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GE Hitachi Nuclear Energy

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BWRX-300 UK Generic Design Assessment (GDA) Chapter 1 – Introduction

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EXECUTIVE SUMMARY

This chapter introduces the structure of the Preliminary Safety Report (PSR) and gives brief descriptions of the chapters contained within it. The PSR is divided into 28 chapters defining the different aspects of safety (nuclear and industrial safety), security, safeguards and environmental protection.

Collectively these chapters review the key aspects of the Boiling Water Reactor X-300 (BWRX-300) for the United Kingdom (UK). This document provides an overview of the purpose, scope, and format of the PSR and a brief description of the BWRX-300 plant design.

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ACRONYMS AND DEFINITIONS

Acronym	Explanation
ALARP	As Low As Reasonably Practicable
AOO	Anticipated Operational Occurrences
BDBA	Beyond Design Basis Accident
BIS	Boron Injection System
BWR	Boiling Water Reactor
CAE	Claims, Arguments, and Evidence
CANDU	Canada Deuterium Uranium
CB	Control Building
CRD	Control Rod Drive
D-in-D	Defence-in-Depth
DBA	Design Basis Accident
DBE	Design Basis Earthquake
DBT	Design Basis Threat
DCIS	Distributed Control and Information System
DEC	Design Extension Condition
DL	Defence Line
DSA	Deterministic Safety Analysis
EA	Environment Agency
EC&I	Electrical Control & Instrumentation
EH	External Hazard
EOC	Emergency Operations Centre
EPS	Electrical Power System
ESBWR	Economic Simplified Boiling Water Reactor
ESF	Engineered Safety Feature
FAT	Factory Acceptance Testing
FMCRD	Fine Motor Control Rod Drive
FPC	Fuel Pool Cooling and Cleanup System
FSF	Fundamental Safety Function
GDA	Generic Design Assessment
GDF	Geological Disposal Facility
GEH	GE-Hitachi Nuclear Energy
HA	Human Actions
HAW	Higher Activity Waste
HCU	Hydraulic Control Unit
HFE	Human Factors Engineering

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Acronym	Explanation
HSI	Human System Interface
HVAC	Heating, Ventilation, and Air Conditioning
HX	Heat Exchanger
I&C	Instrument and Control
IAEA	International Atomic Energy Agency
IC	Isolation Condenser
ICC	Isolation Condenser Pool Cooling and Cleanup System
ICS	Isolation Condenser System
IH	Internal Hazards
ILW	Intermediate Level Waste
LOCA	Loss-of-Coolant Accident
MCR	Main Control Room
NM	Nuclear Material
NPP	Nuclear Power Plant
NRW	Natural Resources Wales
ONR	Office for Nuclear Regulation
ORM	Other Radioactive Material
PCCS	Passive Containment Cooling System
PER	Preliminary Environmental Report
PIE	Postulated Initiating Event
PLSA	Plant Service Area
PSA	Probabilistic Safety Analysis
PSR	Preliminary Safety Report
RB	Reactor Building
RCPB	Reactor Coolant Pressure Boundary
RG	Regulatory Guide
RP	Requesting Party
RPV	Reactor Pressure Vessel
RWB	Radwaste Building
SAR	Safety Analysis Reports
SAT	Site Acceptance Testing
SBWR	Simplified Boiling Water Reactor
SCCV	Steel-plate Composite Containment Vessel
SCDS	Safety Case Development Strategy
SCR	Secondary Control Room
SDC	Shutdown Cooling

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Acronym	Explanation
SFP	Spent Fuel Pool
SSCs	Structures, Systems, and Components
SSG	Specific Safety Guide
TB	Turbine Building
UK	United Kingdom
USNRC	U.S. Nuclear Regulatory Commission

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REVISION SUMMARY

Revision #	Section Modified	Revision Summary
A	All	Initial Issuance
B	All	Update for end of GDA Step 2 consolidation

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1 INTRODUCTION AND GENERAL CONSIDERATIONS

1.1 Introduction

This chapter Introduces the structure of the Preliminary Safety Report (PSR) and gives brief descriptions of the chapters.

The PSR is divided into 28 chapters defining the different aspects of safety (nuclear and industrial safety), security, safeguards and environmental protection. Collectively these chapters review the key aspects of the Boiling Water Reactor X-300 (BWRX-300) for the United Kingdom (UK).

The fundamental objective that will be demonstrated through the Generic Design Assessment (GDA), is that the proposed BWRX-300 could be commissioned, operated, and decommissioned in the UK on a site bounded by the generic site envelope while maintaining the safety and security of people and the environment.

The PSR is written based on the assumption of a single unit design built on a generic UK site that is bounding of all suitable locations i.e., the design meets or exceeds the requirements for all suitable locations.

This document outlines the intended chapters that constitute the PSR safety case structure and a brief overview of the BWRX-300.

1.1.1 Purpose of the Preliminary Safety Report

The PSR is the first major submission of the Generic Design Assessment (GDA) process. The purpose of the PSR is to provide sufficient information for the UK regulators to carry out step 2 of the GDA process. This document outlines the chapters that constitute the PSR safety case structure.

Step 1 is the preparatory part of the design assessment process. Mostly this will involve the Requesting Party (RP) setting up its project management, technical teams, arrangements for the GDA, and writing and preparing submissions for Step 2, including the PSR. It also involves discussions between the RP and Office for Nuclear Regulation (ONR) to ensure full understanding of the requirements and processes that will be applied.

Step 2 is an overview (by the ONR, Natural Resources Wales (NRW) and Environmental Agency (EA)) of the acceptability, in accordance with the UK regulatory regime, of the design fundamentals, including review of key safety, environment and security claims. Step 2 of the GDA process requires the RP to submit a PSR providing an outline description of the reactor equipment and structures, the design and safety philosophy, the codes and standards applied in the design and the quality management systems applied by the designers. The aim is to give the ONR and EA confidence that UK safety standards could be met by the proposed reactor design and that the claimed principles and design criteria are likely to be achievable.

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 objectives, identified through the PSR, have been developed to provide a clear basis for establishing the safety of the design. However, at this stage of the development of the safety case the arguments and underlying evidence to support the objectives have not yet been fully assembled. This will be completed as the GDA process progresses and this may necessitate some adjustment of the objectives during subsequent GDA phases.

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1.1.2 Scope of the Preliminary Safety Report

The PSR is divided into 28 chapters defining the key aspects of safety (nuclear and industrial safety), security and environmental protection. Collectively these chapters review the key aspects of the BWRX-300 for the UK.

The fundamental objective that will be demonstrated through the GDA is that the proposed BWRX-300 can be constructed, commissioned, operated, and decommissioned in the UK on a site bounded by the generic site envelope while maintaining the safety and security of people and the environment.

The PSR is written based on the assumption of a single unit design built on a generic UK site that is bounding of all suitable locations.

1.1.3 Objective

GEH has structured its submission using the 'Claims, Arguments and Evidence', or CAE (Reference 1-38), 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 main submission volumes covering safety, security, safeguards and the environment. These claims are then further subdivided and supported by arguments and evidence, although aspects of the evidence will only become available once the BWRX-300 enters the detailed design phase during site-specific licensing. The CAE structure is provided in full in Appendix A.

