrity of the system are minimised by the large water inventory above the core in the Reactor Pressure Vessel (RPV).

Figure 0-3 illustrates the evolution of the reactor system design. Most BWRs deployed to date have used forced circulation including the BWR/1s through BWR/6s and the ABWR. Natural circulation plants have a separate lineage from the Vallecitos plant through Humboldt Bay and Dodewaard to the SBWR, ESBWR, and BWRX-300.

The first BWR containments were spherical “dry” structures. Dry spherical and cylindrical containments are still used today in Pressurised Water Reactor (PWR) designs. Subsequent BWR’s including the ABWR and ESBWR utilised a pressure suppression containment design that allowed for a smaller size and the ability to accommodate rapid depressurisation of the RPV. The BWRX-300 has gone back to the dry containment configuration because Isolation Condensers (ICs) manage RPV pressure and Safety Relief Valves (SRVs) have been eliminated.

The Mark I containment used for BWR/3 and most BWR/4 plants was the first of the pressure suppression containment designs. The Mark I design has a characteristic “inverted” light bulb configuration for the steel drywell surrounded by a steel torus housing the large pool of water for pressure suppression. The conical Mark II design used for some late BWR/4 and BWR/5 plants is a less-complicated arrangement allowing simplified construction. The Mark III containment design in BWR/6 plants represented a further improvement in simplicity; the containment structure is a right-circular cylinder that is easy to construct while providing access to equipment and space for maintenance activities.

The ABWR containment is smaller than the Mark III containment because the elimination of the recirculation loops translates into a more compact containment and RB. The ESBWR containment is similar in design to the ABWR but is larger to accommodate the passive Emergency Core Cooling System (ECCS).

The BWRX-300 containment is small and simple and is achieved with surface-mounted integral reactor isolation valves to rapidly isolate the flow from a downstream pipe break to minimise pressure and temperature buildup in the containment and an Isolation Condenser System (ICS) to remove energy from the RPV rather than directing that energy into a suppression pool. Figure 0-4 illustrates the history of BWR containment (outlined in red) and RB development.

4.2 Global Experience in Boiling Water Reactor Operation

There have been over 100 BWRs built worldwide. The highest concentration of BWRs is in the U.S. where roughly one-third of the operating reactors are BWRs. Many BWRs are among the best operating plants in the world, performing in the “best of class” category. See Table 0-2 for a list of GE BWRs that have been built and Table 0-3 for a list of non-GE BWRs that have been built.

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5 PLANT OVERVIEW

5.1 Introduction

The BWRX-300 builds on the success and lessons learned from over 60 years of BWR operating history. The BWRX-300 optimises innovation with technology readiness. It relies on fuel that is already licensed and currently in use globally and used by the majority of the BWR fleet. The design optimises material and manufacturing techniques while incorporating innovative passive and simple concepts. The result is a cost-effective advanced reactor design with industry-leading safety[4] and economic performance that can be licensed and constructed in the near term.

The key simplifications of the BWRX-300 are the use of integral reactor isolation valves that mitigate the effects of Loss of Coolant Accidents (LOCAs), and large capacity ICs that provide over pressure protection without the need for SRVs. The ICs also act as the ECCS, utilise natural circulation, and require no Alternating Current (AC) power to perform their functions.

A cutaway of the BWRX-300 RB is shown in Figure 0-5. An artist's rendering of the major systems and how those systems interconnect is shown in Figure 0-6 (005N9751, "BWRX-300 General Description," Reference 0-17).

[4] Based on achieving a Core Damage Frequency significantly lower than that of today's operating BWR fleet – see Section 6.4.10.

5.2 Unique Design Features

Though mostly traditional in BWR design, the BWRX-300 includes several design features that simplify the design and support reduced costs. These features include:

1. Integral reactor isolation valves: The BWRX-300 RPV is equipped with isolation valves that are integral to the RPV that rapidly isolate a ruptured pipe to help mitigate the effects of a LOCA. All large fluid pipes with RPV penetrations are equipped with double isolation valves that are integral to the RPV.
2. No SRVs: SRVs have been eliminated from the BWRX-300 design. The large capacity ICS in conjunction with the large steam volume in the RPV provides overpressure protection in accordance with American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (BPVC), Section III, Class 1 equipment. Historically, inadvertent actuation of BWR SRVs has been the most likely cause of a LOCA but these valves have been eliminated from the BWRX-300 design.[5]
3. Dry containment: The BWRX-300 has a dry containment. Dry containments have been used for many reactor designs and can be shown to effectively contain the releases of steam, water, and Fission Products (FPs) after a LOCA.
4. No external reactor recirculation loops: Elimination of external reactor recirculation pumps and associated piping and a reimagined RPV provides a relatively large inherent reactor coolant volume and nozzle elevations significantly above the core. These features with a reliable passive ECCS provided by the ICs eliminates the need for active emergency core cooling injection systems while ensuring larger safety margins than predecessor BWRs.
5. No need for emergency diesel generators: Elimination of active ECCSs eliminates the need for onsite emergency power systems. Standby diesel generators are provided as a Defence in Depth measure and for asset protection i.e., to protect valuable equipment and systems from potential damage during power outages or other emergencies.
6. Design for simplicity: The BWRX-300 has been designed with simplicity in mind from the start, beginning with a simplified system layout that requires fewer safety systems and safety-related pools of water. This concept has also been adapted to fit with commercial building standards and cost and labour-efficient construction techniques for underground structures. The design has been optimised for constructability.
7. Use of commercial off-the-shelf equipment: Due to its smaller size, the BWRX-300 has been designed to use more commercial off-the-shelf equipment than previous BWRs. For instance, the turbine and generator models have been used on many fossil plant projects such as combined-cycle combustion turbine sites and can be used for this plant, with some small modifications.

An overview of the key BWRX-300 buildings and systems is presented in Section 5. Full details of the key systems are presented in PSR Chapters 4 to 10.

[5] The design has Ultimate Pressure Regulation (UPR) valves that are designed to work in some Design Extension Condition (DEC) complex sequences.

5.3 Key Plant and Systems

This section provides concise descriptions of some of the most important plant buildings and their primary functions, as well as the systems that are housed within each building. Figure 0-7 provides an overview of the plant layout with proposed locations of each of the buildings and structures (008N0279, "BWRX-300 Design Specification for Radwaste Building Structure," Reference 0-18 and 005N1730, "BWRX-300 Reactor Building General Arrangement," Reference 0-19). Further information on buildings and layout is provided in NEDC-34172P, "BWRX-300 UK GDA Preliminary Safety Report Chapter 9B: Civil Structures" (Reference 0-20). Further detail on the BWRX-300 is provided throughout the PSR and PER, with summaries available in NEDC-34163P, "BWRX-300 UK GDA Preliminary Safety Report Chapter 1: Introduction and Overview" (Reference 0-21) and NEDC-34221P, "BWRX-300 UK GDA Preliminary Environmental Report Chapter E4: Information about the Design" (Reference 0-22).

5.3.1 Reactor Building

The RB structure is a cylindrical-shaped building that is deeply embedded below grade, as shown in 006N7823, "BWRX-300 Primary Containment System" (Reference 0-23). The structure is primarily constructed of pre-assembled Diaphragm Plate Steel - Plate Composite modules. These modules enable simpler and greater flexibility during construction. They use a lower volume of concrete than standard reinforced concrete blocks that are typically used in RB construction, as they offer an increased shielding capability. They also achieve high strength and ductility and demonstrate improved resistance to bending and shear loads, as discussed in the 006N8670, "BWRX-300 Modularization Strategy Report" (Reference 0-24).

The primary functions of the RB are set out in the 006N6987, "BWRX-300 Reactor Building Design Specification" (Reference 0-25):

- To house and structurally support the RPV, containment structure, reactor support structures, fuel handling equipment, biological shielding, and any associated equipment and structures
- To provide adequate space for operation, maintenance, and removal of equipment housed within the containment structure during periodic maintenance
- To provide protection for equipment from environment and natural hazards, as well as internal and external hazards
- To support habitability functions of the secondary control room (SCR), such as radiological shielding, toxic gas isolation, and passive cooling for occupancy

The RB consists of the following main systems supporting power generation:

- Nuclear Boiler System (NBS), including RPV
- ICS
- Control Rod Drive System (CRD)

These main systems are supported by the following water cooling and cleanup systems:

- Isolation Condenser Pools Cooling and Cleanup System (ICC)
- Reactor Water Cleanup System (CUW)
- Fuel Pool Cooling and Cleanup System (FPC)
- Condensate Filters and Demineralisers System (CFD) (note this is situated in the TB)

The following systems only operate in specific circumstances, such as during shutdown or emergencies:

- Boron Injection System (BIS)
- Shutdown Cooling System (SDC)
- Refueling and Servicing Equipment

5.3.2 Turbine Building

The TB, as with the RB, is to be predominantly constructed of pre-assembled Diaphragm Plate Steel – Plate Composite modules, and is situated above grade (Reference 0-24). It houses the following main systems, which are critical to safe operation and power generation:

- Main Turbine Equipment (MTE)
- Main Condenser and Auxiliaries (MCA)
- Condensate and Feedwater Heating System (CFS)
- Circulating Water System (CWS)
- Plant Cooling Water System (PCW)

The Offgas System (OGS) also has a number of components situated within the TB. The components are the Offgas Recombiner, Cooler Condenser, Moisture Separator, Refrigeration Dryers, and Gas Analysers.

5.3.3 Radwaste Building

The RWB, which may be constructed of steel-plate modules or using conventional building methods (to be determined during site-specific design), is situated above grade and houses the radioactive waste management systems for the plant (Reference 0-24). The systems process effluent for re-use in the power generation cycle, discharge to the environment (if necessary, as during normal operation there are no expected aqueous radioactive discharges) or collect and prepare wastes for disposal. The RWB houses the following systems:

- Liquid Waste Management System (LWM) – Collection and Filtering, and Waste Sampling subsystems
- Solid Waste Management System (SWM)
- OGS - Offgas Reheater, Charcoal Vault, and Offgas High Efficiency Particulate Air (HEPA) Filter
- Chilled Water Equipment (CWE), (situated on the roof)

5.3.4 Control Building

The CB is to be constructed using conventional building methods (no modules or assemblies) (Reference 0-24) and houses the Main Control Room (MCR) and electrical (including batteries and uninterruptible power supplies), control, and instrumentation equipment, as described in the 006N5991, “BWRX-300 Plant Architecture Definition” (Reference 0-26).

5.3.5 Service Building

The SB has a designated area for staging of new and spent fuel to support refueling operations, an outage center, and administrative areas such as offices (Reference 0-26).

5.3.6 Primary Containment System

The primary containment is a Steel-Plate Composite Containment Vessel (SCCV), constructed using Diaphragm Plate Steel - Plate Composite modules situated within the RB. It is a vertical cylinder approximately 17.5 meters in diameter and 38 meters high. The SCCV encompasses the RPV and its pedestal, bioshield, and all associated piping, equipment, and support structures (Figure 0-8) (Reference 0-23).

The SCCV is an Engineered Safety Feature (ESF) for radiation protection and a physical, leak-tight barrier to protect against radiological releases from the NBS. It is the third and ultimate FP barrier (preceded by the fuel cladding and Reactor Coolant Pressure Boundary (RCPB)) and is also a floodable volume to ensure core coverage in an emergency. It is designed in such a way that ensures the environmental effect of any radioactive release from the plant is reduced to levels that are As Low As Reasonably Achievable (ALARA) and below the authorised limits for discharges.

Additional design features of the Primary Containment System (PCS) include:

- A nitrogen-inerted containment atmosphere during operation (provided by the Containment Inerting System (CIS)), and the ability to purge with air for access during outages. This dry containment eliminates the need for a suppression pool.
- Minimising the number of containment penetrations
- Providing Containment Isolation Valves (CIVs) external to the SCCV for containment penetrations
- Providing means for leak testing of penetrations and the PCS
- Creating a liquid-tight barrier between the open reactor vessel and upper containment during refueling activities

Further information is provided in NEDC-34168P, "BWRX-300 UK GDA Preliminary Safety Report Chapter 6: Engineered Safety Features" (Reference 0-27).

5.3.7 Containment Cooling System

The Containment Cooling System (CCS) performs the cooling function for the SCCV to maintain the containment bulk average temperature within the Environmental Qualification limits of the related equipment located inside containment, as per 006N4761, "BWRX-300 Equipment Qualification Specification" (Reference 0-28). It maintains a habitable temperature for plant personnel entering the containment during outages. It also assists with containment cooldown following a loss of offsite power and hot and cold shutdowns.

The CCS is described in 006N7777, "BWRX-300 Containment Cooling System" (Reference 0-29). It is a closed loop air or nitrogen recirculating cooling system, with a system of ducts and dampers that distribute the gas as needed throughout the SCCV. The gas is cooled by Air Handling Unit (AHU) cooling coils (which are provided with chilled water from the CWE system). All condensate from containment cooling is removed from the bottom of containment to a pressurised sump, part of the Equipment and Floor Drain System (EFS).

5.3.8 Passive Containment Cooling System

The Passive Containment Cooling System (PCCS) passively assists active containment cooling during normal operations but is not required to maintain acceptable normal operating containment temperature conditions. The PCCS, consisting of three independent cooling trains (with water provided by the FPC equipment pool), is only effective if steam discharges into the containment, such as during a LOCA. Heat that is transferred to the PCCS from containment is subsequently transferred to the equipment and reactor cavity pools (part of the FPC) (006N7941, "BWRX-300 Fuel Pool Cooling and Cleanup," Reference 0-30). There is no transfer of radioactivity to the pools in the event of a LOCA (closed loop arrangement).

5.3.9 Reactor Core and Fuel

The reactor core is located within the core shroud of the RPV containing Global Nuclear Fuel GNF2 fuel assemblies, which are used as they have low hydraulic resistance, benefitting natural circulation (See Figure 0-9, 006N1887, "BWRX-300 Fuel Design and Qualification," Reference 0-31). A total of 240 fuel bundles (consisting of zircaloy-clad fuel rods and spacers) are used, each containing a 10x10 array of 78 full-length fuel rods, 14 part-length fuel rods and two large central water rods (to increase moderation). Some full-length rods contain gadolinia to control excess reactivity.

The zircaloy cladding on the fuel rods is considered the primary FP barrier, preventing the release of radioactivity. The inclusion of part-length fuel rods in the design of the fuel bundles increases the efficiency of the fuel. Surrounding the fuel bundle is a zircaloy channel which provides a well-defined coolant flow path through the bundle. It is anticipated that the BWRX-300 fuel channels will be similar to those used in the UK ABWR, approximately 4.3 m long and 15 x 15 cm, as described in GA91-9901-0022-00001, “UK ABWR Generic Design Assessment: Radioactive Waste Management Arrangements” (Reference 0-32).

Lower tie plates are designed to promote seating into the orificed fuel supports and are fitted with a debris filter to prevent foreign material entering the flow channels and potentially damaging the fuel cladding. Spacer grids are distributed throughout to maintain fuel rod spacing. The upper tie plate is also a spacer grid and provides the fuel bundle lifting handle for fuel handling and refueling operations.

The fuel is designed in such a way to minimise the probability of fuel failure and leakage. Strict controls on reactor water chemistry are also employed to minimise the likelihood that the fuel will fail. In addition, the various cleanup systems employing filters and/or demineralisers across the plant remove any particulates and dissolved contaminants that are generated during power operation, further protecting the fuel as water is recycled. Further discussion on fuel design and aqueous effluent cleanup mechanisms is presented in “BWRX-300 UK GDA Preliminary Environment Report Chapter E6: Demonstration of BAT Approach.”

The low core power density compared with previous BWRs, enhanced natural circulation flow due to the RPV height, and high feedwater (FW) temperature maintain thermal hydraulic stability for optimum operation. Further technical information is provided in NEDC-34166P, “BWRX-300 UK GDA Preliminary Safety Report Chapter 4: Reactor (Fuel and Core)” (Reference 0-33).

5.3.10 Nuclear Boiler System

The NBS is described in “BWRX-300 Nuclear Boiler System” (Reference 0-34) and consists of three subsystems that support power generation. These are the:

- RPV
- Main Steam Lines (MSLs)
- RPV Instrumentation

Reactor Pressure Vessel

The RPV is a vertical, cylindrical pressure vessel with a minimum inside diameter of approximately 4 m, a height of approximately 26 m and wall thickness of approximately 136 mm with cladding. The active core is 3.8 m high.

It forms a major part of the RCPB (which contains all pressure-retaining components such as the RPV, piping, and isolation valves), which is the second FP barrier preventing the release of radioactivity generated within the reactor into the environment (although it does not prevent carryover of noble gases in the MSL). See Figure 0-10, 006N7827, “BWRX-300 Nuclear Boiler System,” (Reference 0-34). The RPV contains the path for reactor coolant flow through the fuel and to generate steam to drive the turbines. Flow through the core is by natural circulation, enabled by a relatively large RPV volume and tall chimney region between the top of the core and steam separators above. This design enhances safety by reducing the rate at which reactor pressurisation occurs under accident conditions.

A substantial volume of water above the core, initially provided by FW flow, ensures that reactor water level is maintained at or is above the top of active fuel, and fuel cladding temperature is maintained within normal operating limits. In emergency situations, the ICS maintains this volume to ensure core cooling (Reference 0-34). Further information is provided in PSR Chapter 4.

RPV Internals

The internal components of the RPV are briefly described below from bottom to top and can be seen in Figure 0-11 (Reference 0-34). More detailed information is provided in PSR Chapter 4.

Equipment predominantly associated with the control rods is housed below the reactor core. CRD housings, which support the weight of the CRD components (Section 5.3.11) and four fuel assemblies, are located at the very bottom. Control rod guide tubes provide a means of guiding the control rods into and out of the bottom core plate, and channel water to the rods whilst moving. Orificed fuel supports ensure proper alignment of the control rod blades during operation. The shroud support provides vertical and lateral support for the shroud (which also provides horizontal support for the core) and upper components such as chimney and steam separators.

At the bottom of the reactor core is the core plate, providing vertical and lateral support for the fuel assemblies, control rods, and instrumentation. It links the shroud and shroud support together. At the top of the reactor core is the top guide, providing further lateral support for the fuel assemblies, control rods, and instrumentation.

Bolted to the top guide is the chimney, which forms an annulus with the shroud, separating the subcooled recirculation downward flow (from the steam separators and FW makeup) from the upward steam-water mixture flow exiting the core. The chimney height ensures natural circulation is sustained without the need for forced circulation via pumps.

The chimney head and steam separator assembly form the top of the core discharge mixture plenum. This plenum provides a mixing chamber to homogenise the steam/water mixture before it enters the steam separators. Individual axial flow steam separators (consisting of standpipes and vanes) have no moving parts. Separated water returns to the downcomer annulus for recirculation into the core.

Steam dryers remove any remaining moisture from the steam prior to it exiting the reactor and entering the MSLs towards the turbine. The moisture content of the exiting steam is lower than 0.1% at full reactor power.

At the very top of the RPV is the head vent, which under startup and normal conditions, transports steam and non-condensable gases to one of the MSLs. The non-condensable gases are subsequently routed to the main condenser (Reference 0-34).

Reactor Isolation Valves

All RPV penetrations are situated a minimum of 4 m above the top of active fuel and are fitted with Reactor Isolation Valves (RIVs). These are located on the outside of the RPV (Figure 0-12), two in series, and are fitted on the main steam subsystem, CUW (which takes suction from the internal drains at the RPV bottom head), condensate supply and return lines of the ICS, RPV head vent, and FW lines. Each RIV is able to operate independently and can automatically isolate a line. They are a critical part of the overall LOCA mitigation strategy (minimisation of coolant loss by isolating the entire reactor).

Most RIVs fail closed, except the ICS condensate supply and return lines which fail as-is (normally open), as the ICS protects the reactor core during off-normal conditions. These RIVs close if a break in an ICS train is detected to prevent loss of coolant (Reference 0-34).

Main Steam Lines

Two MSLs are routed from the RPV through RIVs and CIVs (which provide isolation of these lines for line breaks) towards the turbines. A portion of the main steam is distributed to the CFS, the Moisture Separator Reheater System (MSR), and the Turbine Gland Seal System (TGSS) during normal operation. MSL drain lines provide the ability to drain condensate from the MSLs to the condenser during operation (Reference 0-34).

RPV Instrumentation

Core instrumentation consists of Local Power Range Monitors (LPRMs), Gamma Thermometers, and Wide Range Neutron Monitors (WRNMs). Aside from neutron flux measurements, the NBS is supported by additional instrumentation that measures reactor water level, pressure, temperature, core flow, and main steam flow and pressure (Reference 0-34). The in-core instrumentation arrangement is shown in Figure 0-13 (Reference 0-17). Further information is provided in NEDC-34169P “BWRX-300 UK GDA Preliminary Safety Report Chapter 7: Instrumentation and Control” (Reference 0-35) and NEDC-34225P, “BWRX-300 UK GDA Preliminary Environmental Report Chapter E8: Approach to Sampling and Monitoring” (Reference 0-36).

5.3.11 Control Rod Drive System

The CRD is defined in 006N7898, “BWRX-300 Control Rod Drive” (Reference 0-37). The 57 GEH Ultra-HD control rods, which are manufactured from boron carbide or hafnium (with minimal niobium impurity), are neutron absorbing components which provide negative reactivity into the core to allow for the control of reactor power. They are cruciform shaped elements occupying alternative spaces between fuel assemblies through the core (Figure 0-14) (Reference 0-34). Low cobalt stainless steel or nickel alloys are utilised for the control rod blades to minimise neutron activation as far as reasonably practicable. Four fuel assemblies in a cell provide guidance for the insertion and withdrawal of the control rod. The control rods are cooled by core leakage flow to remove heat generated by neutron and gamma absorptions (Reference 0-37). Further detail is provided in NEDC-34167P, “BWRX-300 UK GDA Preliminary Safety Report Chapter 5: Reactor Coolant System and Associated Systems” (Reference 0-38).

The CRD consists of Fine Motion Control Rod Drives (FMCRDs) (Figure 0-15) (Reference 0-17) to provide control rod positioning within the core, hydraulic control units to provide a diverse source of stored energy for fail-safe emergency control rod insertion (scram), and a hydraulic subsystem responsible for the distribution of high pressure water in support of normal operation and scram. A purge water header provides a continuous supply of purge water to the FMCRDs to provide cooling. This purge water flows into the reactor, adding to the overall reactor coolant inventory.

5.3.12 Isolation Condenser System

The ICS is described in “BWRX-300 Isolation Condenser System” (Reference 0-39). The ICS consists of three independent trains, each containing a heat exchanger (or IC) submerged in a dedicated pool of water and is connected to the RPV by steam supply and condensate return piping (Figure 0-16). The ICS removes heat from the reactor coolant (steam) by transferring it to the water in the pools through the heat exchanger tubes, maintaining radiological containment. The three pools are the ultimate heat sink for protecting the reactor core when the main condenser is unavailable and the RPV becomes isolated. It is also an ECCS that provides RPV overpressure protection.

IC heat removal capability is proportional to steam supply pressure. The steam is condensed on the tube side of the heat exchangers and is returned back to the RPV chimney section in a closed loop. The ICs are placed at an elevation above the steam source, creating a natural circulation effect that is driven passively by gravitational force. The IC pools are vented to atmosphere (while being monitored for radiation).

Subcooled condensate return to the RPV chimney results in a steam quenching effect, as well as lowering the pressure at the exit of the reactor core. If RPV conditions fall below the saturation point, the ICS idles until decay heat drives conditions back to saturation, automatically coming back into operation.

