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BWRX-300 UK Generic Design Assessment (GDA)
Chapter 13 – Conduct of Operations

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EXECUTIVE SUMMARY

The BWRX-300 Generic Design Assessment (GDA) Preliminary Safety Report Chapter 13 presents at a high-level how the BWRX-300 design and operational documentation can enable a future duty holder/licensee to implement the safety case in organizational structure/arrangements, training, implementation of the operational safety program, plant procedures and guidelines, and nuclear safety and nuclear security interfaces.

Detail of the duty holder/licensee arrangements are currently unknown and are not specific. Therefore, at this stage the scope of this chapter is limited to a summary of the operational philosophies developed for the BWRX-300 design. For example, the conduct of operations and the approach to defining a minimum staffing level will be described, however, staffing numbers will not be defined and are outside the scope.

Claims and arguments relevant to GDA Step 2 objectives and scope are summarized in Appendix A, along with an As Low As Reasonably Practicable position. Appendix B provides a Forward Action Plan.

ACRONYMS AND ABBREVIATIONS

Acronym	Explanation
ALARP	As Low As Reasonably Practicable
AM	Ageing Management
AOO	Anticipated Operational Occurrences
BDBA	Beyond Design Basis Accidents
BWR	Boiling Water Reactor
CAE	Claims, Argument, Evidence
DBA	Design Basis Accidents
DBT	Design Basis Threat
DEC	Design Extension Condition
EOP	Emergency Operating Procedures
EME	Emergency Mitigating Equipment
GDA	Generic Design Assessment
GEH	GE Hitachi Nuclear Energy
GNF	Global Nuclear Fuel
HFE	Human Factors Engineering
IAEA	International Atomic Energy Agency
I&C	Instrumentation and Control
IE	Initiating Event
INPO	Institute of Nuclear Power Operations
MSQA	Management for Safety and Quality Arrangements
MCR	Main Control Room
NISR	Nuclear Industries Security Regulations
NM	Nuclear Material
OLC	Operational Limits and Conditions
OPEX	Operational Experience
ORM	Other Radioactive Material
PSA	Probabilistic Safety Assessment
PSR	Preliminary Safety Report
QA	Quality Assurance
SA	Severe Accident
SAMG	Severe Accident Management Guidelines
SCDS	Safety Case Development Strategy
SMR	Small Modular Reactor
SNI	Sensitive Nuclear Information
SSCs	Structures, Systems, and Components

Acronym	Explanation
TS	Technical Specifications
UK	United Kingdom
U.S.	United States

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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

13 CONDUCT OF OPERATIONS

Introduction

This chapter presents at a high-level how the BWRX-300 design and operational documentation produced for the Generic Design Assessment (GDA) Preliminary Safety Report (PSR) can enable a future duty holder/licensee to implement the safety case in organizational structure/arrangements, training, implementation of the operational safety program, plant procedures and guidelines, and nuclear safety and nuclear security interfaces.

Detail of the duty holder/licensee arrangements are currently unknown and are not specific. Therefore, at this stage the scope of this chapter is limited to a summary of the operational philosophies developed for the BWRX-300 design. For example, the conduct of operations and the approach to defining a minimum staffing level will be described, however, staffing numbers will not be defined and are outside the scope. This chapter describes the following:

- Organizational structure of future licensee
- Training
- Implementation of the operational safety program
- Plant procedures and guidelines
- Nuclear Safety and Nuclear Security interfaces

Interfaces with other chapters

The following chapters support Chapter 13 - Conduct of Operations:

- Chapter 3 – NEDO-34165, “BWRX-300 UK GDA Ch. 3: Safety Objectives and Design Rules for Structures, Systems and Components,” (SSCs) (Reference 13-1) - describes the approach to delivering the safety objectives and design rules. The safety objectives and design rules provide important input to the design provisions which the conduct of operations are required to compliment.
- Chapter 4 – NEDC-34166P, “BWRX-300 UK GDA Ch. 4: Reactor,” (Reference 13-2) - provides a high-level description of the design of the GNF2 fuel assembly, the core loading pattern, and the associated Reactor Core System for the BWRX-300. The conduct of operations for core management are informed by fuel handling and reactor core details described in Chapter 4.
- Chapter 7 – NEDO-34169, “BWRX-300 UK GDA Ch. 7: Instrumentation and Control,” (I&C) (Reference 13-3) - describes the I&C systems required to support the plant safety strategy described in PSR Chapter 3 (Reference 13-1). This, in turn, provides an input to the conduct of operations.
- Chapter 14 – NEDO-34177, “BWRX-300 UK GDA Ch. 14: Construction and Commissioning,” (Reference 13-4) - provides an assessment and specification of the BWRX-300 plant construction and commissioning, including, but not limited to, civil works, mechanical systems, electrical systems, I&C, ancillary and auxiliary systems and environmental and habitability systems. The configuration control and management of Operational Experience (OPEX) in the design is discussed and detail of the conduct of operations provided.
- Chapter 15 – NEDO-34178, “BWRX-300 UK GDA Ch. 15: Safety Analysis,” (Reference 13-5) - provides the overarching safety analysis including Probabilistic Safety Assessment (PSA), Design Basis Accidents (DBAs), and Beyond Design Basis Accidents (BDBAs), including Design Extension Conditions (DECs) and Severe Accidents.

- Chapter 16 – NEDO-34188, "BWRX-300 UK GDA Ch. 16: Operational Limits and Conditions," (OLCs) (Reference 13-6) - describes the approach to developing OLCs for safe operation. Safety limits will be identified, however, the content of individual Technical Specifications (TSs) is outside the scope of the PSR. The integration of the approach to operating limits and conditions into the conduct of operations is described in this chapter.
- Chapter 17 – NEDO-34189, "BWRX-300 UK GDA Ch. 17: Management for Safety and Quality Assurance," (MSQA) (Reference 13-7) - describes at a high-level the management of safety and Quality Assurance (QA) arrangements used in the development of the Environment, Safety, Security and Safeguards submission for the BWRX-300. The conduct of operations uses the document and records management process outlined in the MSQA chapter.
- Chapter 18 – NEDO-34190, "BWRX-300 UK GDA Ch. 18: Human Factors Engineering," (HFE) (Reference 13-8) - describes the concept of operation for the BWRX-300 including the level of automation and role of humans in the various operating modes, the Main Control Room (MCR) staffing concept and the procedure concept. Integration of these HFE factors into the conduct of operation is described in this chapter.
- Chapter 25 – NEDO-34197, "BWRX-300 UK GDA Ch. 25: Security," (Reference 13-9) - describes the general approach to security as well as physical and cyber security. Chapter 13 acknowledges that the conduct of operation philosophy does not conflict with, and integrates with, the security requirements at this stage of design development.
- Chapter 28 – NEDO-34200, "BWRX-300 UK GDA Ch. 28: Safeguards," (Reference 13-10) - demonstrates understanding of safeguards requirements at the generic level and how they are accommodated in the standard plant design. Chapter 13 acknowledges that the conduct of operation philosophy is integrated with the safeguards requirements.

Claims and arguments relevant to GDA Step 2 objectives and scope are summarized in Appendix A, along with an As Low As Reasonably Practicable (ALARP) position. Appendix B provides a Forward Action Plan.

13.1 Organizational Structure of Future Licensee

The prime responsibility for safety is assigned to the future licensee. This responsibility includes all activities related to operation, directly and indirectly, and the supervision of activities of all other related groups, such as design, supply, manufacture and construction, employers, and contractors, as well as the future licensee itself. This responsibility is discharged in accordance with the future licensee management system.