1.1.4 Format of the Safety Analysis Report

It is recognized by the International Atomic Energy Authority (IAEA) guidance that Safety Analysis Reports (SARs) are developed in an iterative manner to support the appropriate licensing activities at the appropriate time, "Format and Content of the Safety Analysis Report for Nuclear Power Plants, No. GS-G-4.1," (Reference 1-1). Since this release supports GEH's GDA application, this version of the SAR is a PSR and contains sufficient information in line with the design progression to assess and demonstrate the plant can be safely constructed. The PSR will be updated with more detail design information as the design progresses.

The following describes the format of the GEH PSR and includes a brief description of each chapter. Information presented in each chapter is relative to the importance to nuclear safety and the PSR purposes.

PSR Chapter 1: Introduction

The information in this chapter describes how the IAEA guidance has been applied to the GEH BWRX-300 facilities at a high level and the PSR process including their purposes and objectives. Alignment with ONR and EA requirements during the production of the PSR is also outlined here.

PSR Chapter 2: Site Characteristics, NEDC-34164P, "BWRX-300 UK GDA Ch. 2: Site Characteristics," (Reference 1-2)

The UK GDA process is based upon a generic site and this chapter will define the characteristics required of the site. This chapter will describe site information that can be defined on a generic basis, such as the heat sink, grid connections, density, and distribution

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of local population as well as soil. Site characteristics for the Environmental issues are presented separately within the Preliminary Environmental Report (PER).

PSR Chapter 3: Safety Objectives and Design Rules for Structures, Systems, and Components. NEDC-34165P, “BWRX-300 UK GDA Ch. 3: Safety Objectives and Design Rules for SSCs,” (Reference 1-3)

This chapter will present the safety design basis in a UK context. It will address topics including As Low As Reasonably Practicable (ALARP); UK dose targets and limits; the Defence-in-Depth (D-in-D) principle and its application; deterministic design principles; equipment qualification and aging management. The chapter will also present the methodology for categorization and classification of Systems, Structures and Components (SSCs).

PSR Chapter 4: Reactor (Fuel and Core), NEDC-34166P, “BWRX-300 UK GDA Ch. 4: Reactor (Fuel and Core),” (Reference 1-4)

This chapter describes the design of the BWRX-300 Reactor and fuel assembly in detail. This chapter also provides the information on the reactor which demonstrates its capability to fulfil relevant safety functions throughout the design life in all stages of the plant.

PSR Chapter 5: Reactor Coolant System and Associated Systems, NEDC-34167P, “BWRX-300 UK GDA Ch. 5: Reactor Coolant System and Associated Systems,” (Reference 1-5)

This chapter will describe the relevant Reactor Coolant systems and the associated system groups, their required safety and non-safety functions, design bases and provides arguments as to how the functions will be met by the systems. System interfaces / dependencies will also be identified.

PSR Chapter 6: Engineered Safety Features, NEDC-34168P, “BWRX-300 UK GDA Ch. 6: Engineered Safety Features,” (Reference 1-6)

This chapter describes the Engineered Safety Features (ESFs) provided to mitigate the consequences of Anticipated Operational Occurrences (AOOs) and Design Basis Accidents (DBAs) for the BWRX-300 without any core damage, and how these systems comply with their design and safety requirements. ESF design features include passive systems that do not require dependence on external sources of power or operator action to fulfil the Fundamental Safety Function (FSF)

PSR Chapter 7: Instrumentation and Control, NEDC-34169P, “BWRX-300 UK GDA Ch. 7: Instrumentation and Control,” (Reference 1-7)

This chapter describes the relevant systems within the Instrument and Control (I&C) system group, their required safety and non-safety functions, design bases and provides arguments as to how the functions are met by the systems. System interfaces and dependencies will be identified and coordinated with other relevant chapters.

PSR Chapter 8: Electrical Power, NEDC-341570P, “BWRX-300 UK GDA Ch. 8: Electrical Power,” (Reference 11-8)

This chapter describes the Electrical Power System (EPS) that is relied upon to support the plant safety strategy, the general architecture of the electrical system and how this system will comply with its design and safety requirements.

PSR Chapter 9A: Auxiliary Systems, NEDC-34171P, “BWRX-300 UK GDA Ch. 9A: Auxiliary Systems,” (Reference 1-9)

This chapter describes the auxiliary systems of the BWRX-300 that supports its safe and reliable operations, including fuel handling, water, air, heating, ventilation, and air conditioning (HVAC), fire protection, diesel generators and overhead lifting equipment.

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PSR Chapter 9B: Civil Structures, NEDC-34172P, “BWRX-300 UK GDA Ch. 9B: Civil Structures,” (Reference 1-10)

This chapter describes the design of the safety classified structures of the BWRX-300 considered within the scope of the UK GDA, and how these will comply with their design and safety requirements e.g., Radwaste Building (RWB), Control Building (CB), Turbine Building (TB), Plant Service Area (PLSA) and intake and discharge structures.

PSR Chapter 10: Steam and Power Conversion Systems, NEDC-34173P, “BWRX-300 UK GDA Ch. 10: Steam and Power Conversion Systems,” (Reference 1-11)

This chapter describes the design of the power conversion system, including the main turbine equipment, which is comprised of high and low power turbines, turbine gland seal system, turbine lubricating oil system, extraction steam system, electro-hydraulic control system, and the Turbine Auxiliary steam subsystem. This chapter will also describe how these are designed to produce electrical power utilizing the steam generated by the reactor and how they will comply with their design and safety requirements.

PSR Chapter 11: Management of Radioactive Waste, NEDC-34174P, “BWRX-300 UK GDA Ch. 11: Management of Radioactive Waste,” (Reference 1-12)

This chapter will describe the strategy for the management of each waste stream. This chapter demonstrates that it will be possible to design and operate a facility safely for the management of Intermediate Level Waste (ILW), including processing and packaging into a passively safe and disposable form and its onsite storage until such time that Geological Disposal Facility (GDF) becomes available.

PSR Chapter 12: Radiation Protection, NEDC-34175P, “BWRX-300 UK GDA Ch. 12 Radiation Protection,” (Reference 1-13)

This chapter describes the radiation protection design features, the design principles and methodology of the BWRX-300, that ensure occupational radiation exposure to personnel is kept ALARP. The method in which the ALARP principles are applied during plant operation, maintenance, decommissioning, and post-accident is also described in this chapter.

PSR Chapter 13: Conduct of Operations, NEDC-34176P, “BWRX-300 UK GDA Ch. 13: Conduct of Operations,” (Reference 1-14)

This chapter describes, at a high level, how the BWRX-300 design, and operational documentation produced for GDA step 2 can enable a future duty holder/licensee to implement the safety case through organizational structure/arrangements, training, implementation of the operational safety program, plant procedures and guidelines, nuclear safety, and nuclear interfaces.