Condensate flow is controlled by two condensate return valves installed in parallel within the condensate return line. Both condensate return valves are normally closed, fail-open valves which are each rated for the full design flow rate of the train. The valve designs are diverse from one another to eliminate the possibility of a Common Cause Failure (CCF) that could prevent an ICS train from initiating. The condensate return piping for each train is fitted with a loop seal (Figure 0-17, 006N7345, "Isolation Condenser Pools Cooling and Cleanup System," Reference 0-40) to prevent steam bypass directly from the chimney region of the RPV to the IC (preventing reverse flow up the condensate return line).

Non-condensable gases that may accumulate within an IC are either removed by purging during system standby to the NBS MSLs or neutralised by catalytic recombination during system operation. The condensate within each catalytic recombiner drains away, allowing hydrogen and oxygen to be converted to water. The remaining non-condensable gases are abated by a separate gaseous effluent treatment system, the OGS (Section 5.3.16).

One ICS train is necessary to mitigate an Anticipated Operational Occurrence (AOO) and can provide reactor decay heat removal for 72 hours without operator action. Two trains are required for LOCA mitigation (as a result of a large line break and isolation of the RPV) to protect the integrity of the RCPB. Two trains can sustain decay heat removal for seven days without any operator involvement, and longer with IC pool inventory replenishment via mobile water sources.

In off-normal conditions, ICS Trains A and B provide the SDC with a supply flow path to the chimney. The BIS supplies Train C of the ICS with an enriched boron solution which is injected into the reactor chimney 006N7492, "BWRX-300 Isolation Condenser System," (Reference 0-39).

5.3.13 Isolation Condenser Pools Cooling and Cleanup System

Water quality is discussed in 006N6766, "BWRX-300 Water Quality" (Reference 0-41). The ICC processes water from the three IC inner pools and surrounding outer pools to maintain water temperature within design limits and purifies the water to the required reactor quality (Reference 0-41). The inner pools contain the IC heat exchangers which are cooled by the PCW. Each pool is dedicated to one ICS train and is physically separated from other pools by structural partition walls.

The ICC has overall responsibility for the water in the IC pool compartment structure. To ensure continuous safe operation in off-normal conditions, the IC outer pools can be replenished via an Emergency Mitigation Equipment (EME) connection that is external to the RB, which can be used as a long-term makeup water supply. Unidirectional pool makeup conduits also supply makeup water passively from the outer pools to the inner pools during off-normal conditions. The IC outer pools also contain suction surge tanks and a return guard pipe which prevent the draining of the pools in the event of a break in piping below the IC pools.

Each inner and outer IC pool is equipped with an atmospheric vent, which is sized appropriately to provide a means of heat rejection and pressure minimisation. The vents are fitted with louvered covers and screens to prevent foreign material from entering the pools from the environment. They are also fitted with radiation detectors to inform personnel of a potential release of radioactivity to the environment (via the steam effluent) during extended IC deployment in off-normal conditions (i.e., as a result of ICS containment failure, described in Section 5.3.12) (Reference 0-40).

5.3.14 Shutdown Cooling System

The SDC is described in 006N7708, "BWRX-300 Shutdown Cooling System" (Reference 0-42). The primary objective of the SDC system is to provide a decay heat removal pathway, in conjunction with the heat removal capacity of either the main condenser and/or the ICs, when shutting down the plant. It rapidly reduces RPV pressure and temperature from operating conditions to below saturation temperature at atmospheric pressure.

The system is also used to reduce RPV inventory during reactor startup. CRD purge water and excess reactor water volume arising from thermal expansion during heatup must be removed. SDC can also be used in conjunction with the CUW to reduce RPV thermal stratification during startup, caused by continuous input of cold CRD flow through the drives (Reference 0-42).

5.3.15 Boron Injection System

The BIS is described in 006N7417, "Boron Injection System" (Reference 0-43). The BIS introduces sufficient negative reactivity into the reactor primary system in order to assure a reactor shutdown from the full power operating condition to the cold 20°C (68°F) subcritical state with no control rod motion.

It is entirely independent from the CRD, assuring reactor shutdown (during startup or normal operation) by mixing a neutron absorber with the primary coolant. The neutron absorber is an aqueous solution of enriched sodium pentaborate decahydrate ($Na_2B_{10}O_{16}\cdot 10H_2O$), which acts as a neutron poison, halting the fission process. It is only used as a diverse and independent, manually-initiated emergency backup to normal reactor shutdown systems (e.g., control rod insertion), with the capability of holding the reactor subcritical under cold conditions.

The system is designed to prevent unintentional or accidental injection of the solution. The BIS is also isolated from the EFS floor drains with its own separate sump, to allow for hazardous waste management and disposal. This avoids boron entering the primary circuit and the need for extensive cleanup (Reference 0-43).

5.3.16 Offgas System

As described in 006N7899, "BWRX-300 Offgas System" (Reference 0-44), the OGS processes non-condensable gases from the MCA system that are produced through normal operations (Figure 0-18). The main process influent to the system is a mix of steam, air, hydrogen, and radioactive noble gases from the MCA Steam Jet Air Ejectors (SJAEs). The primary objective of the OGS is to process this influent prior to release to the environment from the Heating, Ventilation, and Cooling System (HVS) Continuous Exhaust Air Plenum (CEAP) (Reference 0-44).

Offgas processing involves two primary functions:

- Recombination of hydrogen and oxygen into water to maintain plant water inventory and reduce hydrogen explosion risk
- Controlled adsorptive holdup of the radioactive isotopes of krypton, xenon, and argon to achieve adequate decay, thereby reducing gaseous effluent radioactivity releases from the plant

6 SAFETY

6.1 Claims and Objectives

The Fundamental Objective identified in Section 1.3 is supported by the PSR claims related to safety in Figure 0-19.

This submission does not intend to demonstrate that all claims are substantiated. Instead, it aims to demonstrate that there is a *viable path towards* substantiation for all claims, and that GEH's current and planned activities are consistent with achieving such substantiation. Some activities needed to support substantiation have been captured in a FAP (see Section 12), but others will be addressed as a matter of course as the design matures (i.e., as part of 'normal business').

6.2 Safety Concept and Defence-in-Depth

The BWRX-300 is designed in line with the standards published by the IAEA and implements the IAEA concept of Defence-in-Depth (D-in-D). The concept of D-in-D involves the provision of multiple layers of defence against some undesirable outcome rather than a single, strong defensive layer. In the case of an NPP, the undesirable outcome is the exposure of workers or the public to radiation that exceeds specified limits.

To understand the principle of D-in-D, it is important to differentiate between two types of defensive layering:

1. Physical barriers in place to prevent release of radioactivity: the fuel cladding, RCPB, and containment. The integrity of one or more physical barriers must be maintained to prevent unacceptable releases.
2. Features, functions, and practices used to minimise challenges to the physical barriers, to maintain the integrity of the barriers and, in case a barrier is breached, to ensure the integrity of the remaining barriers.

The physical barriers themselves represent multiple layers of defence against radioactive releases. In the BWRX-300 D-in-D concept, shown in Figure 0-20, the physical barriers are not themselves referred to as “Defense Lines” (DLs). That term is reserved for the layers of defence comprising features, functions, and practices that protect the integrity of the barriers. In fact, the D-in-D concept is largely focused on identifying and organising features, functions, and activities into DL without explicit acknowledgment of the physical barriers. It should, however, be understood that the fundamental purpose of the layered DLs is to ensure the integrity of the layered physical barriers.

The BWRX-300 D-in-D concept uses Fundamental Safety Functions (FSFs) to define the interface between the DLs and the physical barriers.

NOTE: In each plant scenario, if the FSFs are performed successfully, then the corresponding physical barriers remain effective.

The FSFs for the BWRX-300 are:

1. Control of reactivity
2. Removal of heat from the fuel (in the reactor, during fuel storage and handling, and including long-term heat removal)
3. Confinement of radioactive materials, shielding against radiation, and control of planned radioactive releases, as well as limitation of accidental releases

The first two FSFs support the fuel cladding and the RCPB physical barriers. The third FSF supports the containment physical barrier.

If a manual action is credited to perform an FSF, the monitoring and display of plant parameters necessary to perform the manual action successfully are also considered part the of FSF. Therefore, monitoring and display of parameters are not treated as a stand-alone FSF.

Five DLs in the BWRX-300 Safety Strategy specification are consistent with the five levels defined by the International Nuclear Safety Group (INSAG) in INSAG-10, “Defence in Depth in Nuclear Safety” (Reference 0-45).

- Level 1: Prevention of abnormal operation and failures
- Level 2: Control of abnormal operation and detection of failures
- Level 3: Control of accidents within the design basis

- Level 4: Control of severe plant conditions, including prevention of accident progression and mitigation of the consequences of severe accidents (divided into DL4a and DL4b in the BWRX-300 as shown in Figure 0-20
- Level 5: Mitigation of radiological consequences of significant releases of radioactive material

Five DLs are adopted and consistent with the IAEA approach.

DL1 is centred on preventing failures by design and other conservative measures taken to minimise the potential for failures to occur in subsequent lines of defence. These design functions and conservative measures cover the design, construction, operation, and maintenance of the plant in accordance with appropriate safety margins, engineering practices, and quality levels. DL1 also includes the use of appropriate conservatism in analyses.

DL2 contains plant functions designed to control or initiate responses to Postulated Initiating Events (PIEs), especially AOOs before any plant parameters exceed conditions allowed in AOOs. Functions that normally operate to actively control plant parameters are part of DL2. Other functions, such as blocking control rod motion and anticipatory plant scrams, are also part of DL2. Functions in DL2 are assigned to Safety Category 3 and are performed by at least Safety Class (SC) 3 equipment. Functions in DL2 must be performed independently from DL3 functions, and any portion of DL2 functions subject to a CCF must be performed diversely from corresponding portions of functions in DL3.

DL3 contains plant functions that act to mitigate a PIE by preventing core damage, when possible. These functions are to provide a level of assurance for maintaining integrity of the physical barriers that prevent radiological release and to place the plant in a safe state.

DL2 and DL4a (see below) functions provide independence and diversity, to the extent practicable, from DL3 functions in mitigating events caused by a single failure and many CCFs. Because of requirements for redundancy in DL3 functions, a CCF in DL3 is generally a failure of two or more redundant and similar components occurring concurrently. These DL3 CCFs may result from causes such as errors in design or manufacturing, inadequate maintenance or surveillance, environmental factors exceeding design margins, or internal or external hazards.

DL3 also includes functions that maintain the plant in a safe condition following mitigation of PIEs until normal operations are resumed. DL3 functions typically include reactor scram and actuation of ESFs.

DL3 functions are needed when DL2 is not effective at intercepting a PIE or when a PIE is beyond the capabilities of DL2 functions. DL3 functions needed during the first 72 hours following a PIE are assigned to Safety Category 1 and are performed by SC1 equipment. If the equipment is needed to support DL3 functions but only after 72 hours or more then it can be classified as SC2. If the equipment is not required until after seven days, it can be classified as SC3. This graded approach allows SSCs that are used, for example, to replenish DL3 functions for long term heat recovery to be classified at a reduced Safety Category, given the time available post-accident to ensure those functions are available.

By eliminating the need for active support systems such as power supplies, ventilation, or cooling water, and minimising the need for active control functions such as pumps and actively controlled valves during the first 72 hours of an event, systems and equipment involved in performance of DL3 functions needed during the first 72 hours of a PIE are designed for highest reliability.

DL4 represents the boundary between Design Basis Accident (DBA) and Beyond Design Basis Accident (BDBA). DL4 comprises two subsets of functions that are designated as DL4a and DL4b functions. DL4a functions are for Design Extension Conditions (DECs) that occur without core damage, and functions for DECs progressing to core damage are DL4b functions.

DL4a and DL4b safety features specifically designed to prevent core damage and mitigate the consequences of accidents involving significant core damage are, as far as practicable, independent from DL3 systems, with a few, justified exceptions.

DL4a functions are backups to DL3 functions, in the unlikely event a DL3 function fails. DL4a functions are intended to be available to place and maintain the plant in a safe state in case of PIEs that could lead to BDBAs, such that core damage is prevented.

DL4b functions are performed in scenarios leading to core damage. DL4b functions are provided to limit the radiological releases in case of core damage and are aimed at maintaining the containment functions for extreme events, multiple events, or multiple failures that defeat DL2, DL3, and DL4a. The DL4b functions intended for mitigating DEC are functionally and physically separated from the systems intended for other DL functions. Note that safety features designated for DEC with core damage may, if practicable and available, also be used for preventing or minimising significant core damage if it can be demonstrated that such use does not undermine the ability of these systems to perform their primary functions if conditions evolve into a severe accident.

DL4b capabilities are provided to mitigate the effects from a damaged core, to preserve the FSF of confinement of radioactive material, and to limit radioactive releases to within acceptable levels. DL4b features specifically designed to mitigate the consequences of accidents with core damage are independent from systems used in normal operation or used to mitigate AOOs, as far as practicable, with justified exceptions.

DL5 includes emergency preparedness measures to cope with potential unacceptable radiological releases in case the first four DLs are not effective. These are largely off-site measures taken to protect the public in a scenario involving substantial release of radiation.

The adequacy of the DLs is assessed and demonstrated using layered, DSAs that are designed to exercise the different DLs. The PIEs to be considered in these analyses are supported based on rigorous and systematic failure modes and effects analyses of the plant systems, as well as internal and external hazard evaluations, and a Human Operation Hazard Evaluation (HOHE).

Functional and design requirements are derived from the DSAs and D-in-D concept itself, to ensure that the DL functions are implemented in the design consistent with their role in the D-in-D concept, and the credit taken for them in the safety analyses.

6.3 Classification of Structures, Systems, and Components

The BWRX-300 approach to classifying SSCs is consistent with IAEA SSR-2/1 (Rev.1), “Safety of Nuclear Power Plants: Design” (Reference 0-46, and IAEA SSG-30, “Safety Classification of Structures, Systems and Components in Nuclear Power Plants” (Reference 0-47). Classification of SSC is conducted to identify the importance of the SSC with respect to safety.

Classification of SSCs provides a means for applying appropriate design requirements and establishes a graded approach in the selection of materials, and application of Codes and Standards (C&S) used in design, manufacturing, construction, testing, and inspection of individual SSCs.

The BWRX-300 approach to classifying SSCs by SC is based primarily on deterministic methods and is directly traceable to the safety functions performed by the SSC. A fundamental element of the BWRX-300 SSC classification approach is the direct correlation between the DLs in which an SSC performs a function, and the relative safety importance of that function. Functions are categorised into three safety categories, Safety Category 1, Safety Category 2, and Safety Category 3, with Safety Category 1 being the most important.

Primary functions are those that directly perform the FSFs in support of DL2, DL3, DL4a, or DL4b. Safety Categories are applied to the primary functions as follows:

1. Safety Category 1 is assigned to DL3 primary functions. DL3 functions assure the integrity of the barriers to release, place, and maintain the plant in a safe state, and provide independence and diversity for all DL2 and DL4a functions caused by a single failure (and many CCFs). Accordingly, DL3 primary functions are the most important from a safety standpoint.
2. Safety Category 2 is assigned to DL4a primary functions. Both DL2 and DL4a provide a redundant means to address PIEs (generally independent of DL3 functions) and are therefore important from a safety standpoint, although less important than DL3 functions. DL4a functions are a backup to DL3 functions, in the unlikely event a DL3 functions fails, and therefore have a higher consequence of failure than DL2 functions and are more important from a safety standpoint than DL2 functions.
3. Safety Category 3 is assigned to DL2 and DL4b primary functions as they are relatively the least important. DL4b functions address severe accidents, which are extremely unlikely because failure of both DL3 and DL2 or DL4a functions would have to occur. Accordingly, DL4b functions are considered relatively the least important DL functions, despite the high consequence of failure.
4. Non-Safety Category is assigned to all other functions.

Safety Class is assigned to components based on the safety category of the functions they perform, as defined in 005N9461, “BWRX-300 Structures, Systems, and Components Safety Classification” (Reference 0-48):

- Safety Class 1 (SC1) is assigned to SSC that perform a Safety Category 1 function.
- Safety Class 2 (SC2) is assigned to SSC that perform a Safety Category 2 function.
- Safety Class 3 (SC3) is assigned to SSC that perform a Safety Category 3 function.
- Non-Safety Class (SCN) is assigned to all other SSC.

6.4 Summary of the Safety Analysis

6.4.1 Purpose

As defined by the IAEA in Requirement 4 of IAEA Safety Standards Series No. GSR Part 4, “Safety Assessment for Facilities and Activities” (Reference 0-49), “the primary purposes of the safety assessment shall be to determine whether an adequate level of safety has been achieved for a facility or activity and whether the basic safety objectives and safety criteria…have been fulfilled.”

It is therefore the role of the safety analysis[6] to confirm that the BWRX-300 DL approach outlined in Section 6.2 has led to a design that meets the acceptance criteria, such as those for Core Damage and Large Releases. As the design process is iterative, the initial results from the safety analysis feedback in to inform the design. Safety analysis topics include internal hazards, external hazards, DSA, PSA, radiological consequence analysis, and Human Factors Engineering (HFE).

6.4.2 Approach

An introduction to the safety analyses of the BWRX-300 is presented in NEDC-34179P, “BWRX-300 UK GDA Preliminary Safety Report Chapter 15.1: Safety Analysis General Considerations” (Reference 0-50). It includes a description of the scope of the safety analysis and the approach adopted (i.e., conservative or realistic) for each plant state, from normal operation to DECs with core melt.

A ‘design to analysis’ approach has been followed, where design and safety evaluation mature in a stepwise manner. The safety analysis is used to derive requirements, which then feed back into design development and detailing. It is an iterative process, as shown in Figure 0-21 and described in full in the NEDC-33934P, “BWRX-300 Safety Strategy” (Reference 0-51).

A critical part of the safety analysis is to systematically and comprehensively identify SSCs functional and human failures that initiate a PIE, initiate a hazard that leads to a PIE or worsen a hazard that leads to a PIE. The BWRX-300 Safety Strategy (Reference 0-51) process identifies two plant-level failure analyses to identify such failures:

- Functional Failure Analysis (FFA)
- Human Failure Event Analysis (HFEA)

FFA identifies failures of plant systems or equipment with potential to cause a challenge to an FSF. These hazards are identified in Failure Modes and Effect Analyses (FMEAs) performed on the plant systems.

The FFA is limited to random single failures and to CCFs. The system FMEAs are reviewed to identify failures that cause challenges to FSFs. A consolidated list of failures from all system FMEAs can then be generated and organised. Potential functional failure hazard PIE sources are organised by quantitative frequency, using the frequency ranges defined in the Safety Strategy

The HFEA identifies erroneous decisions or human action(s) that lead to an unplanned plant transient. Human operations hazards typically involve unplanned changes to plant equipment status by equipment operators or maintenance personnel.

The scope of each failure analysis encompasses the complete range of normal plant states (i.e., full power, low power, shutdown and refueling) as the type and consequence of each failure may differ depending on the plant state.

[6] For the purposes of this document, the terms ‘safety analysis’ and ‘safety assessment’ are used interchangeably.

The analyses also consider all sources of radioactivity (e.g., spent fuel, fuel being handled, radioactive waste, activated material) in addition to the reactor core itself. The output of the plant-level failure analyses (i.e., potential PIE initiators) is passed on to the fault evaluation process, where PIEs are systematically identified, categorised, and grouped together

In addition to failure analyses, hazard evaluations are performed to ensure that all potential hazards the plant might experience are identified and considered. The BWRX-300 Safety Strategy (Reference 0-51) identifies two hazard evaluations:

- Internal Hazard Evaluation (IHE)
- External Hazard Evaluation (EHE)

A primary objective of each hazard evaluation is to identify hazards with the potential to initiate a PIE and pass that list of hazards to the relevant downstream analysis, PSA and/or the deterministic hazards analyses (e.g., seismic hazard analyses, fire hazards assessment, pipe rupture hazards analysis). The focus of the EHE and IHE is identifying the list of credible hazards and defining the expected frequencies of those credible hazards. The plant is designed to withstand the hazards while maintaining performance of the FSFs through implementation of DL1 design requirements.

Following PIE and event sequence selection, DSA is performed. The DSA objectives include:

- Demonstrating the design meets the acceptance criteria established following a graded approach for each plant state. The graded approach application may lead to acceptance criteria more restrictive for events with higher occurrence probability.
- Provide analytical basis to support the derivation of the plant technical specifications for normal operation.
- Provide analytical basis for establishing and validating accident management procedures and guidelines.

The safety analyses scope plant state includes the following plant states:

- Normal operation
- AOOs
- DBAs
- DECs with or without core damage (i.e., BDBAs)

The BWRX-300 DSA uses a layered analysis approach that includes three types of DSA evaluations:

- Baseline – DSA (BL-DSA)
- Conservative – DSA (CN-DSA)
- Extended – DSA (EX-DSA)

These DSA acceptance criteria are discussed in NEDC-34181P, “BWRX-300 UK GDA Preliminary Safety Report Chapter 15.3: Safety Objectives and Acceptance Criteria, GE-Hitachi Nuclear Energy” (Reference 0-52). The DSA results are compared against the applicable plant state acceptance criteria and dose limits.

PSA is performed to complement the DSA. PSA estimates the overall risk presented by the facility that is compared to the acceptance criteria specified in NEDC-34184P, “BWRX-300 UK GDA Preliminary Safety Report Chapter 15.6: Probabilistic Safety Assessment” (Reference 0-53).

6.4.3 Acceptance Criteria

The DSA and PSA AOO event sequences acceptance criteria are based on or derived from ensuring that the Specified Acceptable Fuel Design Limits (SAFDLs) are met for the following USNRC 10 Code of Federal Regulations (CFR) 50, Appendix A General Design Criteria (GDC) (Reference 0-54):

- GDC-10, “Reactor Design”
- GDC-12, “Suppression of Reactor Power Oscillations”
- GDC-17, “Electric Power Systems”
- GDC-20, “Protection System Functions”
- GDC-25, “Protection System Requirements for Reactivity Control Malfunctions”
- GDC-26, “Reactivity Control System Redundancy and Capability”
- GDC-33, “Reactor Coolant Makeup”
- GDC-34, “Residual Heat Removal”

The DSA DBA Event Sequences acceptance criteria are based on or derived from ensuring that the 10 CFR 50.46(b) (Reference 0-55) acceptance criteria for ECCSs are met.

Detailed consideration of the BWRX-300 Safety Objectives is presented within NEDC-34165P, “BWRX-300 UK GDA Preliminary Safety Report Chapter 3: Safety objectives and design rules for SSCs” (Reference 0-56).

Further detail on the derivation of the acceptance criteria is provided in PSR Subchapter 15.3 (Reference 0-52), with the criteria provided in full in TablesTable 0-4 to Table 0-6 below.

Probabilistic Safety Goals

GEH recognises that the ONR expects UK site licensees to present information that allows ONR’s assessors to judge performance against the numerical targets in the SAPs (Reference 0-7), noting that some of these targets represent legal limits related to radiation exposure.

ONR acknowledges in the SAPs that a safety case does not necessarily require detailed calculation for each target and that intermediate targets such as Core Damage Frequency (CDF) and Large Release Frequency (LRF) can be considered provided that “…the overarching Principles EKP.1 to EKP.5 are not compromised through such approaches.”

The overarching Engineering Key Principles (EKPs) are listed below:

- EKP.1 (Inherent safety) - The underpinning safety aim for any nuclear facility should be an inherently safe design, consistent with the operational purposes of the facility.
- EKP.2 (Fault tolerance) - The sensitivity of the facility to potential faults should be minimised.
- EKP.3 (Defence in depth) - Nuclear facilities should be designed and operated so that defence in depth against potentially significant faults or failures is achieved by the provision of multiple independent barriers to fault progression.
- EKP.4 (Safety function) - The safety function(s) to be delivered within the facility should be identified by a structured analysis.
- EKP.5 (Safety measures) - Safety measures should be identified to deliver the required safety function(s).