PSR Chapter 14: Plant Construction and Commissioning, NEDC-34177P, “BWRX-300 UK GDA Ch. 14: Plant Construction and Commissioning,” (Reference 1-15)

This chapter describes the assessment and specifications of the BWRX-300 plant construction and commissioning including, but not limited to, civil works, mechanical systems, electrical systems, Electrical, Control & Instrumentation (EC&I), ancillary and auxiliary systems, and environmental and habitability systems. It will also consider the range of Factory Acceptance Testing (FAT) and Site Acceptance Testing (SAT) with respect to safety considerations.

PSR Chapter 15: Safety Analysis

This chapter has been split into the following subchapters to ensure all aspects of safety analysis are sufficiently assessed.

15.1: General Considerations, NEDC-34179P, “BWRX-300 UK GDA Ch. 15.1: Safety Analysis- General Considerations,” (Reference 1-16)

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This subchapter provides an introduction to the safety analysis chapter, covering both deterministic and probabilistic analysis. This subchapter includes a description of the scope of the safety analysis and the approach adopted (i.e., conservative, or realistic, as appropriate) for each plant state, for normal operations to Design Extension Conditions (DECs) with core melting. This subchapter also explains how any previously identified generic issues and relevant Operating Experience (OPEX) have been used to enhance the quality of the safety analysis.

15.2: ID Cat & Grouping of Postulated Initiating Events & Accidents, NEDC-34180P, “BWRX-300 UK GDA Ch. 15.2 Safety Analysis- ID, Categorisation and grouping of PIEs and Accident Scenarios,” (Reference 1-17)

This subchapter provides the approach used to identify Postulated Initiating Events (PIE) and accident scenarios for both the deterministic and probabilistic analyses. This includes the use of analytical methods such as screening of D-in-D, master logic diagrams, hazards and operability analysis, and failure and effects analysis.

15.3: Safety Objectives and Acceptance Criteria, NEDC-34181P, “BWRX-300 UK GDA Ch. 15.3: Safety Analysis – Safety Objectives and Acceptance Criteria,” (Reference 1-18)

This subchapter provides the safety objectives of the PSR for the GDA of the BWRX-300. This subchapter will provide a description of how the safety analysis refers to the safety principles and objectives introduced in chapter 3, and the general approach to the design of BWRX-300 SSCs.

15.4 Human Actions, NEDC-34182P, “BWRX-300 UK GDA Ch. 15.4: Safety Analysis – Human Actions,” (Reference 1-19)

The purpose of this subchapter is to describe the approach to identify and model Human Actions (HAs) in the BWRX-300 deterministic and Probabilistic Safety Analysis (PSA). It also describes the approach to substantiation of human actions. The subchapter will present a level of detail commensurate with a two-step GDA (claims and arguments only) and will be structured in line with IAEA Specific Safety Guide (SSG)-61 (noting that SSG-61 does not differentiate between the level of detail required in a PSR and later more detailed safety reports)

15.5: Deterministic Safety Analysis, NEDC-34183P, “BWRX-300 UK GDA Ch. 15.5: Safety Analysis – Deterministic Safety Assessment,” (Reference 1-20)

This subchapter defines the initiating faults and hazards that are reasonably foreseeable, conservatively justifies accident sequences that follow those faults and hazards and assesses the design against engineering principles. The purpose is to demonstrate the fault tolerance of the design, the effectiveness of the safety measures, and to support the claim that all risks associated with the design and its operation have been reduced to ALARP. The Deterministic Safety Analysis (DSA) does not quantify risk. Instead, the adequacy of the design and the suitability and sufficiency of the safety measures are assessed against deterministic targets.

15.6: Probabilistic Safety Assessment, NEDC-34184P, “BWRX-300 UK GDA Ch. 15.6: Safety Analysis – Probabilistic Safety Assessment,” (Reference 1-21)

This subchapter provides a description of the PSA that has been undertaken for the BWRX-300, with an overview of the results and comparison with the safety goals and numerical targets. This subchapter will cover all operating modes, internal events, and internal and External Hazards (EHs), reporting on the work done to date and plans for future work. There will be consideration of all potential sources of radiological release on site, including the Spent Fuel Pool (SFP). It will also present a discussion describing how the PSA has and will continue to support risk-informed design and decision making and support the claim that the BWRX-300 risk is ALARP.

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15.7: Internal Hazards, NEDC-34185P, “BWRX-300 UK GDA Ch. 15.7: Safety Analysis – Internal Hazards,” (Reference 1-22)

This subchapter will provide a description of the Internal Hazards (IH) to be considered within the BWRX-300 GDA PSR. The subchapter will explain the identification process, assessment methodologies, and demonstrate the tolerance for the IHs of the BWRX-300 design.

15.8: External Hazards, NEDC-34186P, “BWRX-300 UK GDA Ch. 15.8: Safety Analysis – External Hazards,” (Reference 1-23)

This subchapter will provide a description of the derivation of EHs to be considered within the BWRX-300 GDA PSR. It will explain the process used to systematically identify and screen natural and man-made hazards. It will summarize the measures inherent in the design to ensure that the fundamental safety functions and the SSCs that deliver them are protected against design basis EHs and combinations thereof.

15.9: Summary of Results of the Safety Analyses, NEDC-34187P, “BWRX-300 UK GDA Ch. 15.9: Safety Analysis – Summary of Results of the Safety Analysis including Fault Schedule,” (Reference 1-24)

This subchapter provides a summary of the overall results of the safety analysis for each event category and covers deterministic and probabilistic analysis. The subchapter should also confirm that all relevant nuclear plant design expectations have been met, and a route for completion of any outstanding aspects should be defined.

PSR Chapter 16: Operational Limits and Conditions of Safe Operation, NEDC-34188P, “BWRX-300 UK GDA Ch. 16: Operational Limits and Conditions of Safe Operation,” (Reference 1-25)

Information in this chapter describes how the facility’s safe operating envelope is evaluated and implemented through a set of operational limits and conditions that prescribe boundaries within which the BWRX-300 operates to assure compliance with the safety analysis inputs, assumptions, and results. The full set of operational limits and conditions is a key element of the licensing basis for a License to Operate, however, the content of individual Technical Specifications is outside the scope of this GDA.

PSR Chapter 17: Management of Safety and Quality Assurance, NEDC-34189P, “BWRX-300 UK GDA Ch. 17: Management for Safety and Quality Assurance,” (Reference 1-26)

The information in this chapter describes how the overall management of all safety related activities is assured throughout the lifecycle of the facility. It describes the general and specific quality management, performance improvement, and safety culture elements of the management systems of the site organization and GEH, which support the development, operation, and eventual retirement of the facility.

PSR Chapter 18: Human Factors Engineering, NEDC-34190P, “BWRX-300 UK GDA Ch. 18: Human Factors Engineering,” (Reference 1-27)

This chapter describes the Human Factors Engineering (HFE) program for the BWRX-300, including arguments to demonstrate the integration of HFE requirements and analyses results into the plant design. This chapter also includes a summary of the BWRX-300 Human System Interface (HSI) design goals and bases, the concept of operation, the analyses undertaken to understand the plant specific HFE requirements related to task performance, the process for HSI, Human Factors verification and validation, and the HF contribution to the identification and substantiation of HAs important to safety in the BWRX-300 safety analyses.

PSR Chapter 19: Emergency Preparedness and Response, NEDC-34191P, “BWRX-300 UK GDA Ch. 19: Emergency Preparedness and Response,” (Reference 1-28)

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The information in this chapter will provide a description of emergency arrangements that will be developed and will be capable of protecting workers (including emergency workers), the public, and the environment in the event of a radiological emergency.

PSR Chapter 20: Environmental Aspects, NEDC-34192P, “BWRX-300 UK GDA Ch. 20: Environmental Aspects,” (Reference 1-29)

Information in this chapter describes the environmental aspects associated with the BWRX-300. It gives a brief description of the approach taken to assess the environmental impacts from construction, operation and decommissioning of the plant. The report looks to confirm that proposed approaches to manage environmental impacts from the BWRX-300 are currently or can in future be compatible in the future with industry best practice and aligned with relevant UK policy, legislation, regulation, and regulatory guidance.

PSR Chapter 21: Decommissioning and End of Life Aspects, NEDC-34193P, “BWRX-300 UK GDA Ch. 21: Decommissioning and End of Life Aspects,” (Reference 1-30)

The information in this chapter demonstrates the set principles of safe decommissioning, including how they have been derived from RGP and OPEX. This includes a description of how the principles for facilitation of safe decommissioning have been applied in the design of the BWRX-300 and how the design process promotes further challenges to ensure decommissioning risks are reduced to ALARP.

PSR Chapter 22: Structural Integrity, NEDC-34194P, “BWRX-300 UK GDA Ch. 22: Structural Integrity,” (Reference 1-31)

This chapter will justify the structural integrity of metallic SSCs of the UK BWRX-300 design and demonstrate that the risk of structural failure is ALARP. The structural reliability will be justified accordingly to the consequences of their failure, as established by a system of component structural integrity classification, which introduces additional requirements for components classified as High Integrity, which are over and above Standard Class 1 SSCs

PSR Chapter 23: Reactor Chemistry, NEDC-34195P, “BWRX-300 UK GDA Ch. 23: Reactor Chemistry,” (Reference 1-32)

This chapter will provide a functional description of the reactor chemistry management and control approach to be implemented for the BWRX-300. Although mainly focused on the main cooling circuit, chemistry regimes will be described and justified for each of the relevant fluid containing systems, where chemistry management and control are important for maintaining plant integrity and safety functions. Additionally, this chapter provides a description of the chemistry injection and dosing systems, the process radiation and environmental monitoring system that measures key chemistry/ radiochemistry control parameters and diagnostic parameters.

PSR Chapter 24: Conventional Safety and Fire Safety, NEDC-34196P, “BWRX-300 UK GDA Ch. 24: Conventional Safety and Fire Safety,” (Reference 1-33)

This chapter will define the fire protection design criteria and how the BWRX-300 meets these requirements. The purpose of this chapter is to summarize the BWRX-300 conventional health and safety and conventional fire safety assessment work for the GDA. The intent of this work is to demonstrate that there are no known design aspects that could prevent compliance with the UK’s conventional health and safety or fire safety requirements and regulatory expectations.

PSR Chapter 25: Security, NEDC-34197P, “BWRX-300 UK GDA Ch. 25: Security Annex,” (Reference 1-34)

This chapter sets out the overall approach to security operations. The information within the Security Assessment and supporting annexes represents a component part of the security informed design that supports defined security outcomes. The security outcomes concerning

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response and mitigative effects are based on the Design Basis Threat (DBT), protection of the Nuclear Material (NM) and Other Radioactive Material (ORM) inventory, the safety SSC that maintains it in a safe state, and security SSC that protects these.

PSR Chapter 26: Interim Storage of Spent Fuel, NEDC-34198P, “BWRX-300 UK GDA Ch. 26: Interim Storage of Spent Fuel,” (Reference 1-35)

This chapter presents information on the spent fuel management arrangements for the BWRX-300 and demonstrates that they can be developed to comply with relevant UK policy, legislation, regulations, and regulatory guidance. Waste storage is an essential component of the Higher Activity Waste (HAW) management lifecycle and provides a safe, secure environment for waste packages awaiting final disposal.

PSR Chapter 27: ALARP Evaluation, NEDC-34199P, “BWRX-300 UK GDA Ch. 27: ALARP Evaluation,” (Reference 1-36)

This chapter presents the ALARP evaluation approach for nuclear and conventional safety of the BWRX-300 so that it could be demonstrated that the design has reduced risk to operators and members of the public as far as reasonably practicable in design development. The maturity of the ALARP demonstration for a two-step GDA is proportionate to the status of the BWRX-300 design.

PSR Chapter 28: Safeguards Annex, NEDC-34200P, “BWRX-300 UK GDA Ch. 28: Safeguards Annex,” (Reference 1-37)

This chapter will present NM Safeguards for the BWRX-300 and will demonstrate that they can be developed to comply with the UKs international treaty obligations and relevant domestic policy, legislation, regulations, and regulatory guidance.

1.2 Project Implementation

The BWRX-300 has been developed in accordance with approved procedures, with appropriate governance and assurance arrangements by a competent and clearly defined organization. Suitable organizational arrangements and appropriate governance are in place to control and manage the design and substantiation of the BWRX-300. Future arrangements can be developed to support an operational facility including normal and emergency arrangements.

1.3 Identification of Interested Parties Regarding Design, Construction and Operation

GEHs BWRX-300 has entered into the GDA process with the objective of gaining regulatory confidence in the acceptability of a conceptual full plant design through Steps 1 and 2 of GDA. It is the intention of GEH to exit GDA at the end of Step 2 before progressing directly to Site Licensing.

The Developer, GEH, is the Design Authority for scoped design activities in accordance with the project execution model until turnover to the Licensee at which point GEH is likely to be appointed as a Responsible Designer. The Licensee is responsible for all activities on site including procurement, construction, commissioning and eventual commercial operation. The Licensee will be supported by a number of contract partners throughout these phases.

Roles and responsibilities for all contract partners, including quality and safety, are defined and accepted through contractual agreements for the project. The partner roles are further described in PSR Chapter 17 (Reference 1-24).

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1.4 General Plant Description

The BWRX-300 is a Boiling Water Reactor (BWR) that employs natural circulation and passive emergency cooling features and is rated at approximately 300 megawatts-electric.

The passive design features of the BWRX-300 provide decay heat removal capability using only installed systems with no reliance on operator actions or external resources for at least 72-hours. For the BWRX-300, a safe stable condition ("stable shutdown") is defined as safe shutdown with average reactor coolant temperature $\leq 215.6^{\circ}\text{C}$ (420°F). Following 72-hours post-accident, onsite or off-site resources are used to power non-safety equipment for proceeding to cold shutdown conditions, as needed.

The BWRX-300 design applies a Defence-in-Depth process for safety assessment and safety analysis to ensure that radiological acceptance criteria are met. The leveraging of passive design features greatly simplifies the design and results in a significant reduction in the total number active SSCs compared to conventional Nuclear Power Plants (NPPs).

The overall safety objectives and the safety strategy employed in the development of the BWRX-300 design are described in detail in PSR Chapter 3 (Reference 1-3).