The BWRX-300 has adopted stringent CDF and LRF targets of 1E-6 per reactor-year and 1E-7 per reactor-year respectively (see Table 0-6), along with the application of radiological protection principles to ensure that normal operational exposures are reduced to levels that are ALARP.

There is a strong alignment between the BWRX-300 safety philosophy and the EKPs. This gives confidence that the application of stringent intermediate CDF and LRF targets combined with the radiological protection principles and safety strategy for the BWRX-300 will ensure that legal limits are not exceeded and that risks can be demonstrated to be tolerable and ALARP.

For GDA, therefore, the PSR continues to use these intermediate targets and provides a demonstration of risk in the context of CDF and LRF. It is the intention that in the next licensing phase a set of numerical targets will be established that are based on the targets presented in the SAPs. The general principle will be to establish targets equivalent to the BSL combined with the requirement for the risks to be ALARP. This is captured in the FAP (see Section 12).

6.4.4 Design Basis Conditions

The BWRX-300 design basis conditions are normal operations, AOOs and DBAs described below:

1. Normal Operation is operation within specified operational limits and conditions, as set out in NEDC-34185P, "BWRX-300 UK GDA Preliminary Safety Report Chapter 16: Operational Limits and Conditions for Safe Operation" (Reference 0-57). The objective of the normal operation safety analysis is to demonstrate that DL1 measures are effective in preventing abnormal operations and failures, thus meeting radiological requirements.
2. AOOs are deviations from normal operation that are expected to occur at least once during the operating lifetime of the reactor facility. The objective of the AOO safety analysis is to demonstrate that DL2 functions are effective for most AOO PIEs in meeting the applicable acceptance criteria.
3. DBAs conditions are identified as deviations from normal operations that are less frequent and more severe than AOOs. An objective of DBA safety analysis is to demonstrate that DL3 functions are effective in mitigating events and meeting the applicable acceptance criteria.

Acceptance criteria applicable to the DSA for each plant state is discussed in PSR Subchapter 15.3 (Reference 0-52). The response to AOOs and DBAs is achieved by SSCs specifically designed to mitigate these events and are assigned DL2 and DL3 functions (see PSR Chapter 3: Safety Objectives and Design Rules for SSCs (Reference 0-56).

6.4.5 Design Extension Conditions without Core Damage

DSA is performed for DECs without core damage demonstrating that releases of radioactive material are kept within acceptable limits and support the PSA determination of no core damage.

DEC analysis include:

- Multiple failures defined as complex sequences identified in the Level 1 PSA or as a PIE with a CCF
- AOO and DBAs with postulated failures of DL2 and DL3 functions analysed in EX-DSA. For these events, the DBA acceptance criteria are used as screening criteria to the evaluation of core damage.
- Low frequency events

- Non-reactor fault sequences (fuel pool accidents) are analysed in Level 1 PSA

The results of the DSA for DECs without core damage are discussed in NEDC-34187P, "BWRX-300 UK GDA Preliminary Safety Report Chapter 15.9: Safety Analysis Summary of Results" (Reference 0-58). The analysis of DECs with core damage are addressed in the Level 2 PSA described in PSR Subchapter15.6 (Reference 0-53).

6.4.6 Design Extension Conditions with Core Damage

DECs with core damage are referred to as Severe Accidents and involve a catastrophic failure, core damage, and FP release. A severe accident is generally considered to begin with the onset of core damage. To the extent that core damage is not practically eliminated, representative DECs with core damage are postulated to provide inputs for the containment design and safety features ensuring containment functionality. This set of accidents is considered in the design of corresponding safety features for DECs and represents a set of bounding cases. Accident scenarios considered for practical elimination are described in PSR Subchapter 15.9 Appendix 15A (Reference 0-58).

Severe accident sequences are selected that identify representative core damage scenarios and corresponding plant damage states that are used as the basis for performing the Severe Accident Analysis (SAA). The scope of severe accident scenario selection corresponds to sequences involving significant core damage that could lead to a containment breach and radioactive release analysed in the Level 2 PSA in PSR Subchapter 15.6 (Reference 0-53). The selected severe accident scenarios are included in a fault evaluation.

The SAA goal is to provide input to accident management for terminating the progression of core damage, maintaining containment integrity as long as possible, and minimising on-site and offsite radioactive material releases. Halting core damage progress prevents RPV failure.

The response to severe accidents considers the use of safety and non-safety, permanent and temporary systems and equipment that are beyond their originally intended functions.

6.4.7 Internal Hazards

Internal hazards are hazards that arise from within the site boundary and that result in the failure of operations or facilities that are under the control of the operating organisation. The IHE identifies conditions originating within the boundaries of the site and with potential to lead to an unplanned plant transient. The internal hazard condition does not directly challenge an FSF (like in Functional Failure Hazard Evaluation (FFHE)), but the effects of the hazard may cause equipment failures. These equipment failures are then evaluated in the DSA and PSA.

Internal hazards include:

- Fires
- Explosions, missiles from rotating or pressurised equipment
- Collapse of structures/falling objects
- Pipe whip, jet effects, and flooding

The IHE addresses both individual hazard sources and combinations of sources.

The sources of internal hazard or combinations are organised by quantitative frequency as potential PIEs and evaluated in the fault evaluation.

FFHE identifies failures of plant systems or equipment with potential to cause a challenge to an FSF. These hazards are identified in FMEA performed on the plant systems.

The HOHE identifies erroneous decisions or human action(s) that lead to an unplanned plant transient. Human operations hazards typically involve unplanned changes to plant equipment status by equipment operators or maintenance personnel.

Many human operations hazards produce the same effects as corresponding equipment failures and the effects of these are included in the FFHE. The HOHE is limited to a single erroneous act that may lead to multiple system responses. The HOHE focuses on identifying unique hazards such as an operator initiating a group command on multiple actuators that is beyond what is considered in a single failure analysis of a particular system.

6.4.8 External Hazards

The EHE includes natural and human-induced hazards that originate from a source that is not under control of the nuclear power plant license holder. The EHE addresses individual hazard sources and combinations of sources:

- Natural external hazards include earthquakes, droughts, floods, high winds, tornadoes, tsunami, and extreme meteorological conditions
- Human-induced external hazards include toxic gas releases, aircraft crashes, or ship collisions

External events are site-specific and are specified in the site evaluation provided in PSR Chapter 2: Site Characteristics (Reference 0-11).

Once the external hazards are identified, the BWRX-300 structures are designed to withstand these external hazards, and the resulting protection against these hazards. PSR Chapter 6: Engineered Safety Features (Reference 0-27) includes control room habitability sensors, alarms, and monitors for human-induced external hazards such as toxic gas.

The sources of external hazard or combinations are organised by quantitative frequency as potential PIEs evaluated in the fault evaluation.

6.4.9 Practical Elimination

Practical elimination is applied to events or sequences of events leading to or involving core damage (a severe accident) where confinement of radioactive materials cannot be reasonably achieved. Event sequences that are either physically impossible or extremely unlikely to occur are considered for practical elimination.

The practical elimination demonstration is performed with accident conditions and phenomena knowledge and is substantiated by relevant evidence.

6.4.10 Probabilistic Safety Assessment

Subchapter 15.6 – Probabilistic Safety Assessment (Reference 0-53) reports how a Level 1 PSA has been developed for internal events in all modes of operation with a Level 2 PSA for full power. Full power hazard Level 1 PSAs, including internal fire, internal flooding, seismic, high wind and heavy load drop have been developed, with some Level 2 analyses for certain hazards. A spent fuel pool PSA has also been produced and is discussed. The chapter is supported by a summary report and methodology report, which go into more detail regarding the assumptions, input data, task outputs and analysis of the results.

The chapter demonstrates that the PSA results have been, and will continue to be, used to risk-inform and support design optioneering to ensure that the risk is ALARP. In addition, given the low risks calculated from the analysis to date, it is expected that the final risk results will continue to show the site risk to be very low relative to traditional safety goals and numerical targets.

The PSA is an iterative process and will continue to be developed as the design develops. The scope and level of detail in the PSA is commensurate with the stage of design development and with a two-step GDA. There are Forward Actions related to PSA, including future work commitments such as the development of numerical dose and risk-based targets against which a full scope Level 3 PSA will be assessed.

The development of the BWRX-300 PSA is an iterative process as more detailed design information becomes available and more analyses are performed. As such, the current PSA results do not present the full site risk from a full scope PSA. They do however show the order of magnitude of expected risk, which can be seen to be very low compared to traditional BWR plants. At this stage in design development, the most important use of the PSA is to provide risk insights to inform design.

The overall calculated risk is very low compared to historical CDF values calculated for existing plants. There is significant margin to the targets typically applied by IAEA member states as documented in IAEA-TECDOC-1874, IAEA-TECDOC-1874, “Hierarchical Structure of Safety Goals for Nuclear Installations” (Reference 0-59).

The total plant CDF, including Fuel Damage Frequency from the spent fuel pool contribution, is 8.73E-07/yr. The early results from the PSA have been used to inform the design, as discussed in PSR Subchapter 15.6 and in Section 11 of this document.

6.4.11 Human Factors Engineering

If a manual operator action plays a role in performing an FSF, the monitoring function of the equipment used to display key plant parameters that are necessary for the operator to perform the manual action successfully are also considered part of the FSF.

The overall goal of the BWRX-300 HFE programme is to reduce the risks and consequences related to human interactions with the plant throughout all phases of the lifecycle. Human actions are defined as human-machine interactions that are credited in the BWRX-300 DSA, PSA or SAA. NEDC-34182P, “BWRX-300 UK GDA Preliminary Safety Report Chapter 15.4: Human Actions, GE-Hitachi Nuclear Energy” (Reference 0-60) describes the approach to identification, modeling and substantiation of these human actions. However, it is not the intention for the PSR to provide detailed substantive analysis of the human actions. That analysis will be developed later in the safety case programme.

The HFE programme takes a proportionate approach to the design and substantiation of human actions based on their level of risk. The programme of HFE activities and analysis informing the design of the plant SSCs is based on clear definition of the full plant set of users and a clearly defined scope of application across the full plant design, operational modes, and lifecycle stages, with focus on important human actions. The human actions will be assigned risk levels based on the following principles:

- Any human actions that are credited in the DSA will be assigned a high-risk level.
- Where the PSA identifies human actions as being risk significant based on measures of risk importance these will also be assigned a high-risk level.
- The remainder of the human actions modeled in the PSA will be assigned a medium-risk level. The risk level determines the HFE application level that will be applied to the human action and the HFE application level defines the graded work scope.

At present, no human actions are credited in the DSA. In relation to the PSA, the analysis is not mature enough to provide the required insights into the risk significance of human actions. The evaluation of human actions will be conducted in an iterative manner throughout the system design lifecycle. If future iterations of the DSA do credit human actions, the Risk Levels will be assessed. They will then be re-assessed for changes with each subsequent revision of the DSA. Similarly, once the PSA is baselined, the important human action risk levels will be assessed and then re-assessed with each subsequent revision.

In relation to the design, additional reviews of the appropriateness of the HFE application level will be undertaken for each human action by considering:

- Complexity of the action

- Anticipated complexity and constraints of the human-machine interface
- Complexity of the system
- Frequency of the task
- Physical environment
- Cognitive environment
- Novelty of the action, system, or human-machine interface technology
- Time sensitivity of the action

This will ensure an appropriate and integrated treatment of the human actions both in the safety analysis and in the design.

6.4.12 Occupational Dose

An estimated occupational dose for the BWRX-300 plant during normal operations has been derived from the BWRX-300 source term data and applying relevant Operational Experience (OPEX). This is a significant element in supporting the facility design and methods of operation to ensure occupational radiation exposures are ALARP. The analysis of the occupational dose is proportionate to the design maturity of BWRX-300.

The estimated annual occupational exposures are considered for six activity categories:

- Radioactive waste handling
- Normal maintenance
- Special (unscheduled) maintenance
- Refueling
- In-service inspection
- Operation and surveillances

The dose assessment is dependent on estimates for dose rates in various occupied areas, frequency of operations, and person-hours for the activities in the six dose assessment categories. The collective worker doses reported are high confidence upper boundary estimates expected based on the current conceptual design, and future design iterations are expected to establish a basis to reduce the estimate.

The collective dose estimate gives an indication of the radiological conditions at a plant and are, therefore, often used by plant operators and regulators to assess the overall performance of the plant operation in relation to radiation protection. Industry occupational exposure data is extracted from the 14-61. “Occupational Exposures at Nuclear Power Plants, Annual Report of the Information System on Occupational Exposure (ISOE) Programme” (Reference 0-61) and compared to the BWRX-300 collective worker dose estimate.

The BWRX-300 collective worker dose estimate of 491 person-mSv/year, albeit conservative, is significantly lower than the average collective worker doses reported at operating BWR reactors over the most recent reporting period (2009 to 2019) and is slightly lower than the average collective worker doses reported in PWRs over that period. When relaxations such as credit for remote operations of process equipment or lower dose rates based on calculations are applied, the BWRX-300 annual estimate will likely be lower than the average collective worker doses observed at operating BWR, PWR, and CANDU reactors (Figure 0-22).

6.4.13 Safety Analysis Results

PSR Subchapter 15.9 Safety Analysis - Summary of Results (Reference 0-58) shows that implementation of the D-in-D concept ensures multiple, independent layers of protection against unacceptable radiation releases. The chapter concludes that for all the bounding AOOs, DBAs, or DEC Events Without Core Damage analysed, none lead to the exceedance of the acceptance criteria summarised in Section 6.4.3. The PSA shows that overall risk is very low compared to historical CDF values for existing plants. There is significant margin to the probabilistic targets typically applied by IAEA member states. At this stage, the results of the safety analysis are preliminary and will mature along with the design development of the BWRX-300. The need for future work has been identified and is captured in the FAP.

7 SECURITY

7.1 Claims and Objectives

The Fundamental Objective identified in Section 1.3 is supported by the claim related to security in Figure 0-23.

This Level 1 claim for security directly links to the Unifying Purpose Statement (UPS) described in the ONR's "Security Assessment Principles for the Civil Nuclear industry" (Reference 0-71), known as the 'SyAPs', and acts as the basis of strategic intent for delivery of a robust informed design, that is measurable in accordance with the standards required in the UK.

This submission does not intend to demonstrate that all claims are substantiated. Instead, it aims to demonstrate that there is a *viable path towards* substantiation for all claims, and that GEH's current and planned activities are consistent with achieving such substantiation. Some activities needed to support substantiation have been captured in a FAP (see Section 12), but others will be addressed as a matter of course as the design matures (i.e., as part of 'normal business').

7.2 Protective Security

Protective security is provided through a combination of a security organisation (including armed personnel), physical barriers, controlled access to the PA, controlled access to Vital Areas (VAs) located within the PA, and administrative policies and procedures for screening and monitoring personnel and material allowed access to the site.

All VAs are located within the PA and within the RB. Much of the vital equipment is within containment which is inaccessible during operation and typically only accessed during refueling intervals and to which access is monitored and controlled. The location of VAs within the RB provides a second physical site barrier and means of access control.

The D-in-D concepts of redundancy and physical separation of redundant systems, as well as simple passive safety systems, further support the physical security of the plant in that multiple vital SSC must be compromised to realise effective radiological sabotage.

All vital systems and components are housed within robust reinforced concrete structures that can only be accessed through a minimal number of normally locked access points that are controlled and monitored by the site security system. Many of the components of vital systems are located below site grade, thereby minimising exposure to external threats.

The BWRX-300 site consists of an Exclusion Zone where public access is restricted; within which is the PA measuring approximately 200 meters by 160 meters, which is a zone further restricted to employees and approved visitors. The PA boundary consists of a physical barrier with an isolation zone on either side of that barrier and detection systems to monitor and assess attempts by persons to cross the barrier.

The PA surrounds all VAs and serves to limit access to important SSC to only persons who have been properly vetted and have a need for access. A visitor access programme to enact due-diligence and order of entry rules, will be implemented to allow unvetted persons who have a valid need to enter the site to be escorted by qualified personnel. The PA perimeter consists of multiple systems which fulfil several security purposes.

To support the BWRX-300 reactor design to be licensable in multiple countries GEH has developed a single bounding proxy Design Basis Threat (DBT) that establishes a set of credible characteristics, capabilities, and techniques for the theft or sabotage of Nuclear Material or Other Radioactive Material.

The GEH proxy DBT, to which the BWRX-300 design is subjected, must meet the DBT detailed criteria for all the potential countries of deployment. The goal of the proxy GEH DBT is to create assurance that the country of licensure’s DBT is achievable within the standard design, and so a move to the country specific DBT from the bounding GEH proxy DBT for detailed and site-specific design is achievable.

The BWRX-300 standard design has undergone systematic, detailed security design reviews to identify potential weaknesses and pathways within the scope of the GEH proxy DBT that could be exploited. This enabled a security informed and improved design that is cognisant of the proxy generic threat.

The BWRX-300 protective security systems are guided by an iterative and ongoing design process that incorporates changes in threat, evolution of identified vulnerabilities, continuous improvement, and advances in standard physical and cyber protection approaches, systems, and technologies.

The BWRX-300 development included a security informed design approach from the early stages of concept design that uses sound engineering principles to have minimum effect on cost and maximum effect on the support of security outcome performance.

GEH used a Security by Design (SyBD) process that involves security vulnerability reviews during plant design in order to resolve proxy DBT and security issues at the most appropriate phases of the design work stage. Placement and number of doors, wall thicknesses to optimise resistance to breaching, and equipment placement to facilitate better target set diversity were all achievable as security was integrated at an early design stage.

The goal of SyBD is to minimise the operational and maintenance costs of security through better utilisation of SSC (including diverse locations) to provide a significant deterrent and defensive benefit versus additional reliance on extrinsic security controls and armed personnel response.

GEH adopts a holistic approach to security where each aspect of security (Deter, Detect, Delay, Deny, Respond, Defend, insider threat) builds on and amplifies other aspects as a means to disincentivise the selection of a BWRX-300 reactor as a target; as well as to increase the effectiveness of the defence and time for onsite or offsite armed responders to intercept intruders before damage leading to severe consequences can be caused.

The BWRX-300 design limits the ability of malicious individuals to cause damage to key systems. This, along with the inherent slower accident progression of the BWRX-300 reactor, reduces or eliminates the reliance on immediate onsite armed responders to prevent substantial offsite radiological releases, which allows for longer-term offsite source response for interdiction and neutralisation. During the design process, the following design enhancements have been made to improve the ability to defend the site against malevolent acts:

- The number of entrances to the RB was minimised while maintaining emergency exits for personnel safety.
- The ICS cooling water pools were moved such that they are no longer in contact with external walls where they were vulnerable to draining by external breaching.
- The Spent Fuel Pool was moved such that it is no longer in contact with external walls where it was vulnerable to draining by external breaching.
- The Spent Fuel Pool walls were thickened and steel clad on both sides of the walls to be substantially more robust against breaching taking into account the quantity of high explosive allowed for in the proxy DBT.
- RB wall construction utilises Diaphragm Plate - Steel Plate Composite Construction, which has substantially better resistance to explosive breaching.
- ICS piping was redesigned to be inaccessible by routing directly from containment to the ICS heat exchangers in the ICS cooling water pools to eliminate a potential exposure to malevolent action.
- Cable routing for critical systems was diverted, to the extent practical, to route directly into containment and minimise the number of locations available for malevolent actions.
- Key doors and access hatches were upgraded to be substantially more robust against explosive breaching. Security credited doors are designed to be equally robust to the walls in which they are located.
- Large ducts and openings were enhanced to maintain the same robustness to breaching as the walls in which they are located.
- Bulk deliveries and hazardous material deliveries were moved outside the PA to reduce the opportunity for introduction of hidden explosive devices.

The security design provides for a strong and resilient defence, predominantly through passive methods, as a means to minimise operation and maintenance costs (e.g., concrete walls, heavy steel doors, and underground facilities). Where active features are used, such as surveillance systems, access control systems, and automatic door closers, the lifetime maintenance and replacement costs are considered in optimising the overall lifetime cost of power.

A layered access control strategy is used to limit access to equipment and components based on the equipment's relative significance to the overall protective strategy.

7.3 Cybersecurity

GEH implements strong cyber security programmes to control the product development lifecycles for all disciplines susceptible to cyber security issues, across both the Computer Based Security Systems and the I&C technology platforms. GEH's product security programme is based on common industry standard frameworks, with the objective of achieving high assurance that unauthorised access to the protection, control, and adjustment systems of the BWRX-300 is prevented.

The GEH cyber security programme is designed to protect the BWRX-300 design and associated standard plant envelope from a cyber-attack or event. GEH initialised this comprehensive programme at the early phases of planning and so allows the licensees to take credit for the cyber security programme and the security features designed into the BWRX-300 systems. BWRX-300 incorporates advanced cyber security principles by leveraging industry standards within the product development, procurement, and deployment lifecycle of the BWRX-300 and its information, communications, and automation systems.

Cyber security has become a critical consideration for the design and usage of digital control systems. These systems, including Computer Based Systems Important to Nuclear Safety, Computer Based Systems Essential to Safe Operations, and Computer-Based Security systems computers, including network communication systems are adequately protected against cyber-attacks up to and including threat categorisation within DBTs, through application of SyBD and D-in-D principles in a cyber security defensive architecture, which extends over the entire equipment lifecycle.

The 006N6731, "BWRX-300 Plant Cyber Security Plan" (CySSP) (Reference 14-72) is designed to protect the BWRX-300 digital I&C systems and associated standard plant envelope from a cyber-attack or event. GEH initialised the CySSP at the early phases of planning, to steer the design of critical systems to ensure D-in-D by implementing proportionate controls to reduce cyber risks to as low as practicable. The CySSP is a conservative set of standards that are consistent with both North American and international standards.

The BWRX-300 CySSP incorporates cyber security principles and recognised good practice throughout the development lifecycle while ensuring regulatory compliance. The objective of the CySSP is to achieve a high assurance that unauthorised access to the protection, control, and adjustment systems of the BWRX-300 is prevented. The main steps of the framework being the following:

- Identify cyber assets and classify them using a graded approach.
- Implement cyber security controls to protect critical essential Assets from cyber security events.
- Apply and maintain a defensive cyber security architecture protective strategy to ensure the capability to identify, protect, detect, respond, and recover from cyber events.
- Ensure that the functions of protected assets identified are not adversely affected due to cyber events.

The wider plan has been designed to align with required cyber security programme elements from the NRC, Canadian Nuclear Safety Commission (CNSC), and IEC to create a global cyber security programme for digital I&C. As cyber security guidance, recognised best practices and regulatory requirements will continue to evolve over time, GEH is committed to taking new requirements into account at the time of licensing submittal.

The defensive cyber security architecture to deliver D-in-D of the I&C network architecture is based on recognised good practice from IEC 61513:2011 and IEC 62859. Establishing cyber security boundaries on groups of systems with similar safety and security significance provides D-in-D, with the aim to delay and disrupt unauthorised lateral movement across the network and provide the ability to detect suspicious activities. In addition, utilisation of unidirectional communication controls further reduces the opportunity for unauthorised lateral movement.

The cyber security methodology used to restrict communication flows between the security levels is based upon the Biba-integrity model and ensures that communication flows unidirectionally from the highest significance security level to the lowest using fail-secure, deterministic communication pathways.

This integrity model takes a nuclear centric approach to protecting cyber assets by prioritising the integrity of systems important to safety over all other systems. The separation of networks into Security Levels allows for the enhanced capability to detect, prevent, delay, mitigate, and recover from cyber-attacks.

Security Levels and Security Zones play two different roles in the Defensive Cyber Security Architecture:

- Security Levels are a high-level grouping of systems based on their common cyber security control requirements and importance to plant protection.
- Security Zones are a more granular segmentation of systems and their networks, and the tightly coupled communications that are essential for the system to perform its critical functions.
    - Security Zones should be self-contained and able to function independently and locally even if the surrounding Security Zones are offline.
    - Security Zones have defined Security Zone boundaries.
    - Security Zones, their Zone boundaries, and required network communications are defined, documented, and maintained for every control system in that system’s SDD.