Basic Technical Characteristics

The principal technical characteristics of the BWRX-300 are provided in Table 1-1.

1.5 Comparison with Other Plant Designs

The BWRX-300 is based on the U.S. Nuclear Regulatory Commission (USNRC) licensed, 1520 MWe Economic Simplified Boiling Water Reactor (ESBWR). The ESBWR is an evolution of the 600 MWe Simplified Boiling Water Reactor (SBWR) that has a significant testing and qualification program directly applicable to the BWRX-300.

The BWRX-300 is the tenth generation of the BWR that incorporates the lessons learned in design, construction, operations, and maintenance from over 100 previous BWRs that have been built, operated, and in some cases, decommissioned.

The BWRX-300 is specifically designed to enhance safety through simplification and reducing its dependence on human intervention. This is achieved through increasing its reliance on natural circulation and natural phenomena-driven safety systems. These safety enhancements, in combination with its reduction in scale and complexity, enable reductions in operating staff, maintenance, and security requirements as well as being easier to decommission.

1.5.1 The BWRX-300 provides clean and flexible baseload electricity at a lifecycle cost that is much lower than the previous generation of NPPs operating today and competitive with other forms of electricity generation such as natural gas, combined-cycle plants and renewables. Additional Information Concerning New Safety Features

Though mostly traditional in BWR design, the BWRX-300 includes several design features that simplify the design and enhance safety, such as:

Reactor Isolation Valve location: The BWRX-300 RPV is equipped with Reactor Isolation Valves which rapidly isolate a ruptured pipe to help mitigate the effects of a Loss-of-Coolant Accident (LOCA). All large fluid pipe systems are equipped with double isolation valves which are integral to the RPV. The valves are located as close as possible to the RPV.

No Safety Relief Valves: Safety relief valves have been eliminated from the BWRX-300 design. The large capacity Isolation Condenser System (ICS) provides overpressure protection. Historically on BWRs, the safety relief valve inadvertent actuation has been the most likely cause of a LOCA and have, therefore, been eliminated from the BWRX-300 design.

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Dry containment: The BWRX-300 has a dry containment that is cooled through natural circulation during DBAs. This has been proven to effectively contain the releases of steam, water, and fission products after a LOCA.

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 emergency core cooling system provided by the ICs eliminates the need for active emergency core cooling injection systems while ensuring greater safety margins than predecessor BWRs.

No need for emergency diesel generators: Elimination of active emergency core cooling systems eliminates the need for onsite emergency power systems. Standby diesel generators are provided for asset protection only i.e., to protect valuable equipment and systems from potential damage during power outages or other emergencies.

Table 1-2 demonstrates how the BWRX-300 design has evolved to maximize passive safety features to achieve FSF in comparison to the design of previous BWR and other types of NPPs.

1.5.2 Industry Incident Reviews

Station Blackout events have historically been the most demanding for BWRs to cope with and have usually been the dominant sequence for Severe Accident scenarios. The BWRX-300 is an advanced passive reactor design that does not require active safety systems. The BWRX-300 design carried forward the passive ICS and containment cooling concepts from the ESBWR. DC power sources are assumed to be available. The systems that support FSF and plant monitoring are designed to operate for 72-hours, without AC power, and without an intake structure that normally provides cooling water. The ICS pools and SFP have enough inventory to provide adequate decay heat removal and fuel cooling for seven days, after which alternate water makeup sources (e.g., flexible mitigation/EME) are used to refill the pools. The Passive Containment Cooling System (PCCS) is designed to passively limit containment pressure and temperature by transferring heat to the equipment pool. The demonstration of plant safety functions during a beyond design basis external event such as an earthquake that creates these conditions is typically part of the diverse and flexible coping strategies that form the basis for compliance of regulatory requirements related to the Fukushima tsunami event.

In April 2012, the Institute of Nuclear Power Operations conducted an independent review of the Fukushima nuclear accident with the purpose of identifying operational and organizational lessons learned from the accident. The results of this review are well documented.

The Fukushima accident was a Beyond Design Basis event. DECAs are a selected subset of Beyond Design Basis Accident (BDBA) conditions.

The BWRX-300 is designed for DECAs, and these are described in detail in 006N5064, "BWRX-300 Safety Strategy," (Reference 1-40).

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1.6 Information on the Plant Layout and Other Aspects

The BWRX-300 Major systems, RPV and internals, and ICSs are shown in Section 1.7.

Descriptions of the Reactor, Turbine, Control, and RWB, the plant services area and reactor auxiliary bay are outlined here. Figure 1-1 shows the proposed layout of these buildings:

Reactor Building

The Reactor Building (RB) is a Safety Category 1 and Seismic Category A structure. It is a cylindrical shaped structure embedded in a vertical shaft to a depth of approximately 36 m below-grade. The Reactor Pressure Vessel (RPV), Steel-plate Composite Containment Vessel (SCCV) and other important systems and components are located in the deeply embedded RB vertical right-cylinder shaft to mitigate effects of external events, including aircraft impact, adverse weather, fires, and earthquakes. The Secondary Control Room (SCR) is located in the RB. The below-grade portion also contains reactor support and safety class systems and the Safety Class 1 power supply and equipment. The reactor cavity pool is above the containment dome. Also, within the RB, three separate ICS pools sit next to the reactor cavity pool above the SCCV, with one IC located in each pool. The Fuel Pool is also located in the RB.

Turbine Building

The TB houses the steam turbine generator, standby diesel generators, main condenser, condensate and feedwater systems, turbine generator support systems, and parts of the Off-gas System (excluding the off-gas charcoal adsorbers).

While considered a separate functional area from the TB, the northern portion of the PLSA is structurally integrated with the TB. See below for a description of the PLSA.

The TB is a Safety Class 2 structure and is categorized as non-seismic i.e., it is not designed to withstand seismic events such as earthquakes. However, it is evaluated for seismic interaction to ensure that it will not compromise the structural integrity and safety functions of the adjacent Seismic Category A RB following a Design Basis Earthquake (DBE) or extreme tornado wind conditions.

Control Building

The CB houses the Main Control Room (MCR), Emergency Operations Centre (EOC), electrical, control, and instrumentation equipment. The CB is a Safety Class 2 structure and is categorized as non-seismic. Additionally, it is evaluated for seismic interaction to ensure that it does not compromise the structural integrity and safety functions of the adjacent Seismic Category A RB following a DBE or extreme tornado wind conditions. The CB serves as the main entrance and exit for the Power Block unit during normal operations.

While considered a separate functional area from the Control Building, the southern portion of the PLSA is structurally integrated with the Control Building. See below for a description of the PLSA.

Radwaste Building

The RWB houses rooms and equipment for handling, processing, and packaging liquid and solid radioactive wastes as well as the Off-gas System charcoal adsorbers that are used for processing radioactive gas. Some of these systems contain highly radioactive materials. The RWB is classified as a Safety Class 3 building and categorized as RW-IIa in accordance with Regulatory Guide (RG) 1.143, Rev. 2 (Reference 1-39). Additionally, it is also evaluated for seismic interaction to ensure that it will not compromise the structural integrity or safety functions of the adjacent Seismic Category A RB following a DBE or extreme Tornado wind conditions.

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Plant Services Area and Reactor Auxiliary Bay

The PLSA houses a decontamination area, a contaminated part/tool storage room, an I&C calibration room, a truck space for cask removal, a hot machine shop, laydown areas for new fuel and the Fine Motion Control Rods Drive (FMCRD), and a miscellaneous storage area.

While the PLSA is a separate functional area from the CB and TB, the northern portion of the PLSA shares a foundation and is structurally integrated with the TB and the southern portion of the PLSA shares a foundation and is structurally integrated with the CB.

A portion of the PLSA, the Reactor Auxiliary Bay, is constructed on a separate foundation with respect to the portions of the PLSA that are adjacent to the CB and TB. The functions performed in the Reactor Auxiliary Bay include new fuel and spent fuel cask transit, equipment ingress and egress to the RB, and personnel access to the RB. The Reactor Auxiliary Bay is a Safety Class 2 structure and is categorized as non-seismic. Additionally, it is evaluated for seismic interaction to ensure that it does not compromise the structural integrity and safety functions of the adjacent Seismic Category A RB following a DBE or extreme Tornado wind conditions.

1.7 Drawings and Other More Detailed Information

A simplified representation of the major BWRX-300 systems and the flow of the reactor coolant is provided in Figure 1-2. A summary description of the major nuclear steam supply systems and components is provided below. Each of these systems are described in detail in applicable chapters of the PSR.

1.7.1 Reactor Pressure Vessel and Internals

The RPV is a vertical, cylindrical pressure vessel fabricated with forged rings and rolled plate welded together, with a removable top head, head flange, seals, and bolting. The vessel also includes penetrations, nozzles, and the shroud support. The RPV has a minimum inside diameter of approximately 4 m, a wall thickness of approximately 14 cm with cladding, and a height of approximately 26 m. The bottom of the active fuel region is approximately 5.2 m from the bottom of the vessel and the active core is 3.8 m high. The vertical orientated and tall vessel permits the development of natural circulation driving forces to produce sufficient core coolant flow.

A diagram of the BWRX-300 RPV assembly is shown in Figure 1-3. The RPV, together with its internals, provides guidance and support for the FMCRDs.

The major reactor internal components include:

- Core (fuel, channels, control rods and instrumentation)
- Core support and alignment structures (shroud, shroud support, top guide, core plate control rod guide tube, Control Rod Drive (CRD) housings, and orificed fuel support)
- Chimney
- Chimney head and steam separator assembly.
- Steam dryer assembly.
- Feedwater spargers
- In-core guide tubes

The fuel assemblies (including fuel rods and channels), control rods, chimney head, steam separators, steam dryer, and in-core instrumentation assemblies are removable when the reactor vessel is opened for refuelling or maintenance. The RPV shroud support is designed to support the shroud, as well as the components connected to the shroud, including the steam

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separator, chimney, core plate, and top guide. The fuel bundles are supported by the orifice fuel support, the control rod guide tube, and the CRD housing.

1.7.2 Reactor Pressure Vessel Isolation Valves

The BWRX-300 reactor incorporates isolation valves attached directly to the RPV. The function of the isolation valves is to close, limiting the loss of coolant from large and medium pipe breaks. The RPV isolation concept consists of two Reactor Isolation Valves in series. Each of the Reactor Isolation Valves is independently able to isolate the line.

1.7.3 Control Rod Drive System

The CRD system includes three major elements: FMCRD mechanisms; Hydraulic Control Unit (HCU) assemblies; and the Control Rod Drive Hydraulic subsystem. The FMCRDs are designed to provide electric-motor-driven positioning for normal insertion and withdrawal of the control rods and hydraulic powered rapid control rod insertion (scram) in response to manual or automatic signals. The hydraulic power required for scram is provided by high-pressure water stored in the individual HCUs. In addition to hydraulic powered scram, the FMCRDs also provide electric motor-driven run-in of all control rods as a path to rod insertion that is diverse from the hydraulic powered scram.

1.7.4 Isolation Condenser System

The ICS removes decay heat after any reactor isolation and shutdown event during power operations. The ICS decay heat removal limits increase in steam pressure and maintains the RPV pressure and water inventory at an acceptable level. The ICS consists of three independent loops that each contain a Heat Exchanger (HX) with capacity of approximately 33 MW, or approximately 3.7% of rated thermal power. Thermal energy removal condenses steam on the tube side and transfers heat by heating/evaporating water in the Isolation Condenser (IC) pools which are vented to the atmosphere. The arrangement of the ICS HX is shown in Figure 1-4.

1.7.5 Instrumentation and Control

The BWRX-300 Distributed Control and Information System (DCIS) is an integrated control and monitoring system for the power plant. The DCIS is arranged in three safety classified DCIS segments and a non-safety class segment with appropriate levels of hardware and software quality corresponding to the system functions they control and their Defence Line (DL) location. The DCIS provides control, monitoring, alarming and recording functions. The various bus segments of the integrated DCIS are designed to operate autonomously.

Control of reactivity in various postulated events is achieved by the instrumentation and control systems. Channels, trip logic, trip actuators, manual controls, and scram logic circuitry initiates the rapid insertion of control rods by hydraulic force to scram the reactor when unsafe conditions are detected. The hydraulic scram is actuated on signals derived from safety analyses and includes signals such as high core neutron power, RPV pressure, low RPV level, high containment temperature and high steam line flow.

1.7.6 Containment

The BWRX-300 Primary Containment Vessel encloses the RPV and some of its related systems and components. The Primary Containment Vessel is a leak-tight nitrogen inert gas space surrounding the RPV and the Reactor Coolant Pressure Boundary (RCPB). It provides a leak-tight barrier to prevent the release of radioactive fission products, steam, and water in the unlikely event of a LOCA. The BWRX-300 uses a traditional containment system for the ultimate containment of radioactive materials for various postulated events. The containment shape is a vertical cylinder approximately 18 meters outside diameter and 38 meters high. It is integral to and surrounded by the RB.

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1.7.7 Passive Containment Cooling System

The PCCS is a passive containment heat removal system that maintains the containment within its pressure limits for DBAs such as a LOCA. It consists of several low-pressure natural circulation HXs that transfer heat from the containment to the reactor cavity pool which is located above the containment upper head and is filled with water during normal operation. The reactor cavity pool is vented to the atmosphere. PCCS operation requires no sensing, control, logic, or power actuated devices for operation.

1.7.8 Boron Injection System

The Boron Injection System (BIS) is a complementary design feature that provides an additional means to place the plant in a cold shutdown mode. The BIS provides an additional means of negative reactivity insertion to bring the reactor subcritical during events when the control rod insertion (hydraulic and motor) is not successful.

1.7.9 Reactor Water Cleanup System

The Reactor Water Cleanup System provides the design functions of a cleanup flow path from the RPV to filter/demineralizer skids during most reactor operating modes. The cleanup or filtration function and ion removal function is performed by the condensate system.