8 SAFEGUARDS

8.1 Claims and Objectives

The Fundamental Objective identified in Section 1.3 is supported by the claims related to safeguards in Figure 0-24.

This submission does not intend to demonstrate that all claims are substantiated. Instead, it aims to demonstrate that there is a *viable path towards* substantiation for all claims, and that GEH's current and planned activities are consistent with achieving such substantiation. Some activities needed to support substantiation have been captured in a FAP (see Section 12), but others will be addressed as a matter of course as the design matures (i.e., as part of 'normal business').

8.2 Safeguards by Design

Nuclear material safeguards are measures to prevent the proliferation of nuclear weapons by ensuring that states do not divert nuclear materials or technologies from peaceful uses to a clandestine nuclear weapons program. In most countries, safeguards are verified by the IAEA under a Comprehensive Safeguards Agreement, a Voluntary Offer Arrangement or an Item-Specific Arrangement.

Many of the activities identified with relation to nuclear safeguards capability requirements will not occur until a prospective operator has obtained the necessary site license and become the licensee/duty holder.

For GDA, GEH intend to demonstrate that nuclear safeguards have been considered in the design, i.e., Safeguards by Design (SBD), and that the BWRX-300 can meet the expectations and requirements for potential deployment in the UK for relevant UK and international legislation. A more detailed description is provided in PSR Chapter 28 – Safeguards (Reference 0-16).

Safeguards are applied to Qualifying Nuclear Facilities within the UK. The UK's withdrawal from Euratom as a consequence of Brexit led to both the renegotiation of the Voluntary Offer Agreement and Additional Protocol with IAEA and a requirement for new domestic Safeguards Regulations. "The Nuclear Safeguards Act (NSA) 2018" (Reference 0-73) granted the Secretary of State (i.e., the responsible Government Minister) power to enact "nuclear safeguards regulations." These are:

1. The Nuclear Safeguards (EU Exit) Regulations 2019 (NSR19) which ensures that qualifying nuclear material, facilities, or equipment are only available for use for civil activities (whether in the UK or elsewhere).

2. "The Nuclear Safeguards (Fissionable Material and Relevant International Agreements) (EU Exit) (Amendment) Regulations 2021 (2021/492)" (Reference 0-74) which give effect to provisions of various international agreements.

Qualifying nuclear material is defined as:

1. fissionable material specified in the Safeguards regulations as:
    A. i.e., material designated by the Secretary of State as fissionable material for the purposes of the definition of "qualifying nuclear material".
2. source material in the form of:
    A. uranium metal, alloy or compound, or
    B. thorium metal, alloy or compound, or
3. ore containing a substance from which a source material falling within paragraph 2 is capable of being derived.

The following activities are within the scope of the nuclear safeguards assessment of the BWRX-300 design:

- Receipt of fresh fuel
- Transfer of fresh fuel to the storage pool
- Transfer of fresh fuel from the storage pool to the reactor core
- Transfer of irradiated fuel from the reactor core to the storage pool and vice versa

- Export of spent fuel from the storage pool for dry storage
- Dry storage of spent fuel on-site
- Management of waste (potentially) bearing fissile material

SBD is the process of including the consideration of international Safeguards throughout all phases of a nuclear facility project. SBD does not introduce new requirements but rather presents an opportunity to facilitate the cost-effective implementation of existing requirements. A voluntary best practice, SBD allows for informed design choices that optimise economic, operational, safety, and security factors, in addition to international Safeguards.

The BWRX-300 is a tenth generation BWR technology, therefore safeguards requirements are well established for practices of nuclear material accountancy and prevention of material diversion. In addition, the choice of a well-established fuel design provides confidence that nuclear material safeguarding principles can be demonstrated for BWRX-300.

9 ENVIRONMENT

9.1 Claims and Objectives

The Fundamental Objective identified in Section 1.3 is supported by the PER claims related to environmental protection in Figure 0-25.

This submission does not intend to demonstrate that all claims are substantiated. Instead, it aims to demonstrate that there is a *viable path towards* substantiation for all claims, and that GEH's current and planned activities are consistent with achieving such substantiation. Some activities needed to support substantiation have been captured in a FAP (see Section 12), but others will be addressed as a matter of course as the design matures (i.e., as part of 'normal business').

9.2 Radioactive Waste Management and Discharges

The early-stage justification of the radioactive waste management arrangements for the BWRX-300 provides a demonstration of how the BWRX-300 can be developed to integrate with UK radioactive waste management and wider environmental protection requirements. Because GDA Step 2 represents a single unit design in a non-specified location it does not yet demonstrate that the design has been fully optimised for UK deployment.

The overall environmental objective is to demonstrate that the design of the BWRX-300 SMR has been optimised to reduce environmental effects to ALARA throughout the whole lifecycle (construction, commissioning, operation, and decommissioning).

BWRX-300 plant design features prevent, or when this is not practicable, minimise the creation of radioactive waste and spent fuel (radioactivity and quantity). The generation of radioactive waste during the operation of the BWRX-300 is undesirable due to (i) the potentially harmful effects of exposure to radiation for workers, members of the public and the environment, and (ii) the time, trouble and cost incurred in its management. Refer to NEDC-34222P, "BWRX-300 UK GDA Preliminary Environment Report Chapter E5: Radioactive Waste Management Arrangements" (Reference 14-62) for further details.

As a tenth-generation BWR the evolution of the BWRX-300 design has sought to avoid the generation of radioactive waste at source. Where this has not been practicable, efforts have been made to minimise the activity and quantity of radioactive waste that will require subsequent management and disposal by permitted means. This is evidence of applying the waste hierarchy and a demonstration of the application of BAT.

As discussed in NEDC-34222P, "BWRX-300 UK GDA Preliminary Environment Report Chapter E6: Demonstration of BAT Approach" (Reference 0-63), an example of such a design feature is in the design and manufacture of fuel and how it gives a low rate of fuel failure. The fuel assemblies present the largest source of radionuclides that are created as a result of nuclear fission in the reactor. Collectively these radionuclides are referred to as FPs. Any release of FPs from the fuel into the steam circuit or cooling pool water has the potential to become radioactive waste that will ultimately require treatment and/or discharge to the environment. Ensuring that these FPs remain in the fuel and its cladding is a key element of the design and operation of the BWRX-300, and the single most important factor in preventing the generation of radioactive wastes.

The manufacturer of the fuel for the BWRX-300, Global Nuclear Fuel, is engaged in a long-standing and comprehensive programme of work to improve the performance of its products and to reduce the frequency of fuel failures.

Radioactive waste management systems are present to minimise the activity and quantity of radioactive waste where it is not practicable to prevent generation (Reference 0-22). This includes:

- Gaseous radwaste management - the OGS is a state-of-the-art feature that holds up the release of noble gases enabling them decay to low levels before being released to the environment
- LWM, and minimal release philosophy – state-of-the art liquid waste management design that minimises liquid releases
- Reactor coolant cleanup systems – the CFD is a state-of-the-art design that capture contamination and lowers effluent releases
- The Process Radiation and Environmental Monitoring System provides continuous and periodic monitoring to allow determination of the content of radioactive material in various gaseous, liquid process, and effluent streams

The PER presents arrangements for the management of radioactive wastes arising from the commissioning, operation, and subsequent decommissioning of a BWRX-300 SMR, including considerations for the management of spent fuel[7] as radioactive waste. As set out in the CP1009; DESNZ CP1009, "Civil Nuclear: Roadmap to 2050," current UK government policy is that spent fuel will not be reprocessed and will therefore be designated for future geological disposal in the UK (Reference 0-64).

The BWRX-300 SMR design has evolved from the previous ESBWR and ABWR designs but is primarily influenced by the design simplifications introduced for the ESBWR. This has resulted in corresponding simplifications in systems that produce radioactive wastes and on the resultant radioactive wastes themselves.

The radioactive wastes generated through the commissioning, operation, and decommissioning of the BWRX-300 are similar in nature to those generated by predecessor BWR designs. Therefore, a baseline confidence in radioactive waste management and discharges aligning with UK expectations can be derived from the assessment of the UK ABWR. However, at this early stage of development, the BWRX-300 design does not fully reflect alignment with UK expectations and requirements for the compliant management of radioactive wastes arising from its commissioning and operation. These aspects will form part of future design development.

The generation of radioactive waste is a direct and inevitable consequence of the use of nuclear energy for power generation. The generation of radioactive waste is justified by balancing the benefits of generation of large quantities of low carbon electricity, and wider beneficial societal effects, with the detriments associated with managing, storing, and ultimately disposing of the radioactive wastes generated.

The NEDC-34228P, "BWRX-300 UK GDA, Integrated Waste Strategy" (Reference 0-65) provides a broad justification against the principle of sustainability, and, specifically, shows how new nuclear build supports achievement of the relevant United Nations Sustainable Development Goals described in "The 17 Goals: Sustainable Development" (Reference 0-66).

The wastes associated with the commissioning, operation, and decommissioning of the BWRX-300 are similar to those produced by the ABWR, but with a few notable differences. The wastes will arise through four primary means:

- Treatment and purification of aqueous fluids (reactor coolant, condensate, FW, pools, and plant drainage systems) giving rise to secondary (wastes arising as a result of materials coming into contact with radioactive substances) wet solid wastes
- Generation of SF
- Operational, maintenance and repair/refurbishment activities giving rise to several waste types
- Decommissioning, giving rise to a final batch of spent fuel and a range of decommissioning wastes

At this early stage, the BWRX-300 design comprises the power block which houses the reactor, turbine, radwaste, control, and service buildings and this is presented as a generic design capable of being deployed anywhere in the world. In order for the design to integrate in a UK context, and to align with UK regulatory expectations, additional capabilities will be required outside of the power block to provide additional support functions, such as radioactive waste processing and storage and spent fuel storage.

Because these capabilities are not included within the power block design and require tailoring to fit appropriate UK strategy and context, they are presented as an indicative scope which

[7] 'Spent fuel' is nuclear fuel that has been irradiated.

will be subject to more detailed consideration, design development, and associated justification and substantiation at the site-specific development stage. It is recognised that considerations, such as siting multiple units on a site, are likely to significantly affect decision making relating to these aspects and it is therefore appropriate to present them as indicative only at this stage.

Waste process diagrams are presented in PER Chapter 5 (Reference 0-62) which show the full lifecycle 'cradle to grave' strategy for the management of each waste stream from generation to eventual disposal, and clearly state the provision of waste management capabilities within and outside of the power block (defined as definite scope and indicative scope). Detail of certain aspects that cannot be confirmed at this stage (for example repackaging of spent fuel into a final disposable form) are excluded from scope and clearly identified on the waste process diagrams.

Indicative scope will be subject to future development and decision making, including the establishment of relevant BAT and ALARP justifications.

<u>Waste Categories</u>

In the UK, radioactive waste is "any substance or object that has no further use, and is contaminated by, or incorporates, radioactivity above certain levels defined in UK legislation".

Radioactive waste is then classified according to how much radioactivity it contains and the heat that this radioactivity produces. The broad definitions relate to the disposal requirements and designate radioactive wastes as 'Higher Activity Waste' (HAW) or 'Lower Activity Waste' (LAW).

- HAW comprises High Level Waste (HLW), Intermediate Level Waste (ILW) and a small fraction of Low Level Waste (LLW).
- LAW comprises both LLW and Very Low Level Waste.

The waste categories are then further defined in relation to the specific disposal requirements associated with UK radioactive waste management policy.

In addition to the above categories, where wastes that have the potential to be exposed to radioactive contamination or neutron irradiation can be shown to contain radioactivity at levels below the requirement for them to be managed as radioactive waste, these wastes can be classified as out of scope of regulatory control (i.e., the wastes are not considered radioactive for purposes of UK legislation), and are termed Out of Scope wastes.

<u>Higher Activity Wastes</u>

The use of high efficiency back-washable filters and deep bed demineralisers throughout the design has reduced the number of wet solid radioactive waste streams to two, namely;

- Filter backwash sludges
- Spent bead resins

At present, based on the source term in 008N0133, "BWRX-300 Solid Waste Management System - Contained Source Activity" (Reference 0-67), these wet solid wastes align to the UK radioactive waste classification of ILW, which necessitates on site management and eventual disposal to Geological Disposal Facility (GDF), inferring a requirement for on-site processing and storage capabilities. The current source term is highly conservative and has been derived to present a bounding case for the purposes of radiation protection and shielding design.

The BWRX-300 design incorporates enhanced design aspects that have the potential to further reduce radioactivity in these waste streams. These include.

- Use of improved GNF2 fuel – this is anticipated to result in a lower incidence of fuel cladding failures that will have a positive impact on all downstream source term values.
- Increased use of stainless steel throughout the design – this is anticipated to result in reduced corrosion and erosion particulate generation throughout the plant.
- Reduced cobalt inventory - the BWRX-300 material selection strategy focusses on reducing cobalt inventory wherever practicable throughout the plant design.
- Enhanced water chemistry regime – this is anticipated to result in further reduction of corrosion and erosion particulate as above, and minimisation of cobalt deposition on coolant facing surfaces.

These design enhancements are presently difficult to quantify in terms of their contribution to an improved radwaste End User Source Term (EUST). The designs for on-site wet solid wastes processing and storage should not be finalised until the full effectiveness of these design improvements have been established (i.e., after the first wastes have been generated by operation of a BWRX-300).

BWRX-300 design is predicated on US requirements, and the radioactive waste management arrangements provide the interface to the US radioactive waste classification scheme and related disposition criteria. It is recognised that there are differences between US and UK radioactive waste management policy and the design of a UK BWRX-300 will require alignment with UK requirements. Assessment of the source term for wet solid radioactive wastes (Reference 0-67) indicates that they would meet the UK criteria of ILW, resulting in a requirement for on-site processing and storage pending availability of a national GDF.

Other HAW streams arising from operation of the BWRX-300 are very similar to those previously considered for the UK ABWR.

- Spent fuel bundles
- Irradiated control rods
- Irradiated instrumentation assemblies

These are considered to present no significant issues for management and disposal to a GDF. Spent fuel disposal requirements are awaited from NWS before firm decisions can be reached on the precise arrangements for packaging for disposal. Irradiated wastes will initially be managed as HLW in a similar manner to spent fuel and will subsequently be recovered and repackaged as Dry Solid ILW (DSILW) after an appropriate decay storage period.

Lower Activity Wastes

LAW streams arising from operation of the BWRX-300 are very similar to those previously considered for the UK ABWR;

- Spent Heating, Ventilation, and Air Conditioning (HVAC) filters
- Effluent Filter Modules
- Heterogeneous dry solid wastes

These are considered to present no significant issues for management and disposition as LAW. Waste management arrangements will be implemented to ensure compliance with the Low Level Waste Repository (LLWR) Waste Acceptance Criteria (WAC) defined in WSC-WAC-OVR, “Waste Services Contract: Waste Acceptance Criteria – Overview” (Reference 0-68) and to ensure appropriate application of the waste hierarchy to optimise disposal. Wastes will be appropriately segregated, packaged, characterised, sentenced, and consigned in accordance with the requirements of the relevant LLWR WAC document.

Future Responsibilities

During a future site-specific development phase decisions will be required which are likely to include exploration of the economic viability of a plant and considerations of important aspects such as multiplication of units to form a nuclear licensed site. Specifically in relation to spent fuel and radioactive waste management arrangements, this may include determination of appropriate strategies for the on-site management of waste from multiple units, taking into account economies of scale, demonstration of an optimised approach, construction timings, and required design integration activities.

The following key aspects are therefore considered the future responsibilities of an organisation undertaking a UK site-specific development of the BWRX-300.

1. Considerations of the implications of a multiple unit site with respect to BAT, optimisation, facility integration and sizing, construction timings, etc.
2. Technical decision making and related demonstration of BAT and optimisation relating to:
    - Spent fuel management, including siting, sizing, and form of dry cask storage
    - HLW management, including integrated storage with spent fuel casks, decay storage period, and provision of future DSILW processing capability
    - Wet Solid ILW management, including decision making on the choice of final waste container, immobilisation method and form of ILW storage facility (i.e., shielded or unshielded), and store sizing taking account of potential additional requirement for storage of packages arising from processing of decay stored DSILW – control rods etc.
    - Arrangements to align DSILW management with the UK LLW waste services framework, including on-site segregation, packaging, and characterisation requirements to enable application of the waste hierarchy and compliance with relevant LLW WAC (Reference 0-68)

Decommissioning Wastes

Decommissioning of the BWRX-300 is assumed to be conducted in a similar manner to that applied to existing LWRs:

- Post Operational Clean Out (POCO)
- Reactor coolant circuit decontamination
- RPV dismantling
- POCO of fuel pool, equipment pool and reactor cavity
- Balance of plant dismantling and demolition

POCO wastes will comprise further wastes analogous to those produced during operation:

- Final core offload of spent fuel
- All control rods
- All reactor instrumentation assemblies
- Final batches of filter backwash sludges and spent resins, HVAC filters, etc
- Offload of spent charcoal from OGS delay beds
- Contaminated and irradiated plant and equipment from pools, including fuel storage racks

The actual sequence of waste arisings will be dependent on the decommissioning strategy and sequence applied.

Following POCO, it is assumed that the coolant circuit will undergo appropriate in-situ decontamination using an intrusive chemical/mechanical process that generates a concentrated sludge/particulate waste stream. If a liquid transport medium is used as a component of the applied in-situ decontamination technique this will also need to undergo appropriate abatement, resulting in further sludges and potentially further spent resins/filtrates/filters etc.

It is anticipated that RPV dismantling will focus on the establishment of a cutting plan that enables resultant metal sections to be segregated according to waste category. This may also entail decay storage strategies dependent on levels of activation of the cut sections. Because the BWRX-300 utilises a taller chimney design, it is assumed that some RPV components (moisture separator, steam dryer, etc.), located further from the neutron flux, will be less irradiated than similar components in previous designs. This may reduce the volume of decommissioning HAW.

Fuel storage racks in the fuel pool will undergo a degree of neutron irradiation from storage of irradiated spent fuel. It is anticipated that these may be irradiated to ILW levels and would be managed as a decommissioning ILW stream.

All other wastes arising from decommissioning are assumed to meet UK LAW disposal criteria and will be managed in line with prevailing disposal requirements at the time.

Discharges

Annual average gaseous and liquid effluent releases for long-term normal operation of BWRX-300, including AOOs, are found in NEDC-34225P, “BWRX-300 UK GDA Preliminary Environment Report Chapter E9: Prospective Radiological Assessment” (Reference 0-69).

The public dose from gaseous and liquid effluent releases during normal operation (and any AOOs) is expected to be well below regulatory limits.

As the operating decisions on liquid discharges have not yet been determined, three scenarios are presented for liquid and gaseous discharges under normal operations over a year. The methodology used can also be found in NEDC-34223P, “BWRX-300 UK GDA Preliminary Environment Report Chapter E7: Radioactive Discharges” (Reference 0-70).

It should be noted that the discharge assessments will require further refinement (after Step 2 of the GDA) to generate a refined model Primary Source Term (PST) and EUST. Once refined PST and EUST are determined then updated assessments of gaseous and aqueous liquid discharges, headroom factors, and proposed discharge limits can be presented. GEH fully expect the discharge activities to reduce once refined EUST and the aqueous liquid discharge volume are confirmed.

Assessment of the current fault list for the BWRX-300 has not resulted in the identification of any relevant AOOs. The current safety analysis relates primarily to reactor faults and, as such, faults that could primarily result in fuel damage. All of the faults listed present adequate mitigation through design and therefore do not give rise to environmental effect consequences within AOO frequency.

The environmental effects from accidents are not within the scope of GDA for environmental regulators.

10 DECOMMISSIONING AND END-OF-LIFE

The requirements to reduce risks to ALARP and to reduce environmental effects to ALARA also apply in the decommissioning phase of a plant’s life. During GDA, GEH intends to provide confidence that this can be achieved for the BWRX-300. Although likely many years from now, experience shows that decommissioning risks can be reduced if this activity is considered during design. As part of the GDA process and future licensing and permitting activities in England and Wales, GEH plans to demonstrate that the BWRX-300 can be safely decommissioned using today's technology applying BAT principles.

The BWRX-300 design accounts for the decommissioning process, incorporating best practice and allowing for the development of new technologies. NEDC-34193P, "BWRX-300 UK GDA Preliminary Safety Report Chapter 21: Decommissioning and End of Life Aspects" (Reference 0-75, sets out a more detailed description of the decommissioning strategy with expected timelines for a decommissioning and end-of life phase. 006N8745, "BWRX-300 Incorporation of Decommissioning in Design Considerations" (Reference 0-76) discusses how decommissioning has been considered during design.

International OPEX has demonstrated that BWRs can be readily decommissioned, in compliance with regulations and safety principles, and most likely will be subject to enhanced methods in comparison to ongoing and completed decommissioning projects. The small modular design, materials choices, SSC design, and benefit of improved decontamination and dismantling techniques are expected to further reduce decommissioning complexity.

The responsibility for planning decommissioning will lie with the future site licensee. However, at this stage, a viable case has been made that this plan can be implemented, based on the BWRX-300 design and worldwide OPEX. A decommissioning strategy will be selected early in the BWRX-300 SMR life cycle, to form the basis for planning for decommissioning and facilitate achieving the desired end-state of the decommissioning project. The following are the established strategy timing options for consideration (as recognised by the IAEA Safety Standards):

- Immediate (prompt) decommissioning:
  - decontamination, dismantling, and clean-up occurs without delays
- Deferred decommissioning:
  - the facility is placed in a period of care and maintenance (C&M) followed by decontamination, dismantling and clean-up.
  - to conduct activities directed at placing certain buildings or facilities, locations, or sites in a safe and secure interim end-state, followed by a period of safe storage before decontamination, dismantling and/or clean-up

Decommissioning strategies are evaluated in a systematic and traceable fashion so that the relative advantages and disadvantages of each strategy can be objectively compared. In-depth studies will be performed as warranted over the life cycle of the plant, to refine and solidify the recommended decommissioning strategy to account for the following issues, which may have relevant consequences for decommissioning (Reference 0-76):

- Changes in site conditions, or incidents and events
- Changes to the proposed decommissioning objectives
- Changes to ownership or management structure
- Advances in decommissioning technology
- Significant modifications to the facility, location, or site
- Updated schedule, cost, and funding information
- OPEX and lessons learned

- Revised regulatory requirements
- Availability of facilities, locations, or sites for the management of radioactive waste

OPEX demonstrates that decommissioning of reactor facilities is facilitated if considered during the design phase. Assessment of future facility decommissioning and dismantling activities at the design phase includes consideration of OPEX gained from the decommissioning of existing facilities, as well as those facilities that are in long-term safe storage.

The following are considerations for future plant decommissioning and dismantling activities:

- Minimisation and control of activation products, incorporating the prevention of contamination of surfaces and structures, and the removal of activation products from process streams
- Appropriate materials selection to minimise activation, allow ease of decontamination and minimise the amount of higher risk (corrosive/toxic/hazardous) waste
- Enabling effective on-site dismantling and decontamination through facility design and practices
- Considering safe ease-of-access for decommissioning when determining the site layout
- Designing for the management of radioactive waste generated during both operation and during decommissioning, including minimising the mixing of waste streams and enabling the characterisation of waste. Consideration should also be given to the effect of new facilities being built or existing facilities being expanded.