1.7.10 Shutdown Cooling System

The Shutdown Cooling (SDC) System is designed to support RPV Startup and Shutdown/Cooldown Operations. The SDC consists of two independent trains with a motor-driven pump, a HX, required valves, piping, controls, and power inputs.

1.7.11 ICS Pool Cooling and Cleanup System

The Isolation Condenser Pool Cooling and Cleanup System (ICC) is designed to maintain cool and clean water in the ICS pools.

The primary function of the ICC is to remove heat from the ICS pools such that the bulk temperature of water in the pools is maintained below prescribed limits, and thereby ensure the readiness of the ICS to perform its safety function. Secondary functions of the ICC include maintaining the cleanliness of the ICS pool water and providing the capability to add clean makeup water during normal reactor operations to offset the routine and minor loss of water inventory due to evaporation.

1.7.12 Fuel Pool Cooling and Clean System

The Fuel Pool Cooling and Cleanup System (FPC) provides continuous cooling by removal of the decay heat from the spent fuel and maintains the Fuel Pool temperature below specified values. The system also maintains water level and water quality in the fuel pool, and reactor cavity pool. The FPC consists of one cooling and cleanup train provided with 100% capacity during normal operation (including pool maximum heat load).

1.7.13 Containment Inerting System

The Containment Inerting System precludes the combustion of hydrogen and prevents damage to essential equipment and structures. It establishes and maintains an inert atmosphere ($\leq 4\%$ dry-basis-percent oxygen) within containment during plant operating modes except during shutdown for refuelling/maintenance and for limited periods of time during low power operation for inspection. The system also maintains a slightly positive pressure in containment to prevent air (oxygen) in-leakage into the inerted spaces from the RB.

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Table 1-1: Principal Characteristics of Interest for the BWRX-300

Parameter Description	Value	Comments
Type of plant	Boiling Water Reactor	
Core coolant	Light Water	
Neutron moderator	Light Water	
Nuclear Steam Supply System layout	Direct-Cycle	
Primary circulation	Natural	
Thermodynamic cycle	Rankine	
Type of containment structure	Dry	
Reactor thermal power level	870 MWth	
Normal Heat Sink	Once Through Cooling System	
Ultimate Heat Sink	ICS pools	Pools are vented to atmosphere
Plant gross electrical power output	~ 300 MWe	
Plant Footprint	~ 9,800 m ²	Rectangular building envelope
Site Footprint	~ 30,000 m ²	Fenced area
Design life	60 years	
Exclusion Zone	350 m (radius)	Measured from exterior of the RB
Seismic Design (DBE)	0.310 g (horizontal & vertical)	Bounding rock peak ground acceleration
	0.532 g (horizontal)	Bounding surface peak ground acceleration
	0.516 g (vertical)	
Reactor Design Pressure	10.3 MPa	
Fuel	UO ₂ pellets	

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Parameter Description	Value	Comments
Fuel enrichment	<5% U-235	
RPV diameter (ID)	~ 4 m	
RPV height (Inside)	~ 26 m	
Control rod drive type	FMCRD	
Containment Vessel type	Steel-plate Composite	
Fuel pool capacity	Up to 8 years of full-power operation	Fuel pool accommodates up to 8 years of spent fuel plus one core load of new fuel and one full core off-load
Refuelling cycle	12 - 24 months	
Emergency Power Supply	Safety Class 1 DC batteries	Capable of sustaining required loads for 72 hours

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Table 1-2: Comparison of BWRX-300 to Other Nuclear Power Plant Types

Fundamental Safety Function	BWRX-300	BWRs	PWRs	CANDU
Control Reactivity	Two independent means of shut down	Two independent means of shut down.	Two independent means of shut down.	Two independent means of shut down.
Fuel Cooling	Passive natural circulation	Active forced circulation	Active forced circulation	Active forced circulation
Contain Reactivity	Dry passive cooling	Wet active cooling	Active cooling	Dry reactor building Wet vacuum building Active reactor building cooling

Legend:

Canada Deuterium Uranium (CANDU)

Pressurized Water Reactor (PWR)

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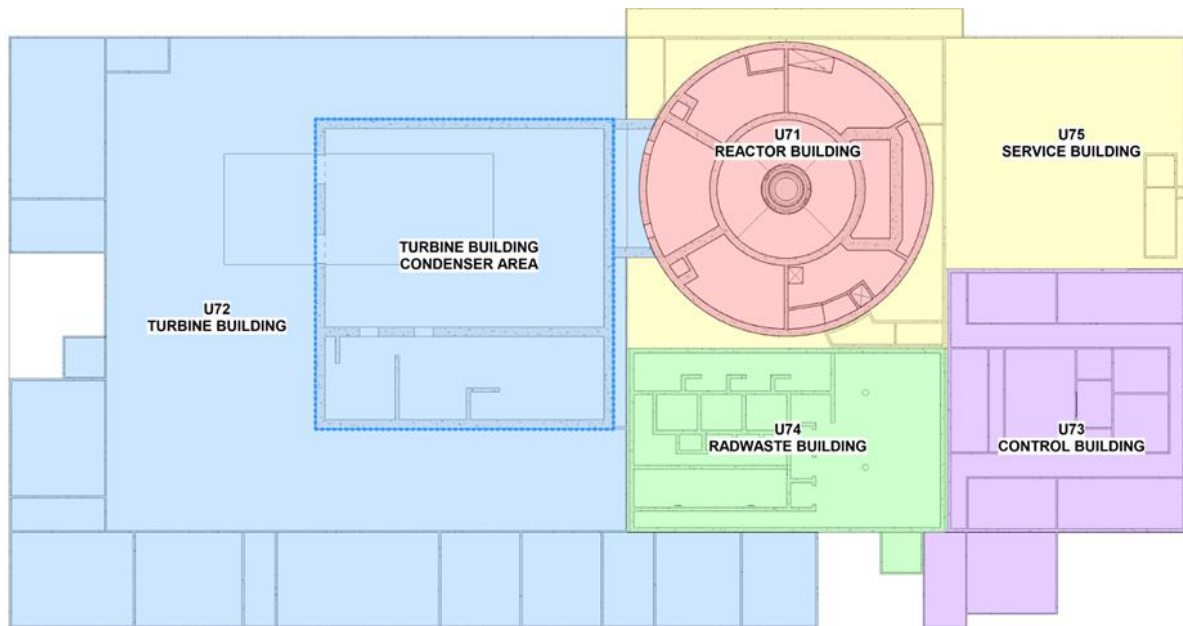
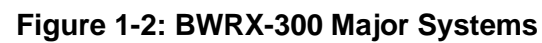


Figure 1-1: Representation of BWRX-300 Standard Power Block



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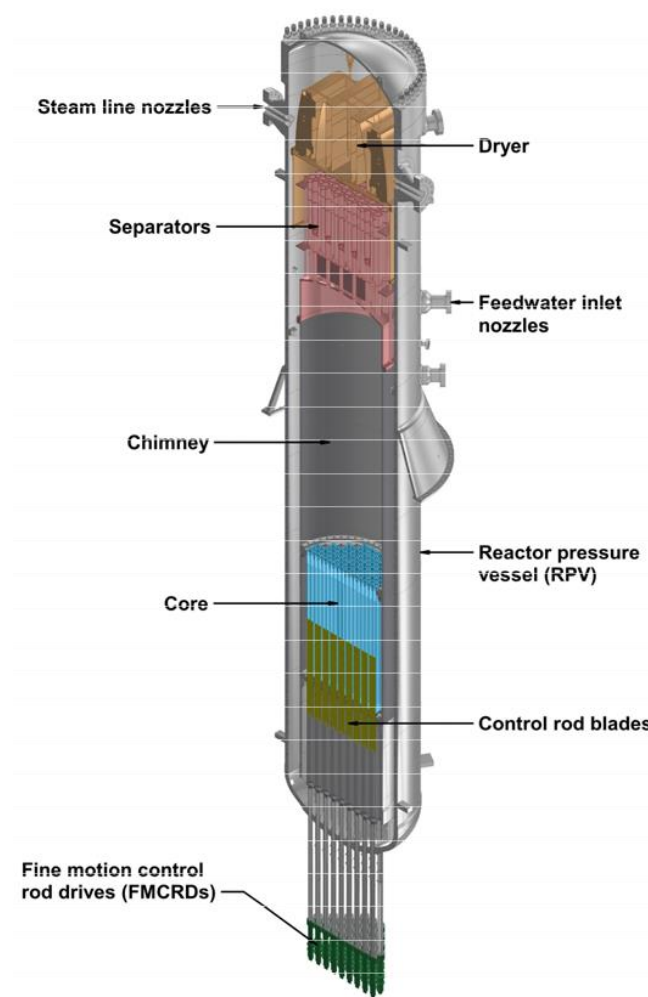


Figure 1-3: BWRX-300 RPV and Internals

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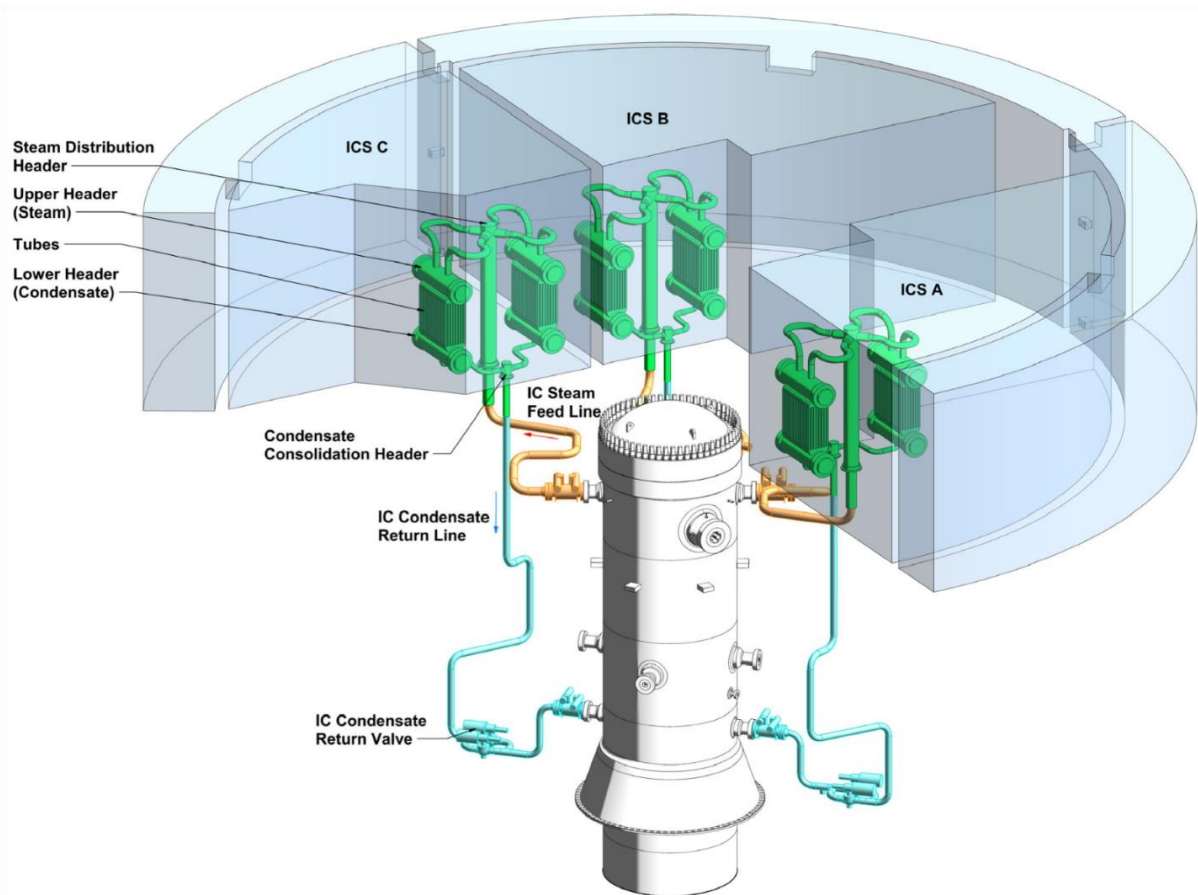


Figure 1-4: Isolation Condenser System

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APPENDIX A CLAIMS, ARGUMENTS AND EVIDENCE

The ONR “Safety Assessment Principles for Nuclear Facilities,” (Reference 1-41) identify ONR’s expectation that a safety case should clearly set out the trail from safety claims, through arguments to evidence. The Claims, Arguments, and Evidence (CAE) approach can be explained as follows:

1. Claims (assertions) are statements that indicate why a facility is safe
2. Arguments (reasoning) explain the approaches to satisfying the claims
3. Evidence (facts) supports and forms the basis (justification) of the arguments
4. The GDA CAE structure is defined within the Safety Case Development Strategy (SCDS), NEDC-34140P, “BWRX-300 UK GDA Safety Case Development Strategy,” (Reference 1-42) and is a logical breakdown of an overall claim that:

“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”.

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

This chapter does not directly demonstrate compliance against the Level 3 sub-claims that are identified within the SCDS (Reference 1-42).

It is not the intention to generate a comprehensive suite of evidence to support the derived arguments, as this is beyond the scope of GDA Step 2. However, where evidence sources are available, examples are provided.

Risk Reduction As Low As Reasonably Practicable

It is important to note that nuclear safety risks cannot be demonstrated to have been reduced to ALARP within the scope of a 2-Step GDA. It is considered that the most that can be realistically achieved is to provide a reasoned justification that the BWRX-300 SMR design aspects will effectively contribute to the development of a future ALARP statement. In this respect, this chapter contributes to the overall future ALARP case by demonstrating that:

- The chapter-specific arguments derived may be supported by existing and future planned evidence sources covering the following topics:
 - Relevant Good Practice has demonstrably been followed,
 - Operational Experience has been taken into account within the design process,
 - All reasonably practicable options to reduce risk have been incorporated within the design.
- Applicable Level 3 sub-claims are supported, as defined within the SCDS (Reference 1-42).

Probabilistic safety aspects of the ALARP argument are addressed within PSR Chapter 15.