Design features relevant to decommissioning, in-keeping with the design principles, are discussed in the sub-sections below (based on information provided in Reference 0-76):

<u>Site Plot Plan</u>

The BWRX-300 design has been optimised for constructability, with a focus on simplifying safety systems and the incorporation of fewer pools of water. A compact design, which minimises below ground excavation and buried utilities, beyond those required for the reactor building, may be beneficial for dismantling the facility during decommissioning. Space allocation is considered to accommodate construction activities (e.g., access area around the plant, areas for laydown), thereby facilitating the ability to decommission and dismantle the plant once a license for decommissioning the plant is granted.

<u>Modular Construction</u>

A modularisation strategy will be used to design and construct the BWRX-300 at the designated site. The module and skid assemblies are intended to be built off-site, transported to the site, and erected on-site. This modularisation strategy will provide guidance in selection of disassembly methods employed during decommissioning.

<u>Control of Materials During Design</u>

Specific guidance has been exercised for plant systems materials to minimise corrosion products during plant operation as a design requirement. This provides restrictions regarding the use of cobalt-based alloys and cobalt in stainless steel and nickel-based alloys. These restrictions reduce personnel dose exposure during plant operation and decommissioning activities. Materials choices have considered minimising corrosion products, including protecting non-stainless steel equipment with a non-corrosive layer to aid decontamination.

11 ALARP EVALUATION, BAT DEMONSTRATION AND DESIGN OPTIMISATION FOR SECURITY AND SAFEGUARDS

11.1 ALARP

11.1.1 What is ALARP?

Based upon the requirements of the “Health and Safety at Work etc. Act 1974” (Reference 0-77), it is necessary to show that the risks to the workers and the public are ALARP. This requires that all reasonable measures are taken in the design, construction and operation of the plant to minimise the radiation dose received by workers and public, unless such measures are grossly disproportionate to the risk avoided. The ALARP methodology and evaluation are provided in NEDC-34199P, “BWRX-300 UK GDA Preliminary Safety Report Chapter 27: ALARP Evaluation” (Reference 0-78).

In simple terms, the concept of ALARP is a requirement to take all reasonably practicable measures to reduce risk. Sometimes, the risk owner must carry out a detailed analysis to identify the reasonably practicable measures. In many commonly encountered situations, it is well understood what measures are needed to reduce a risk to ALARP. The collection of measures that when applied to a situation usually lead to risks being reduced to ALARP is referred to as the ‘Relevant Good Practice’, or RGP. ONR considers standards, approaches or guidance to be RGP if they have judged compliance with it as a means of satisfying the law. Sources of RGP include:

- Guidance within Approved Codes of Practice; for example, the Provision and Use of Work Equipment Regulations 1998
- ONR guidance including ONR’s SAPs, TAGs and TIGs
- Standards produced by standards making organisations, for example British Standards Institution (BSI), International Electrotechnical Commission (IEC), IAEA and Western European Nuclear Regulators’ Association (WENRA)
- Guidance agreed by a body representing an industrial/occupational sector
- Well defined and established standard practice adopted by an industrial/operational sector

The development of RGP and standards includes ALARP considerations so in many cases meeting them can be sufficient, although there remains a legal requirement to take additional reasonably practicable measures that would further reduce risk. Where standards and RGP are less evident or not fully applicable, the risk owner must identify and implement measures to reduce risk up until the costs of any additional measures (in terms of money, time or trouble – i.e., the sacrifice) would be grossly disproportionate to the further risk reduction that would be achieved (the safety benefit).

In addition to the concept of ALARP in nuclear safety, UK regulations require that nuclear operators must maintain all radioactive discharges to the environment at a level which is ALARA. This includes consideration of all relevant factors such as protection of the environment and other social and economic effects.

11.1.2 ALARP in the Two-Step GDA Process

Demonstrating that risks have been reduced to ALARP is the highest-level safety objective for the BWRX-300 design, as summarised in the Level 1 Safety Claim (see Section 6.1). As introduced in Section 1.2, **this submission does not intend to demonstrate that all claims are substantiated**. Instead, it aims to demonstrate that there is a *viable path towards* substantiation for all claims, and that GEH’s current and planned activities are consistent with achieving such substantiation.

For safety, this means that the submission—this document, the PSR chapters, and the supporting references—does not aim to demonstrate that risks have been reduced to ALARP, and no 'ALARP justification' or 'ALARP evaluation' is offered. Instead, the submission presents a *viable path towards* such a demonstration. This is not a shortfall on the part of the Requesting Party; GEH believes the arguments and evidence provided within PSR Chapter 27 – ALARP Evaluation (Reference 0-78) show that efforts have been taken where practicable to a) optimise the design to reduce risks, b) to record this process, and c) set out next steps. By demonstrating that a suitable ALARP process has been established and that the organisation is capable of implementing the process by providing examples (see PSR Chapter 27), GEH aims to provide confidence that there is a path to reaching an 'ALARP position'.

The level of detail in PSR Chapter 27 will increase as licensing progresses beyond the two-step GDA and the design and supporting analyses mature. Note that the level of design maturity is not uniform across the plant (see Section 2.4), and that this also applies to the maturity of the 'ALARP evaluation'. Should GEH take the BWRX-300 through to site licensing and construction, a mature demonstration will be made available to the regulatory bodies.

GEH has developed an iterative, three-step process to demonstrating ALARP:

- Phase 1: Holistic review of BWRX-300
- Phase 2: Specific review of potential improvements
- Phase 3: Holistic evaluation of ALARP position

GEH's intent is to complete Phase 1 as part of the two-step GDA, with Phases 2 and 3 being completed in subsequent licensing stages. Note that following Phase 3, the BWRX-300 Design Reference is re-baselined and the ALARP evaluation iterates back to Phase 1. This iterative approach embeds the ALARP principle in the design development process, maximising the opportunities for risk reduction and minimising the likelihood of options being foreclosed.

11.1.3 Reducing Risk by Design

GEH has developed a set of design principles that emphasise the elimination of hazards and general reduction of risk. The design accounts for OPEX from the large operating fleet of BWRs and applies RGP such as in reducing operational exposure through improved plant layout and task design for operator tasks. GEH's ALARP approach includes consideration of the following four aspects:

- Demonstration that international OPEX has been taken into account in the overall design philosophy and in specific system designs
- Demonstration that RGP has been applied, including C&S comparison/justification
- Identification and evaluation of options (Optioneering)
- Risk assessment, as a way of understanding the significance of the issue to the holistic demonstration of ALARP, i.e., to demonstrate that the design has no disproportionately large risk contributors, and that shortfalls against numerical targets have been reduced ALARP

Where the above approaches identify reasonably practicable improvements that reduce risk, these are then considered for incorporation into the design. A proportionate assessment and optioneering process is undertaken for identified shortfalls, identifying potential improvements that could be implemented, weighed against appropriate criteria, and applying engineering judgement to select option(s) proportionate to the risk level. A robust and traceable decision making process is established within GEH processes.

The BWRX-300 design choices have been informed by reducing the largest contributors to risk in the PSA, increasing passive safety features and a reduction in the requirement for intervention by plant operators.

Although occurring mostly outside the UK's health and safety legal framework, the evolution of BWR design over the decades (see Section 1) implicitly addressed the ALARP principle by incorporating OPEX, emerging technologies, and RGP to improve safety. The BWR/1 to BWR/6 development pathway culminated in the ABWR as the pinnacle of active safety. The number of RPV penetrations below top of active fuel were reduced and eventually eliminated, with the number and complexity of recirculating loops also reducing before being finally eliminated by the move to reactor internal pumps in the ABWR. The SBWR/ESBWR development path took this one step further by enabling natural circulation and thus eliminating the requirement for forced circulation.

Although there are many more, some key risk-reducing design simplifications that occurred during the BWRX-300 design evolution include:

- Adopting a single reactor cooling loop, which runs through the RPV/core and also through the turbines, thus eliminating the need for steam generators
- Use of a taller vessel to achieve effective natural recirculation flow without the use of any internal pumps or external recirculating loops (and their associated pumps, valves, piping and snubbers)
- Unlike SBWR/ESBWR, BWRX-300 retains a full-height core such that standard BWR fuel can be used.
- Internal steam separator and dryer which eliminates the need for an external steam drum
- Challenges to the system are minimised by the large water inventory above the core in the RPV.
- The fuel assemblies (including fuel rods and channels), control rods, chimney head, steam separators, steam dryer, and incore instrumentation assemblies are removable when the reactor vessel is opened for refuelling or maintenance. These items build on designs from the ESBWR.
- The ICS provides a large inventory of water for passive cooling. ICS eliminates the need for a suppression pool, which allows for a smaller containment design.

During GDA Step 1, one of the primary tasks undertaken was a C&S review, which is documented in NEDC-34139P, "BWRX-300 UK Codes and Standards Assessment" (Reference 0-79). This assessment compares the US/Canadian to European/UK C&S equivalents across a GEH defined suite of safety and control areas. Whilst detailed code comparison is outside of the scope of the GDA and will be performed at a later licensing stage, the review completed to date gives confidence that the main C&S used in the development of the BWRX-300 are accepted as RGP in the UK.

In terms of RGP review of analysis, the approach to safety, including fundamental objectives, applying defence in depth principles, categorisation of safety functions, and classification of safety features to deliver those functions is derived from IAEA guidance and internationally recognised good practice.

A comparison of application of safety category and SSC classification for BWRX-300 and UK expectations has been undertaken in NEDC-34161P, "BWRX-300 UK Generic Design Assessment (GDA) Comparison of BWRX-300 Approach to Categorization & Classification with UK expectations" (Reference 0-80). Whilst future enhancements in the BWRX-300 approach to categorisation of safety functions and classification of SSCs have been identified, these are unlikely to lead to any deficiencies in the acceptability of the design in the UK. However, FAP items have been raised to address these in Reference 0-80.

The ALARP principle is also applicable to Conventional Health and Safety (CHS) and Conventional Fire (CF) risk. UK regulatory expectations on CHS/CFS constitute UK RGP. Initial reviews of these CHS/CFS regulatory expectations have been performed during GDA Step 1 and where potential gaps to such expectations have been identified forward actions have been raised to manage these. The BWRX-300 design is based upon decades of BWR operating experience, which is expected to support CHS/CFS risk reduction.

11.1.4 Use of Probabilistic Safety Assessment

An overview of the PSA is provided in Section 6.4.10 of this document. The PSA feeds into the integrated design engineering process depicted in Figure 0-26. This supports GEH's ALARP strategy by using risk analysis to drive safety improvements to the design.

Risk insights from the PSA have been used as part of optioneering for the BWRX-300 design. At the time of writing, insight from the PSA has:

- Shown a need for an alternative RPV depressurization mechanism in addition to the ICS. The Ultimate Pressure Regulation (UPR) is currently under design development
- Shown a need for a filtered containment vent system. PSA also supported the sizing of such a system.
- Supported the decision to provide a boration mechanism as a diverse means of reactivity control
- Precluded the need for a new RPV nozzle to accommodate the boration mechanism
- Influenced the sizing/operation of CRD injection to provide reactor inventory makeup
- Supported development of seismic capacity requirements for some equipment
- Shown the importance of spatial separation in some areas during certain fire scenarios
- Influenced the development of shutdown nuclear safety strategies
- Suggested that the feasibility and effectiveness of a seismic anticipatory scram function should be investigated
- Showed the risk reduction benefit of developing diverse makeup functions to the pools

The PSA will continue to provide risk insights to support design optioneering and risk reduction in future licensing phases.

11.1.5 Next Steps

The good practices established by GEH, coupled with the evidence of application presented in the design evolution of the BWRX-300 demonstrate a clear commitment to the ALARP principle. Phase 1 of the ALARP review will complete during GDA. Once complete, an assessment will be made of the significance of any shortfalls, including their effect on the demonstration that risk is reduced to ALARP. The safety analysis demonstrates that the acceptance criteria can be met, while recognising that work is needed to present this information in a way that allows assessment against the ONR's SAP numerical targets.

This submission does not intend to demonstrate that all claims are substantiated, but that a path to substantiation is achievable. By demonstrating that a suitable ALARP process has been established and that the organisation is capable of implementing the process by providing examples (see PSR Chapter 27), GEH is confident that there is a path to reaching an 'ALARP position'. The level of detail in PSR Chapter 27 will increase as licensing progresses beyond the two-step GDA and the design and supporting analyses mature.

11.2 Best Available Techniques

In England and Wales, there is a duty through the "Environmental Permitting Regulations 2016" (EPR16) (Reference 0-81) to make sure radiological protection is optimised in activities that generate radioactive waste. The requirement to apply BAT to minimise the generation, disposal and effects of radioactive wastes is set out in Part 4 of EPR16.

The BWRX-300 Level 1 Environment claim (see Section 9.1) is to ensure adequate protection of people and the environment from harm at all lifecycle stages of the power station. This claim, and the Level 2 claims, are supported by the use of BATs. The demonstration that the design and operation of the BWRX-300 have been optimised through the application of BAT is presented in PER Chapter 6 (Reference 0-63). The claims and arguments relating to BAT have been further developed into a set of detailed arguments, with two of these arguments being supported by evidence as worked examples. As for ALARP, GEH plans to fully demonstrate BAT during later licensing phases, and aims in GDA to provide confidence that such a position can be reached based on the processes that have been adopted and a number of examples.

A key feature of the adopted methodology is the integration of BAT into the engineering design process alongside safety and security principles to achieve integrated design optimisation for the BWRX-300. The methodology takes into account applicable regulatory requirements and associated guidance, as well as RGP. BAT is most relevant for those systems and processes that give rise to radioactive wastes or gaseous or liquid radioactive discharges to the environment.

GEH has built upon the existing BAT methodology used for the UK ABWR set out in GA91-9901-0021-00001 (XE-GD-0096), "UK ABWR Generic Design Assessment: Approach to Optimisation" (Reference 0-82). This breaks the process down into the key BAT-related permit conditions. Following this process, GEH will demonstrate that it has done everything reasonably practicable to:

- Prevent and minimise (in terms of radioactivity) the creation of radioactive waste.
- Minimise (in terms of radioactivity) discharges of gaseous and aqueous radioactive wastes.
- Minimise the effect of those discharges on people, and adequately protect other species.
- Minimise (in terms of mass and volume) solid and non-aqueous liquid radioactive wastes and spent fuel.
- Select the optimal disposal routes (taking account of the waste hierarchy and the proximity principle) for those wastes; which also includes the suitability of disposal for those wastes where there is currently no available disposal route.

The methodology proposes a systematic and evidence-based approach that aims to demonstrate that the design, manufacture, construction, commissioning, operation, and decommissioning of the BWRX-300 will be optimised to protect members of the public and to minimise the effect on the environment from exposure to ionising radiation.

The basis of the overall BAT demonstration is meeting the relevant subclaims below based on the arguments supporting the claims presented in Appendix A and PER Chapter 6.

The claims and arguments for the BWRX-300 Demonstration of BAT have been developed using information held by GEH. Evidence has been provided as a worked example for Arguments 1.1.1 and 1.1.2 to demonstrate how information is used to develop the claims and arguments.

- Prevention or, where this is not practicable, minimisation of the creation of radioactive waste and spent fuel (radioactivity and quantity)
  - Argument 1.1.1 - Design and manufacture of fuel gives low rate of fuel failure
  - Argument 1.1.2 - Effective management of fuel maintains low failure rate

Further evidence will be provided for all claims and arguments in future licensing phases. Gaps and uncertainties identified during the development of the arguments have been recorded as Forward Actions to ensure that they are managed and closed out at the most appropriate time in the project lifecycle.

Collectively the CAE model supports the demonstration that BAT has been applied to the BWRX-300 design, allowing examination and challenge and where applicable identifying key gaps or uncertainties.

GEH believes that the arguments set out and the collation of further evidence, combined with the completing of forward actions will demonstrate that the BWRX-300 has been optimised in accordance with those elements of the environmental regulators' guidance that require the application of BAT.

11.3 Design Optimisation

The requirements to apply BAT and to reduce risk to ALARP often complement each other; applying BAT helps to reduce risk, or reducing risk is achievable by applying BAT. The same applies when considering nuclear security and safeguards. For example, Section 8.2 presents some design modifications made to address nuclear security that also improved nuclear safety. The location of the ICS pools was changed to reduce vulnerability to external breaching but this also improved resilience to external hazards. Similarly, the delivery receipt location was moved to minimise the ease of introducing explosive devices close to the PA, and this also reduced the effect of any internal hazards from such deliveries.

However, sometimes conflicts do occur. For example, in the management of radioactive wastes, applying BAT to reduce radioactive discharges—and therefore reduce radiation exposure to the public and environment—might increase exposure for workers. Similar conflicts can arise between achieving safety or protecting the environment and ensuring nuclear security and safeguards.

The potential for such conflicts and the need to manage them is addressed within the IAEA Safety Standards Series and Nuclear Security Series. For example, IAEA SSR-2/1 Requirement 8: Interfaces of safety with security and safeguards, says that “safety measures, nuclear security measures and arrangements for the State system of accounting for, and control of, nuclear material for a nuclear power plant shall be designed and implemented in an integrated manner so that they do not compromise one another.”

GEH’s approach to GDA is in line with IAEA’s requirements. Rather than presenting a “holistic” case that attempts to consider safety, security, safeguards, and environment together at all times, GEH’s approach is to identify the interfaces between the topics. Where there are interfaces and conflicts arise, the process of design optimisation weighs the benefits against time, cost, complexity, operability, and other topics. This process may involve experts on various subject matters, as well as representatives from management (Reference 0-46).”

12 FORWARD ACTION PLAN

The BWRX-300 GDA submissions represent a snapshot in time for the project. Necessarily, the level of detail presented in the GDA submissions is limited and more detailed analyses and design substantiation will be required in later licensing stages. Whilst the GDA submissions are based on international standards and methodologies, in some instances work will be required in the future to adapt these to meet national expectations. These additional activities are captured as Forward Action items throughout the submissions and collated in NEDC-34274P "BWRX-300 UK GDA Forward Action Plan" (Reference 0-83).

The FAP captures the actions required for the program to progress from GDA to a site-specific phase and captures any commitments made in response to Regulatory Queries and Regulatory Observations. The FAP is not required to capture the 'normal business' of Safety, Security, Safeguards, and Environmental case development as the design progresses from concept to design for construction and commissioning.

FAP items can arise from several sources:

- Assumptions and commitments in the GDA submissions that require future verification/ implementation, for example, by the future constructor and/or plant operator
- A gap in the underpinning of the GDA submissions currently under development
- A potential gap in a future phase of submissions if additional work is not performed
- A gap identified by the regulators during GDA Step 2

Potential FAP items identified during development of the GDA submissions are recorded in each chapter and collated in a central register. The register is reviewed regularly by the GEH UK Licensing Manager and the FAP items are sentenced into one of the three phases.

- Within Step 2: These are actions where it is considered that the GDA submissions are not fully underpinned at this time and additional work will be performed during the regulatory assessment period to provide additional confidence that a suitable justification has been made during the GDA.
- During PCSR/ PCER development: These are actions where it is considered that the GDA submissions are adequately underpinned, but either there are assumptions that require verification in the next licensing phase or there are underpinning methods or analyses that would require adaptation/ development to support a UK PCSR/PCER.
- Before Site Licence (and associated permits) Application: These are actions that will only be discharged once a future licensee is engaged in the programme.

Once FAP items are sentenced and reconciled, the authors of the GDA submission documents are informed of any amendments and the commitments and FAP items in the GDA submissions are then updated to align with the central register.

The FAP interfaces into the GEH commitment management, design management and corrective action programmes:

- Where the FAP identifies that additional work should be performed then this will be planned and delivered in line with GEH standard work planning processes
- Where the FAP identifies a potential change to the standard plant design then this will be managed through GEH design management processes
- Where the FAP identifies a significant shortfall that requires corrective action then this will be managed through the GEH Corrective Action Program

13 CONCLUSIONS

The BWRX-300 design maximises experience and learning from previous generations of BWRs while incorporating limited but impactful innovations that improve both safety and performance. The BWRX-300 has many features common with the UK ABWR, which completed a four-step GDA and was granted Design Acceptance Confirmation and a Statement of Design Acceptability from the ONR, EA, and NRW in 2017. The BWRX-300 offers further safety improvements compared to the UK ABWR through its adoption of natural circulation (passive safety) and design innovations such as integral reactor isolation valves.

The purpose of this document has been to summarise GEH's main submission as part of this two-step GDA process. This main submission is made up of four volumes that address the topics of safety, security, safeguards, and the environment across the lifecycle of the BWRX-300. GEH believes the submission and its supporting references is broad enough in scope and detail to enable a meaningful assessment of the BWRX-300 by the ONR, EA and NRW as part of Step 2 of the GDA process.

The Fundamental Objective of the submission is to show 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 England and Wales. This objective sits at the top of a hierarchy of CAE (summarised in Appendix A), that succinctly expresses how nuclear safety, security, safeguards, and environmental protection will be achieved for the BWRX-300.

GEH believes that this submission and its supporting references provide confidence that there is a viable path towards substantiation for all the claims in the CAE hierarchy. This includes those claims related to reducing risk to ALARP and to the application of BAT. GEH's current and planned activities are consistent with achieving this substantiation. Some activities needed to support substantiation have been captured in a FAP (see Section 12), but others will be addressed as a matter of course as the design matures (i.e., as part of 'normal business'). GEH has not identified any insurmountable obstacles to substantiating the claims.

Given this path to substantiation, GEH believes the submission shows the feasibility of constructing, commissioning, operating, and decommissioning the BWRX-300 design at a site in England or Wales.

Table 0-1: Evolution of the GE Boiling Water Reactor

Product Line	First Commercial Operation Date	Representative Plant/Characteristics
BWR/1	1960	Dresden 1 - Initial commercial-size BWR
BWR/2	1969	Oyster Creek – Turnkey project purchased solely on economics. Large direct cycle External recirculation pumps
BWR/3	1971	Dresden 2 – First jet pump application Improved Emergency Core Cooling System (ECCS) Core spray and flood capability
BWR/4	1972	Vermont Yankee – Increased power density (20%)
BWR/5	1978	Tokai 2 – Improved ECCS Valve flow control for recirculation system
BWR/6	1981	Kuosheng 1 – Compact control room Solid-state nuclear system protection system Advanced containment design
ABWR	1996	Kashiwazaki-Kariwa 6 – Reactor internal pumps Fine Motion Control Rod Drives (FMCRDs) Advanced control room, digital and fibreoptic technology Improved ECCS: high/low-pressure flooders
SBWR	-	Natural circulation Passive ECCS Passive containment cooling
ESBWR	-	Natural circulation Passive ECCS Passive containment cooling
BWRX-300	-	Loss-of-Coolant Accident (LOCA) mitigation through integral isolation valves Natural circulation Passive heat removal systems Reactor Building (RB) built from Diaphragm Plate Steel - Plate Composite modules.

Table 0-2: GE BWRs Worldwide

Country	Number of Units	Total Installed Capacity (MW)
Germany	2	252
India	2	300
Italy	2	1010
Japan	8	6307
Mexico	2	1552
Netherlands	1	55
Spain	1	1064
Switzerland	2	1593
Taiwan	6	5942
USA	41	37113

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Table 0-3: Non-GE BWRs Worldwide

Country	Number of Units	Total Installed Capacity (MW)
Canada	1	250
Finland	2	1780
Germany	9	7305
Japan	28	26675
Spain	1	446
Sweden	10	8107
USA	3	129

Table 0-4: Anticipated Operational Occurrence Deterministic Safety Analysis Acceptance Criteria

Fission Product Barrier or Fundamental Safety Function	Qualitative Acceptance Criteria	Quantitative Acceptance Criteria
General	An AOO will not escalate to a more serious accident condition unless other faults occur independently.	Not applicable.
	There is no consequential failure or loss of function of any fission product barrier.	Not applicable.
Fuel Rod	Fuel temperature results will not cause a loss of fuel rod mechanical integrity.	The calculated maximum fuel center temperature T_{center} remains below the fuel melting point T_{melt}.
	Fuel pellet-cladding results do not lead to loss of fuel rod mechanical integrity	The cladding strain acceptance criteria defined in Section 5.0 of NEDC-33840P, “The PRIME Model for Transient Analysis of Fuel Road Thermal-Mechanical Performance” (Reference 14-84).
	Fuel cladding temperature results will not cause a fuel rod failure.	The calculated core Minimum Critical Power Ratio (MCPR) ensures that 99.9% of the fuel rods in the core are not susceptible to boiling transition during AOO events. With the reactor steam dome pressure less than 4.72 MPaG (685 psig), the calculated reactor thermal power is less than 25% of rated thermal power.
Reactor Coolant Pressure Boundary	Design conditions of the reactor coolant pressure boundary are not exceeded during the most severe pressurisation transient.	The calculated peak pressure associated with the reactor coolant pressure boundary does not exceed 110% of the design pressure or 11.38 MPaG (1650 psig).
	The reactor coolant pressure boundary maintains sufficient reactor coolant inventory for core cooling.	The calculated reactor water level is maintained at or above top of active fuel.

Fission Product Barrier or Fundamental Safety Function	Qualitative Acceptance Criteria	Quantitative Acceptance Criteria
Primary Containment	Containment integrity is maintained. If an AOO results in an energy release to the containment, or loss of containment heat removal, then containment stresses (i.e., pressure and temperature) are limited such that there is no loss of a containment barrier safety function, and thus, the containment remains within its design limit values.	No AOOs result in a significant energy release to containment, or prolonged loss of normal containment cooling. The normal operation limits and conditions are applied to containment, and no AOO containment quantitative criteria is needed.
Long-Term Heat Removal	SSCs important for preserving the integrity of the reactor core and the containment can remove residual heat for an extended period both during and after all applicable PIEs considered in all Operational States, including AOOs.	Following AOO events that do not result in shutdown, a controlled condition is achieved. Following AOO events that require shutdown, the core remains shutdown independent of operator action or offsite support for at least 72 hours. For AOO events that rely on Defense Line 3 (DL3) mitigation for long-term cooling the DL3 functions can provide cooling for at least 72 hours without operator action or offsite support.

Table 0-5: Design Basis Accident Acceptance Criteria

Fission Product Barrier or Fundamental Safety Function	Qualitative Acceptance Criteria	Quantitative Acceptance Criteria
General	Except for fuel cladding, there is no loss of function of any fission product barrier.	Not applicable
Fuel Rod	The number of fuel rod failures is conservatively estimated for DBAs.	The calculated number of failed rods does not result in exceeding the applicable radiological dose acceptance criteria.
	Mechanical fracturing of a fuel assembly under DBA loading conditions does not result in losing the ability to cool the fuel assembly.	The mechanical integrity of the fuel is established from the mechanical and thermal fuel analysis described in PSR Chapter 4: Reactor Section 4.2.2 (Reference 14-33).
Fuel Cooling	The calculated fuel cladding temperature is maintained at an acceptably low value and decay heat is removed for the extended period required by the long-lived radioactivity remaining in the core.	The calculated PCT remains less than 1204°C (2200°F). The calculated total oxidation of the cladding nowhere exceeds 0.17 times the total cladding thickness before oxidation for DBAs where exceeding the oxidation thickness challenges the capability to cool the core.
Reactor Coolant Pressure Boundary	Design conditions of the reactor coolant pressure boundary are not exceeded during the most severe pressurisation transient because of a DBA.	The calculated peak pressure associated with the reactor coolant pressure boundary does not exceed 120% of the design pressure or 12.41 MPaG (1800 psig).
	The reactor coolant pressure boundary maintains sufficient reactor coolant inventory for core cooling	Conformance is demonstrated by meeting the fuel cooling and long-term heat removal criteria.
Primary Containment	Containment pressures and temperatures are maintained below the design values.	The calculated containment pressure does not exceed the design pressure 0.414 MPaG (60 psig). The calculated containment shell temperature does not exceed the design temperature 165.6°C (330°F).

Fission Product Barrier or Fundamental Safety Function	Qualitative Acceptance Criteria	Quantitative Acceptance Criteria
	The local combustible gas concentrations in the containment are within the range where deflagration or detonation cannot occur.	Containment atmosphere remains sufficiently mixed such that deflagration or detonation thresholds are not exceeded.
	Containment capability will be retained to reduce the containment pressure and temperature following a DBA to minimise the release of fission products to the environment and to preserve containment integrity and leak tightness.	The calculated containment pressure reduces to less than 50% of the calculated peak pressure for the most limiting LOCA within 24 hours.
Reactivity Control	Reactivity control required to bring the reactor to cold shutdown is maintained.	Shutdown margin is established to assure that the reactor can be brought subcritical with the highest-worth control rod pair withdrawn when the core is in its most reactive condition. The subcriticality value is 0.38% $\Delta k/k$ with the highest-worth control rod pair analytically determined.
Long-Term Heat Removal	SSCs important for preserving the integrity of the reactor core and the containment are capable of removing residual heat for an extended period both during and after all applicable PIEs considered in all operational states, and DBAs.	Long-term cooling is maintained for a minimum of 72 hours independent of operator action and offsite support, and for 30 days with credit for operator actions and on-site resources. For DBA events that result in shutdown, the plant can achieve and maintain safe-shutdown conditions with the average reactor coolant temperature below 215.6°C (420°F).

Table 0-6: Probabilistic Safety Goals

Qualitative Acceptance Criteria	Quantitative Acceptance Criteria
Core damage frequency	The sum of frequencies of all event sequences that can lead to significant core degradation shall be less than 1E-6 per reactor year
Large release frequency	The calculated sum of frequencies of all event sequences that can lead to any large release shall be less than 1E-7 per reactor year.

The BWRX-300: how is safety achieved?

The BWRX-300 design maximises experience and learning from previous generations of BWRs while incorporating limited, but impactful, innovations that improve both safety and performance.

10th Generation BWR

The BWRX-300 is a tenth generation BWR leveraging the benefits and lessons learned from previous generations. This heritage includes the Advanced Boiling Water Reactor (ABWR), which successively completed a four-step GDA in the UK in 2017.

Natural circulation

The BWRX-300 is the simplest design to date, using a tall vessel design to achieve natural circulation whilst maintaining a standard core height. It leverages technology from the ESBWR that has already been licensed in the United States.

Standardised fuel

The standard height core allows the BWRX-300 to use Global Nuclear Fuel's proven GNF2 fuel assemblies that are manufactured and sold to over 80% of the BWR fleet; more than 26,000 GNF2 fuel assemblies have been delivered worldwide as of 2023. The GNF2 fuel assemblies have low hydraulic resistance which benefits natural circulation.

Targeted innovation

While being grounded in mature BWR technology, the BWRX-300 includes targeted innovations that enhance safety and reduce the risk of accidents. For example, the integral Reactor Pressure Vessel (RPV) isolation valves improve the plant's ability to mitigate losses of coolant. The Isolation Condenser System (ICS) passively removes heat from the reactor without the need for a bulky suppression pool or relief valves inside the containment structure. The ICS also ensures a large water inventory is available above the core at all times. This also has the effect of making the containment simpler and smaller than previous generations of BWRs.

Designed for safety

As required by the International Atomic Energy Agency (IAEA), safety is achieved primarily through 'defence-in-depth', i.e., by providing complimentary layers of safety features that allow the plant to safely respond to challenges such as equipment failures or extreme weather events. This is backed up by risk analysis that shows the risk of accidents that could impact people and the environment is extremely low. GEH's iterative design process ensures that insights from the risk analysis are fed back into the design to improve safety and to ensure that no one system or component contributes disproportionately to plant risk.

Figure 0-1: How Safety Has Been Achieved for the BWRX-300 Design.

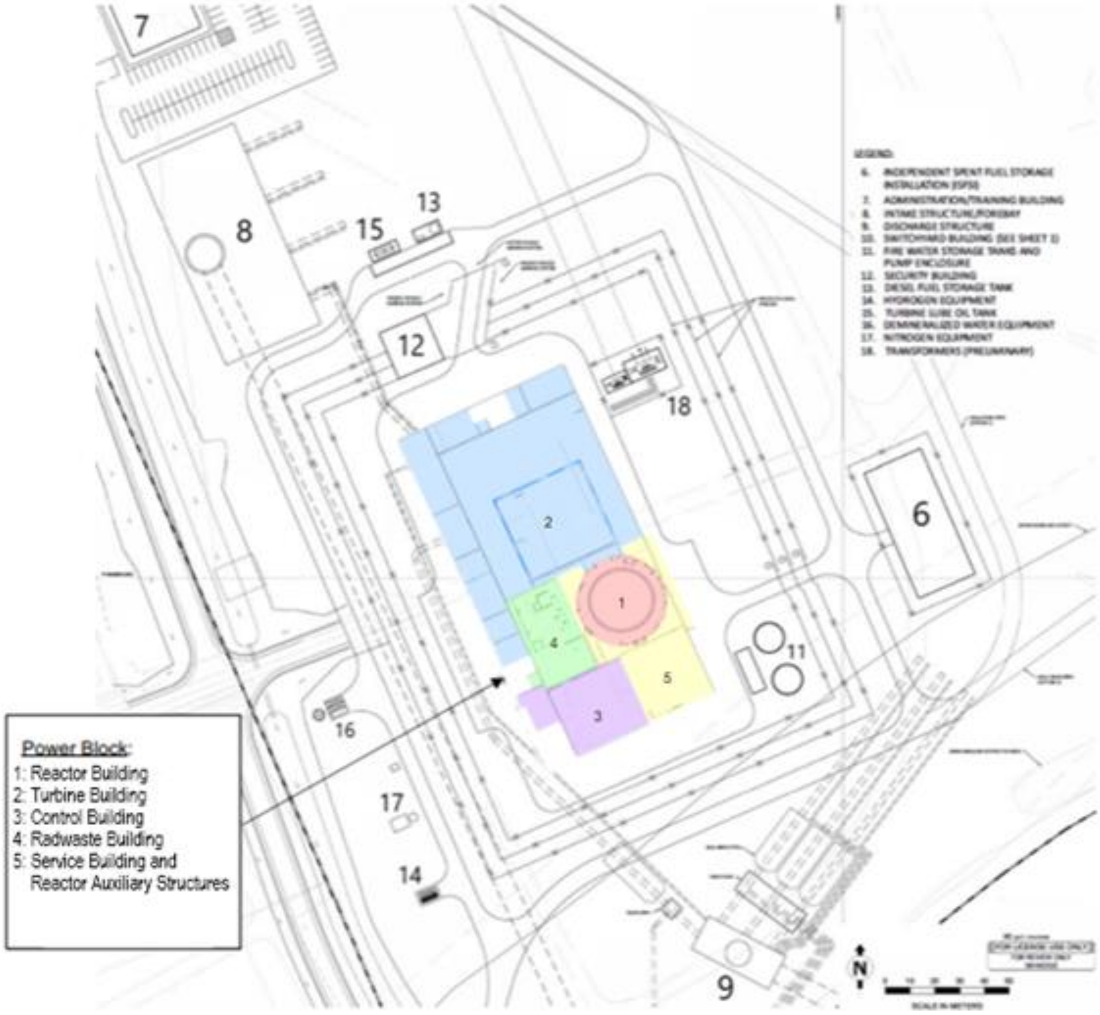

Figure 0-2: Representative BWRX-300 GDA Site Layout with Power Block Highlighted

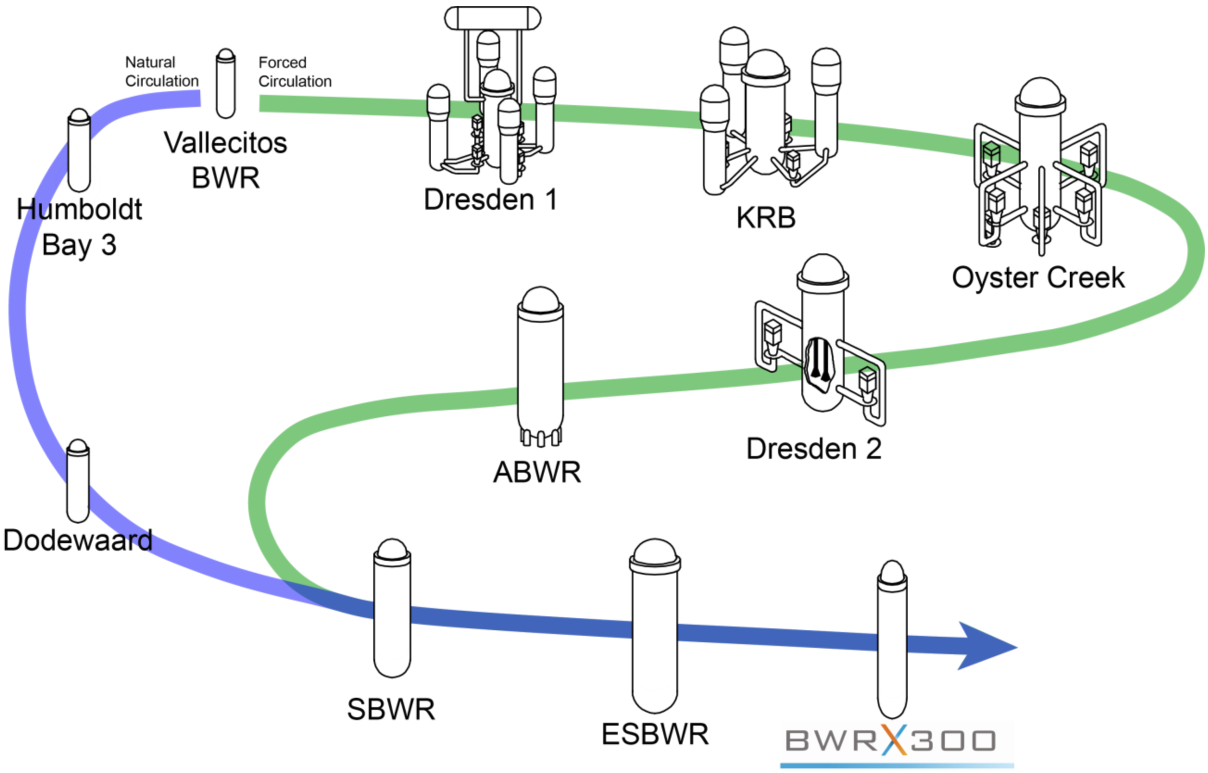

Figure 0-3: Boiling Water Reactor Design Evolution

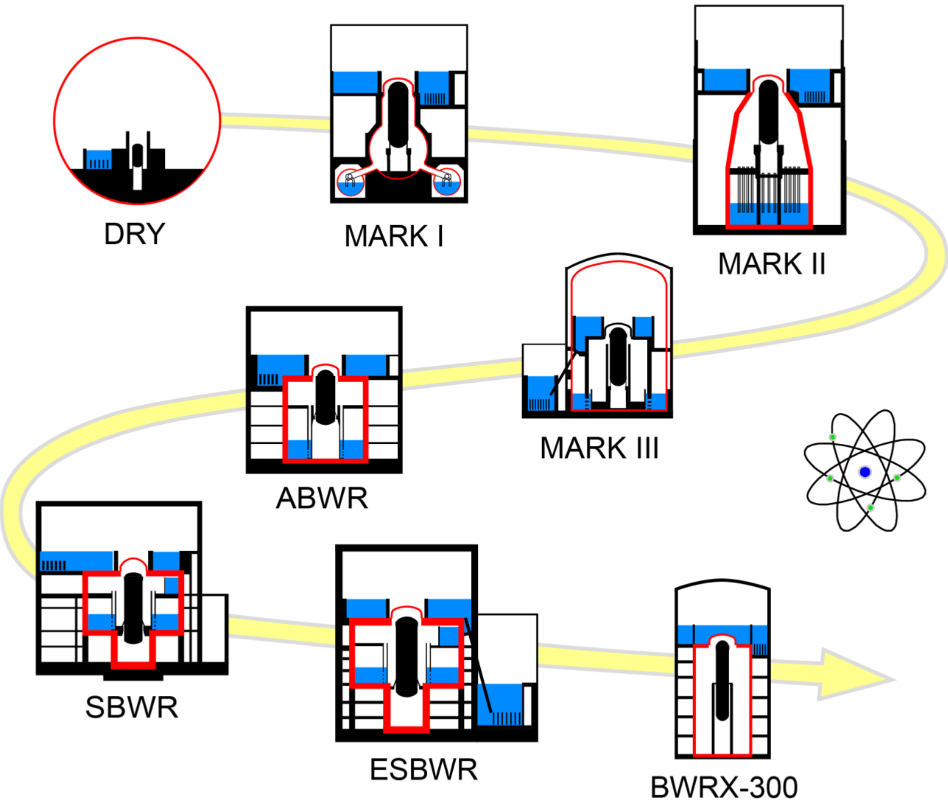

Figure 0-4: GEH Containment Designs

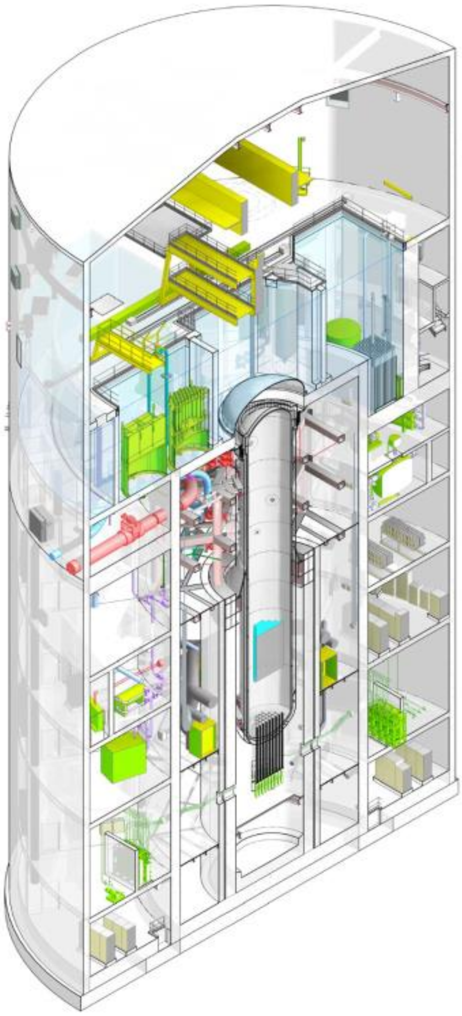

Figure 0-5: BWRX-300 Reactor Building and Containment Design, from "BWRX-300 General Description"

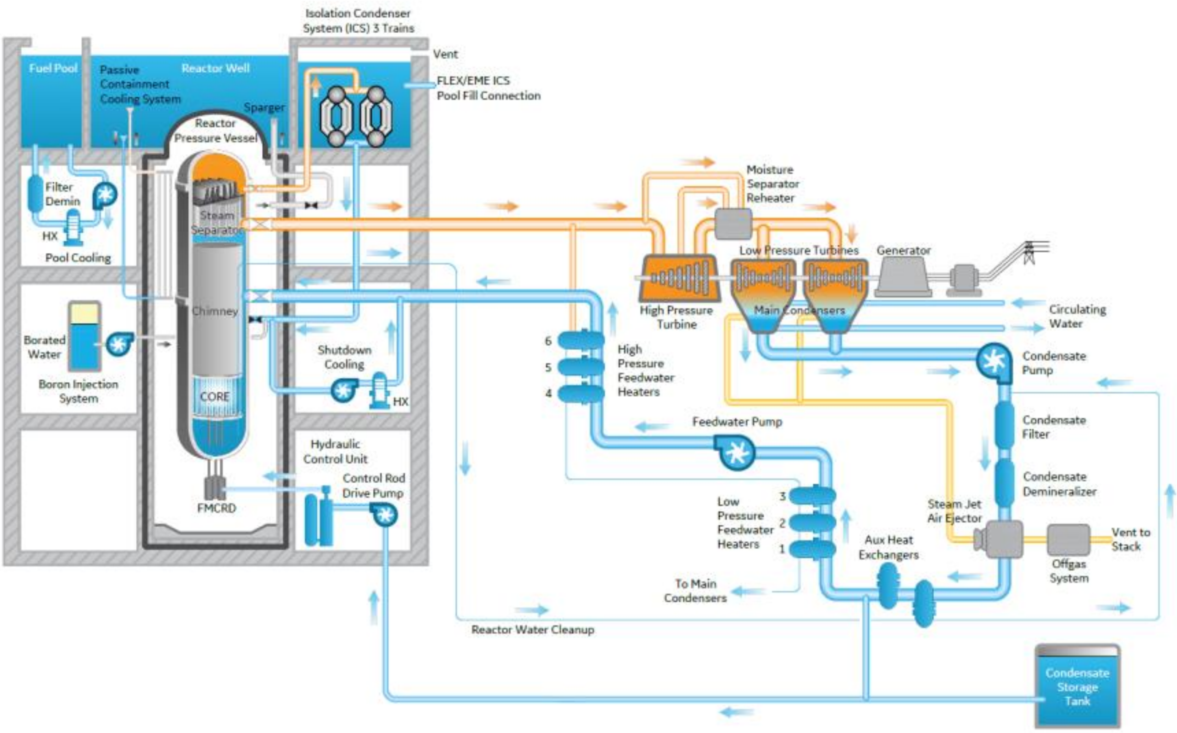

Figure 0-6: BWRX-300 Major Systems

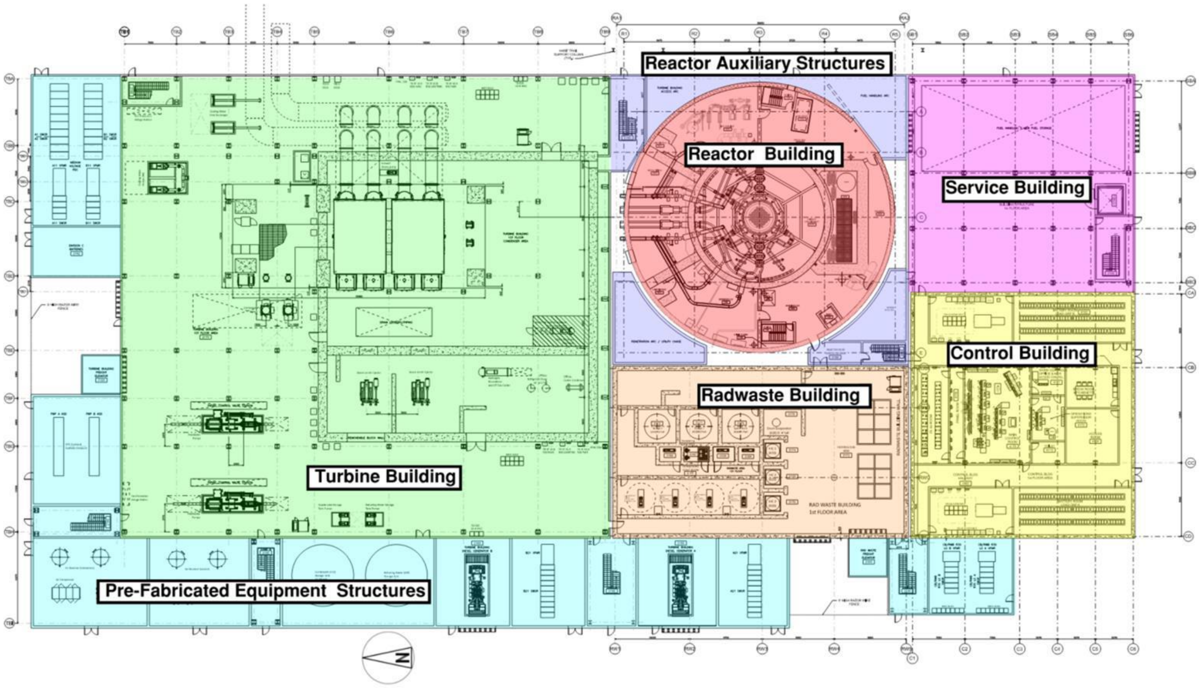

Figure 0-7: BWRX-300 Plant Power Block Building Boundaries Plan

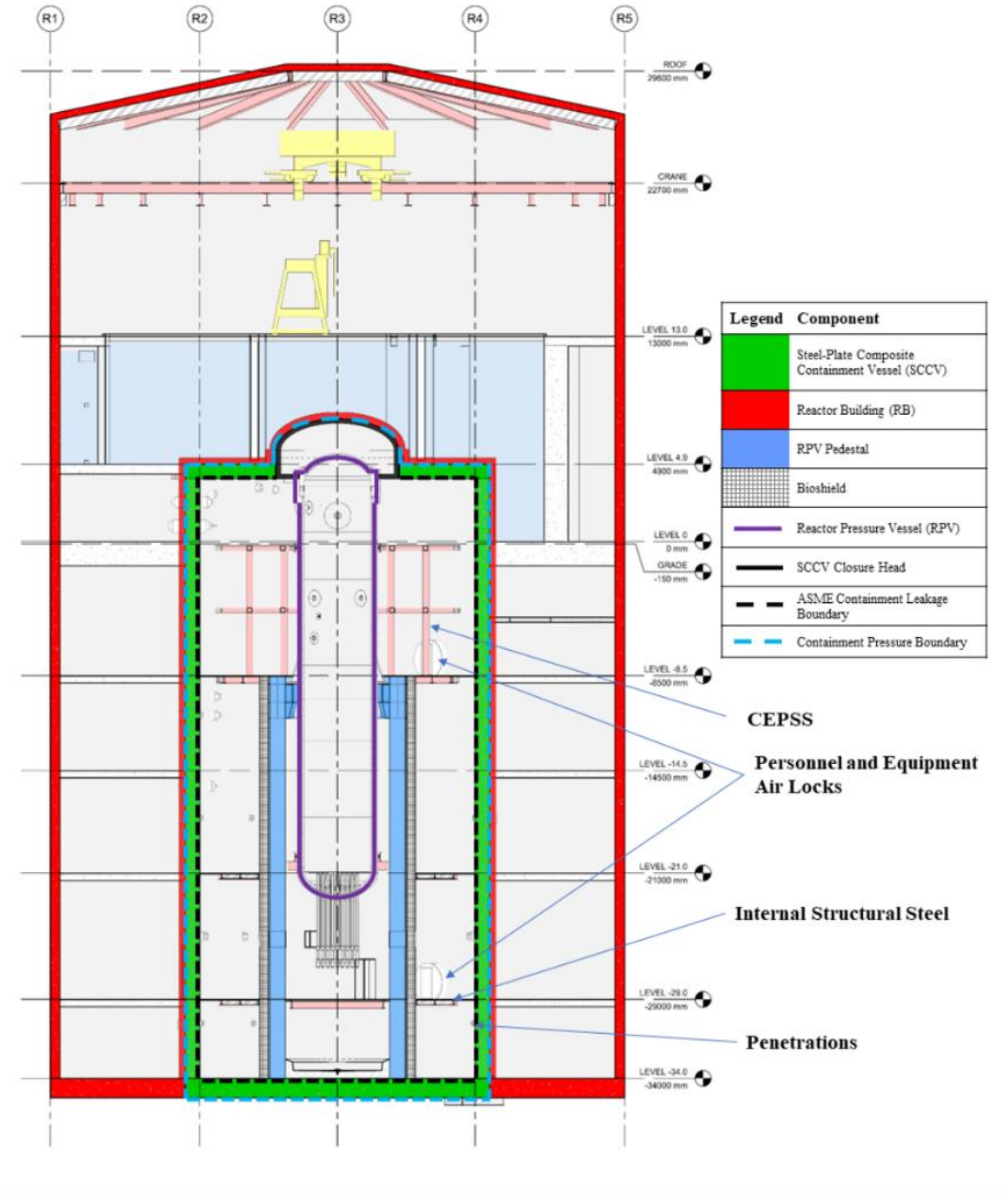

Figure 0-8: BWRX-300 Integrated Reactor Building Boundaries

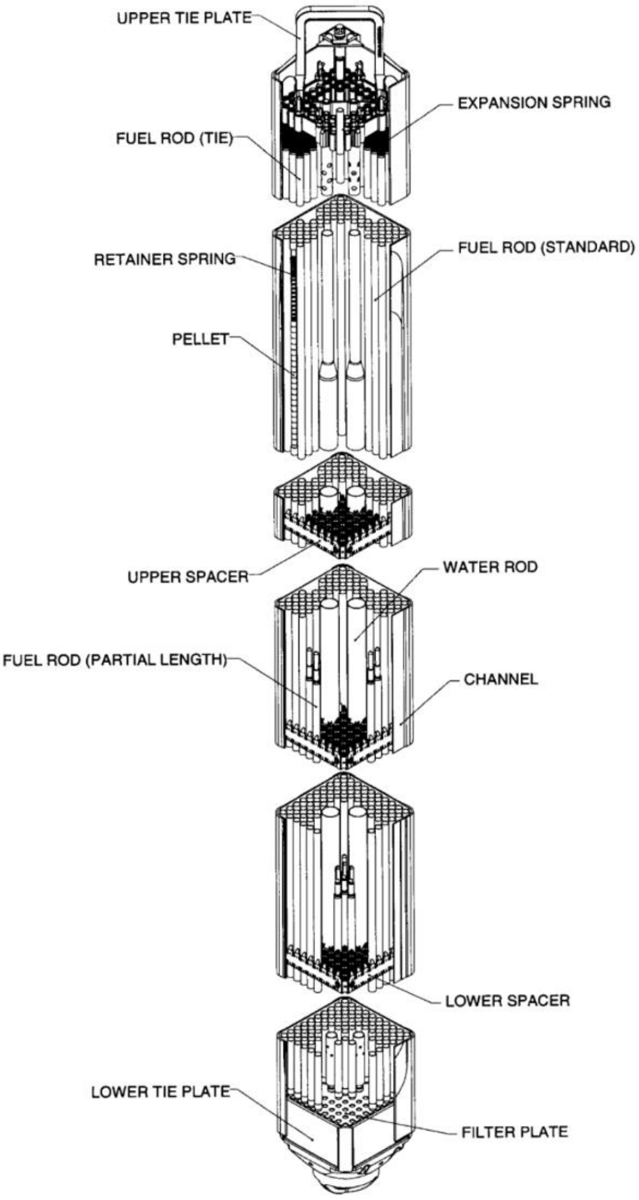

Figure 0-9: GNF2 Fuel Bundle as per "BWRX-300 Fuel Design and Qualification"

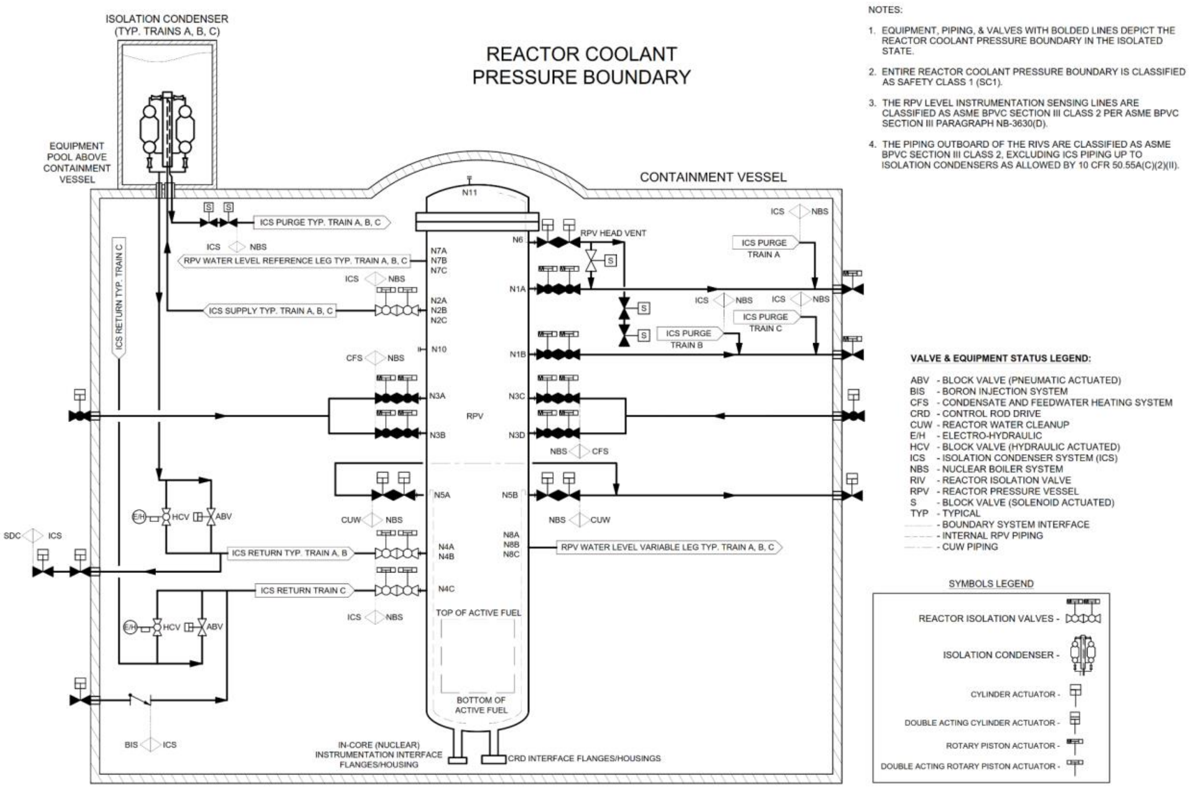

Figure 0-10: Reactor Coolant Pressure Boundary

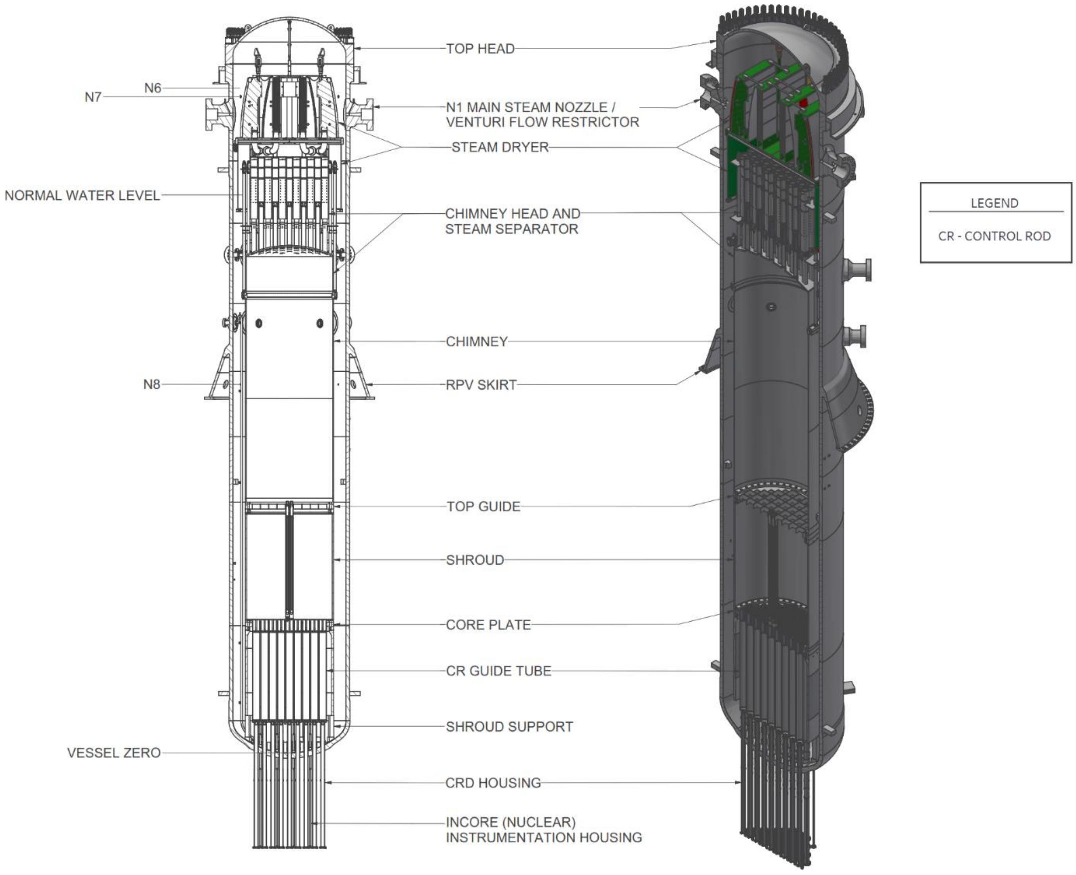

Figure 0-11: RPV Internals

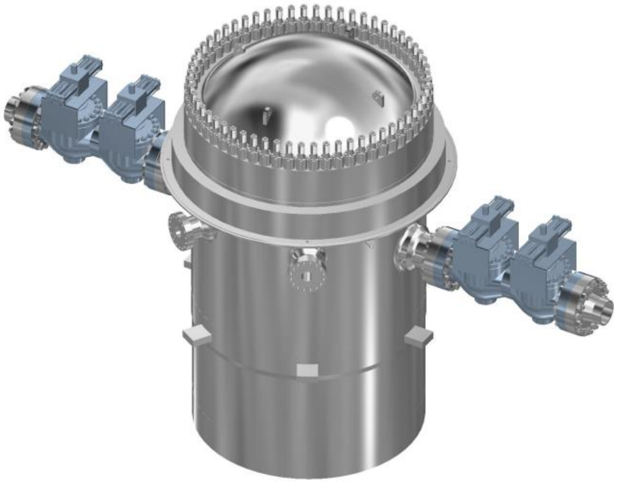

Figure 0-12: Reactor Isolation Valve Arrangement

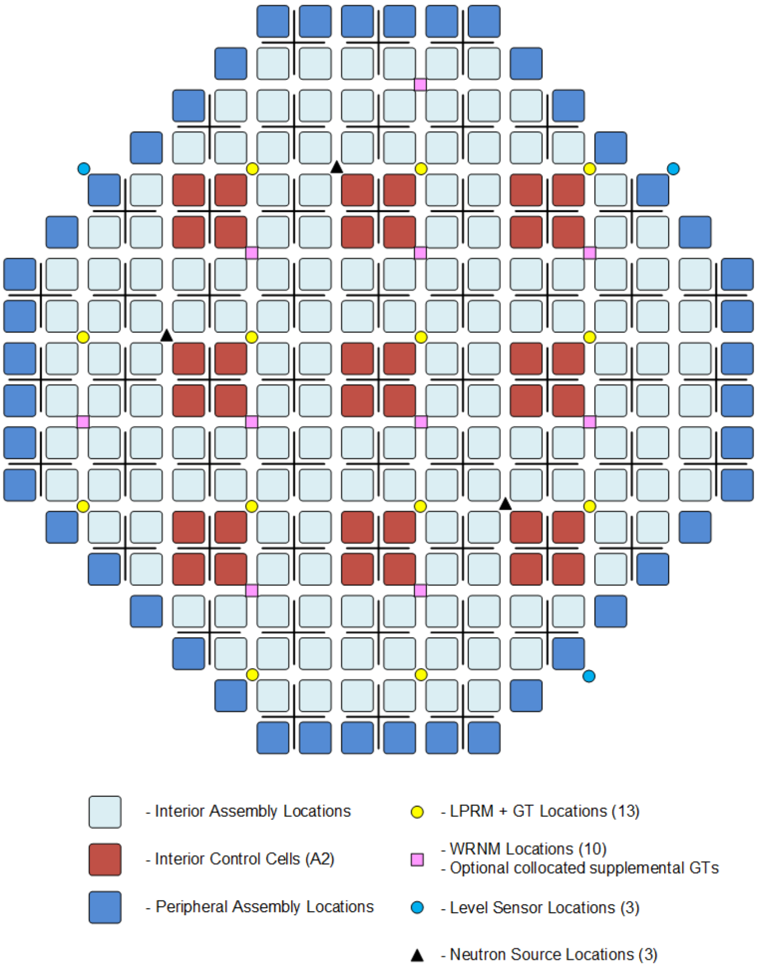

Figure 0-13: Incore Instrumentation Arrangement

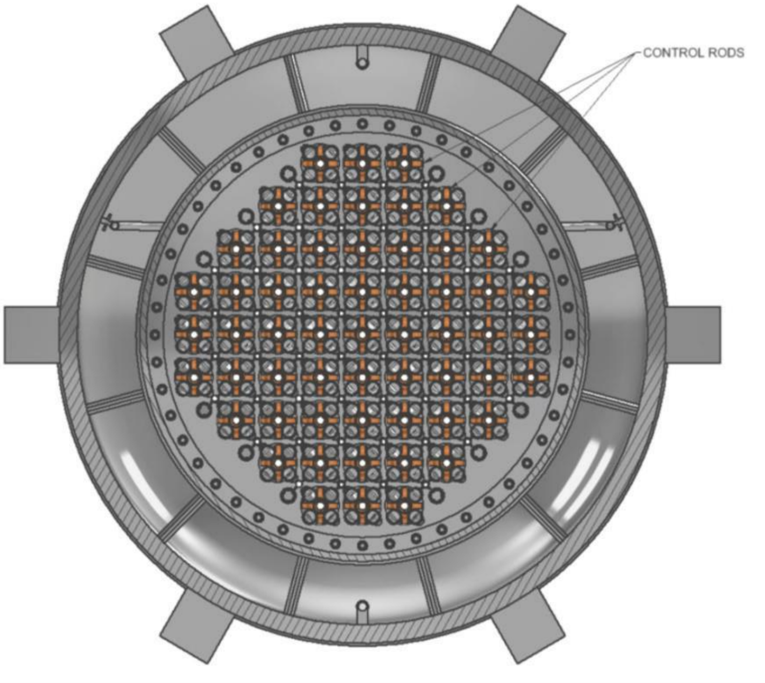

Figure 0-14: Control Rod Arrangement

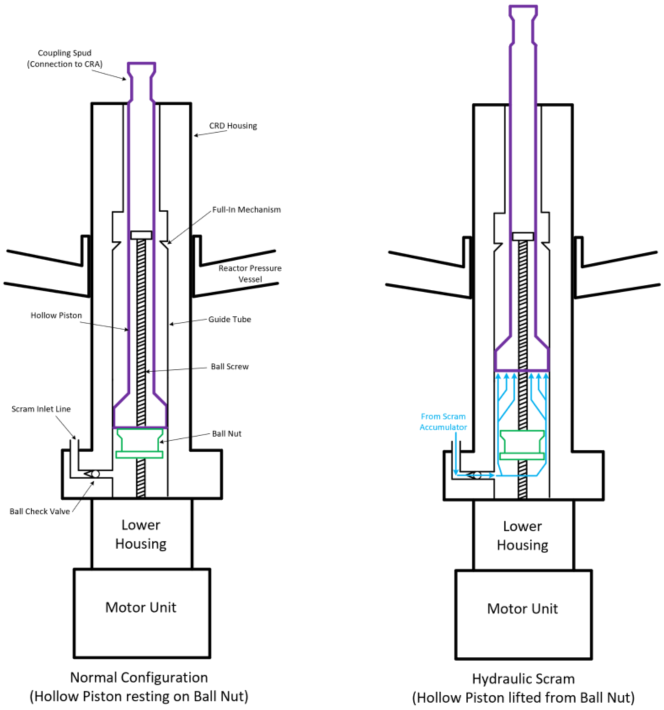

Figure 0-15: Fine Motion Control Rod Drive

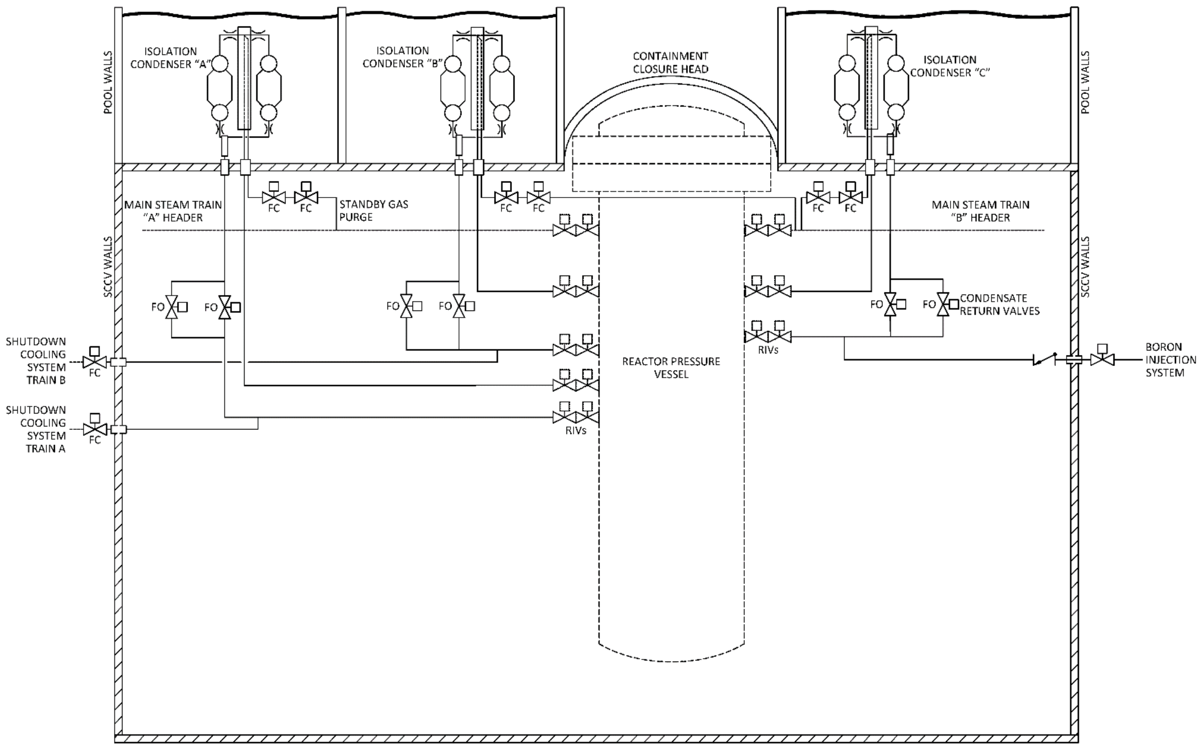

Figure 0-16: ICS Schematic

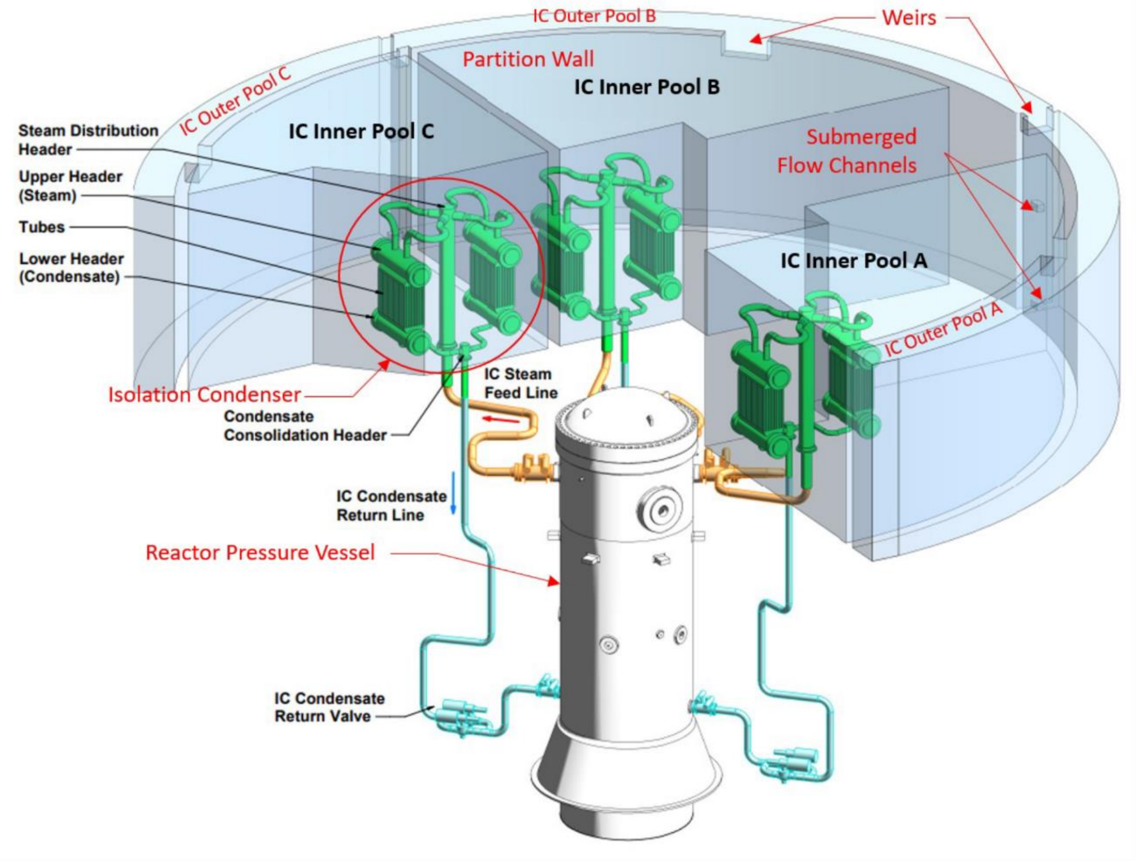

Figure 0-17: ICS and ICC Schematic from "Isolation Condenser Pools Cooling and Cleanup System"

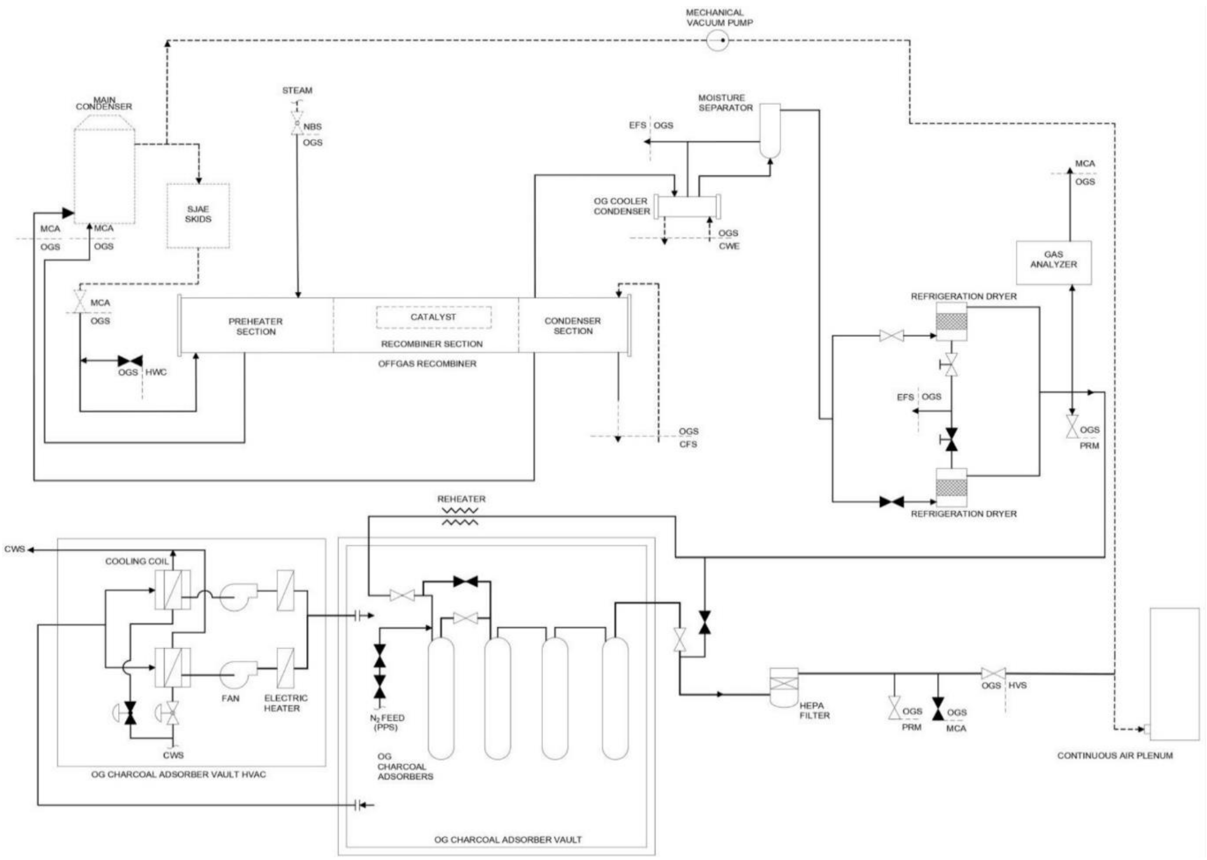

Figure 0-18: Offgas System Simplified Diagram

Level 1 Safety Claim:

- The safety risks to workers and the public during the construction, commissioning, operation and decommissioning of the BWRX-300 have been reduced as low as reasonably practicable (ALARP).

Level 2 Safety Claims:

- The functions of systems and structures have been derived and substantiated taking into account Relevant Good Practice (RGP) and Operational Experience (OPEX), and processes are in place to maintain these through-life (Engineering Analysis).
- The BWRX-300 has been developed in accordance with approved procedures, with appropriate governance and assurance arrangements by a competent and clearly defined organisation (Safety Case Area).
- A suitable and sufficient safety analysis has been undertaken which presents a comprehensive fault and hazard analysis that specifies the requirements on the safety measures and informs emergency arrangements (Safety Analysis).
- Safety risks have been reduced as low as reasonably practicable.

Figure 0-19: BWRX-300 Safety Claims

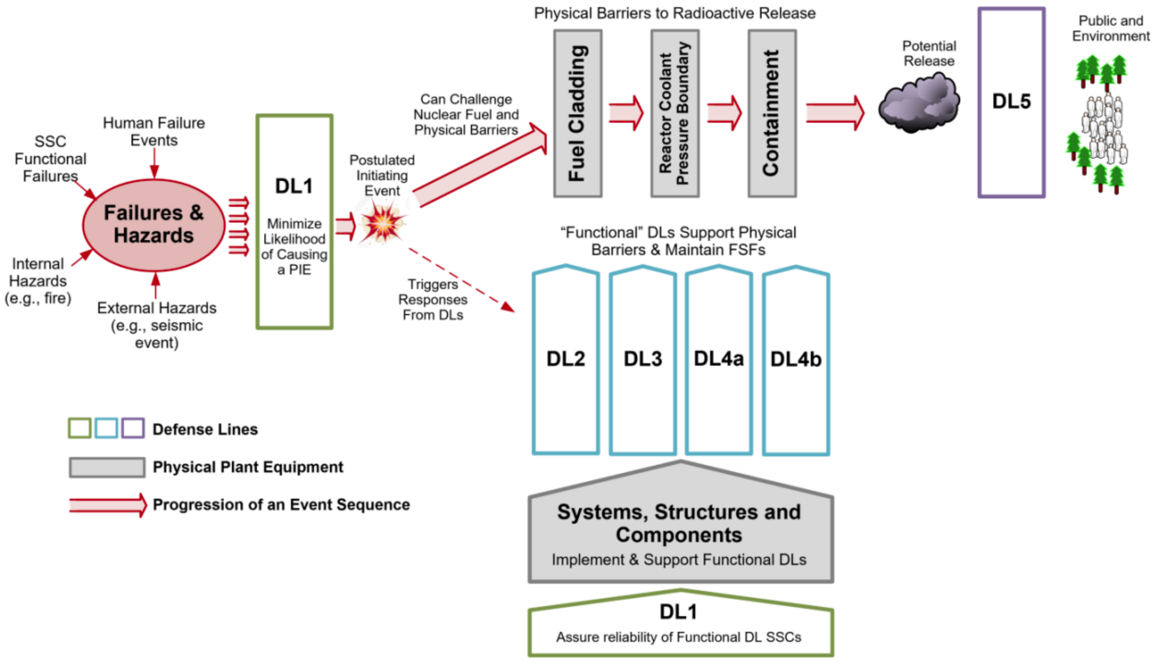

Figure 0-20: BWRX-300 Defence-in-Depth Concept

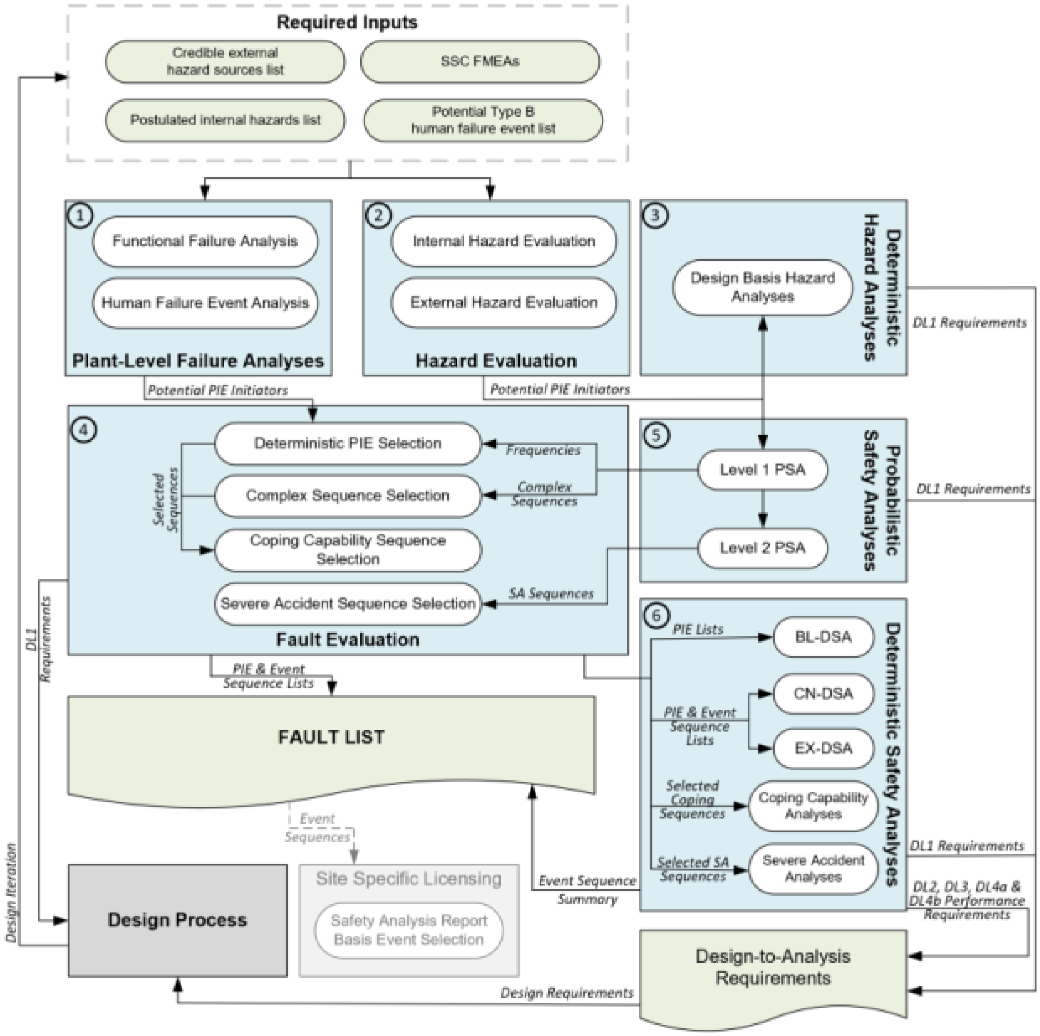

Figure 0-21: Safety Strategy Evaluation and Analysis Framework

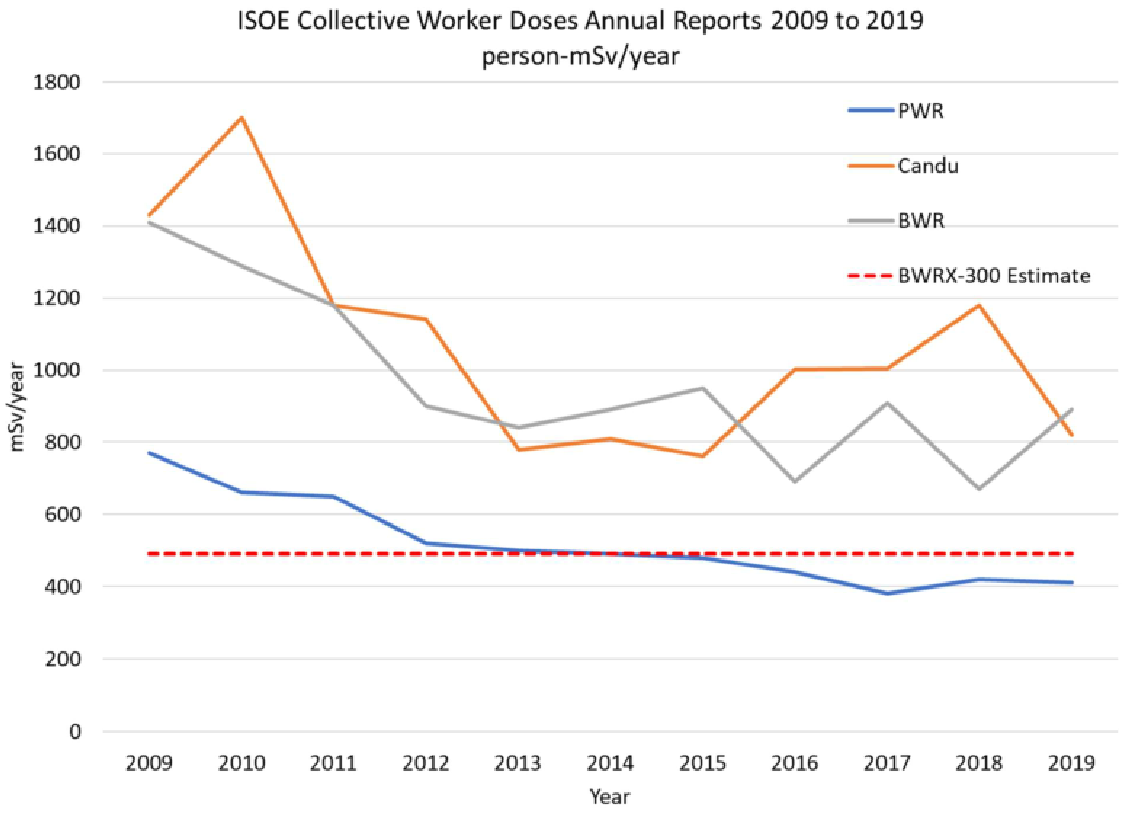

Figure 0-22: Comparison of BWRX-300 Collective Dose Estimate (491 person-mSv/year) with ISOE Industry Operating Data

Level 1 Security Claim:

The nuclear security arrangements of the BWRX-300 shall protect the public and environment from the risks arising from an unacceptable radiological consequence resulting from:

- Malicious actions of sabotage of nuclear material, other radioactive material and/or structures, systems, and components maintaining or supporting plant and nuclear safety;
- The theft of nuclear material and other radioactive material;
- The compromise of Sensitive Nuclear Information (SNI)

Figure 0-23: BWRX-300 Security Claims

Level 1 Safeguards Claim:

- Appropriate nuclear material accountancy is undertaken to minimize the potential nuclear materials to be used for non-peaceful purposes

Level 2 Safeguards Claims:

- The design process for the BWRX-300 reactor has followed IAEA's guidance on International Safeguards in the Design of Nuclear Reactors.
- The BWRX-300 reactor may be operated according to the guidance in the ONR Nuclear Material Accountancy, Control, and Safeguards Assessment Principles.
- The BWRX-300 design considers safeguards' interface with safety, security and waste management issues.

Figure 0-24: BWRX-300 Safeguards Claims

Level 1 Environment Claim:

- The design of the BWRX-300 SMR has been optimised to reduce environmental impacts to As Low As Reasonably Achievable (ALARA) throughout the whole lifecycle (construction, commissioning, operation and decommissioning).

Level 2 Environment Claims:

- Prevention or, where this is not practicable, minimisation of the creation of radioactive waste and spent fuel.
- Minimisation of the activity of gaseous radioactive waste disposed of by discharge to the environment.
- Minimisation of the activity of aqueous radioactive waste disposed of by discharge to the environment.
- Minimisation of the volume of solid radioactive waste disposed of by transfer to other premises.
- Selection of the optimal disposal routes for wastes and spent fuel.
- Minimisation of the impact of radioactive discharges on members of the public and the environment

Figure 0-25: BWRX-300 Environment Claims

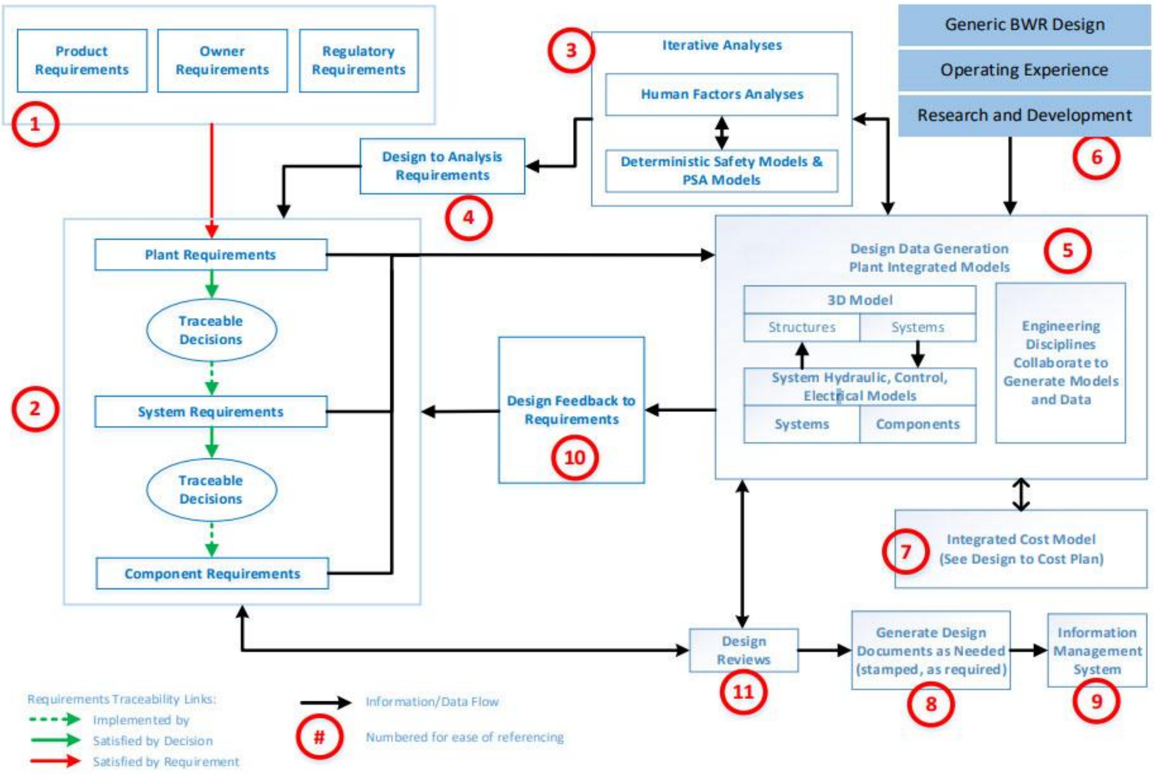

Figure 0-26: Integrated Design Engineering Process

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APPENDIX A CLAIMS, ARGUMENTS AND EVIDENCE

Fundamental Objective	Level 1 Claims	Level 2 Claims
The BWRX-300 is capable of being constructed, operated, and decommissioned in accordance with the standards of safety, security, safeguards and environmental protection required in the UK.	**Safety** The safety risks to workers and the public during the construction, commissioning, operation, and decommissioning of the BWRX-300 have been reduced as low as reasonably practicable (ALARP).	The functions of systems and structures have been derived and substantiated taking into account Relevant Good Practice (RGP) and Operational Experience (OPEX), and processes are in place to maintain these through-life. (Engineering Analysis).
		The BWRX-300 has been developed in accordance with approved procedures, with appropriate governance and assurance arrangements by a competent and clearly defined organisation (Safety Case Area).
		A suitable and sufficient safety analysis has been undertaken which presents a comprehensive fault and hazard analysis that specifies the requirements on the safety measures and informs emergency arrangements. (Safety Analysis).
		Safety risks have been reduced as low as reasonably practicable.
	Security The nuclear security arrangements of the BWRX-300 shall protect the public and environment from the risks arising from an unacceptable radiological consequence resulting from: • Malicious actions of sabotage of nuclear material, other radioactive material and/or structures, systems, and components maintaining or supporting plant and nuclear safety • The theft of nuclear material and other radioactive material or • The compromise of Sensitive Nuclear Information (SNI).	The nuclear security arrangements create protection from malicious harm through a threat informed, proportionate solution, cognizant of the detail within the DBT. Security shall be an integrated component of engineering and digital architectural design that seeks to reduce vulnerabilities through minimising inherent risk, over attempting to secure or mitigate them post-design.
		The nuclear security arrangements provide multiple barriers for protection against malevolent acts, including physical protection systems, engineered safety provisions, cyber protection systems, and measures for post-event management. The concept of defence-in-depth shall be applied to all design-related security activities to ensure they are subject to overlapping provisions, independent to the extent practicable, and that the failure of a preceding barrier shall not compromise the integrity and effectiveness of subsequent barriers.
		The nuclear security arrangements are designed in cognizance and in response to counter and mitigate the modern threat environment that stems from a dynamic, intelligent adversary, who acts in a deliberate, planned

Fundamental Objective	Level 1 Claims	Level 2 Claims
	0-85	fashion. Application of the Design Basis Threat (DBT) is used to determine these attributes and characteristics, as well as maintain presence of a credible threat in all phases of the plant design and operational lifecycles.
The BWRX-300 is capable of being constructed, operated, and decommissioned in accordance with the standards of safety, security, safeguards and environmental protection required in the UK.	**Safeguards** 0-86 Appropriate nuclear material accountancy is undertaken to minimise the potential nuclear materials to be used for non-peaceful purposes.	The design process for the BWRX-300 reactor has followed IAEA's guidance on International Safeguards in the Design of Nuclear Reactors.
		The BWRX-300 reactor may be operated according to the guidance in the ONR Nuclear Material Accountancy, Control, and Safeguards Assessment Principles.
		The BWRX-300 design considers safeguards' interface with safety, security and waste management issues.
	Environment The design of the BWRX-300 SMR has been optimised to reduce environmental effects to As Low As Reasonably Achievable (ALARA) throughout the whole lifecycle (construction, commissioning, operation, and decommissioning).	Prevention or, where this is not practicable, minimisation of the creation of radioactive waste and spent fuel.
		Minimisation of the activity of gaseous radioactive waste disposed of by discharge to the environment.
		Minimisation of the activity of aqueous radioactive waste disposed of by discharge to the environment.
		Minimisation of the volume of solid radioactive waste disposed of by transfer to other premises.
		Selection of the optimal disposal routes for wastes and spent fuel.
		Minimisation of the effect of radioactive discharges on members of the public and the environment.