13.1.1 Organizational Structure

The BWRX-300 is a simpler and safer design, compared with traditional Boiling Water Reactors (BWRs), requiring a smaller organization while still meeting the requirements of International Atomic Energy Agency (IAEA) Specific Safety Guide No. SSG-72, "The Operating Organization for Nuclear Power Plants," (Reference 13-11). The organizational structure framework is expected to be defined by the future licensee with all details including roles and responsibilities.

Upper tier management staffing levels are expected to be similar to those at existing Light Water Reactor facilities. For a single BWRX-300 unit, it is expected that some roles are combined from traditional operating models and could change if additional units are built on the same site.

Staffing levels required to operate the BWRX-300 are expected to be defined based on the safety analysis (with consideration and integration of HFE, NEDO-34190 (Reference 13-8)), plant maintenance and outage programs as they are defined. Staff performing operations and maintenance are expected to be qualified as determined using training needs analysis, developed using assumptions and findings of HFE analyses, GEH Concept of Operations and overall operational and maintenance philosophy. Nuclear Baseline requirements will also be defined following a detailed assessment by the future operator. The minimum staffing level complement is expected to be determined in accordance with the future licensee requirements.

The Plant Manager is expected to be accountable to the future licensee management, the national regulator, and the public to ensure the facility is operated and maintained with due diligence and in a manner consistent with the future licensee requirements.

The Operations and Maintenance Manager(s) ensure all aspects of the managed systems for operations and maintenance are implemented. For a single BWRX-300 unit, it is proposed that an experienced nuclear operator shall be selected as the Operations Manager. The Maintenance Manager may not be required to be an experienced nuclear operator.

The Shift Manager is accountable for ensuring that the facility is operated within its operating licence.

The site organization may be augmented with support from the future fleet organization, which includes the engineering function. The framework for changing and developing the fleet organizational structure is expected to be defined as the design is progressed with all programs put in place. Areas in the submission may include:

- Components and Equipment Surveillance
- Major Components
- Equipment Reliability
- Reactor Safety Program
- Ageing Manangement (AM)
- Risk and Reliability

- Chemistry
- Welding
- Environmental Qualification
- Pressure Boundary

13.1.2 Qualifications of Plant Personnel

The Plant Manager and Operations Manager positions assigned to the BWRX-300 shall be filled by staff who are experienced nuclear operators. This could include Small Modular Reactor (SMR), BWR or Pressurized Water Reactor experience.

Role Profiles and qualifications are expected to be developed via a systematic approach to training, and based on the nuclear safety significance of the role.

13.2 Training

13.2.1 General Approach

The BWRX-300 training program should be developed using a systematic approach to the training process that ensures worker competence and qualification to fulfil roles and responsibilities.

The training system should be developed and implemented to adhere to two fundamental principles:

- Performance based training is focused on the essential knowledge, skills, and safety attributes required to meet the job requirements (derived from HFE task analysis) and nuclear safety specific needs throughout the lifecycle of the facility.
- Systematically developed training is defined, produced, and maintained through an iterative and interactive series of steps, leading from the identification and satisfaction of a training requirement.

Training requirements are applied in a manner commensurate with risk. Training related processes and procedures may vary based on the safety significance and complexity of the work being performed. The training systems/programs and requirements include:

- Identification of the performance requirements of a specific job or duty area by conducting a job task analysis
- General worker training, initial job training, and continuing training based on a task analysis of the knowledge and skills required to perform each task and any attributes related to safety
- Training that is designed, developed, and implemented to meet qualification requirements
- Trainers that meet and maintain documented qualification requirements
- Formal evaluations and records used to confirm and document that workers are qualified to perform their duties
- A training change management process that systematically analyses procedural, equipment, and job description changes (including OPEX feedback) that may require changes to tasks and lead to training modifications
- Establishment of future/continued training requirements based on job and task analysis, and training needs analyses processes
- Periodic training program evaluations, with results incorporated into the training improvement process
- Creation and maintenance of worker training and qualification records
- Assurance that workers receive the level of training related to nuclear safety significance that corresponds to their employment and position duties; including, but not limited to, radiation safety, conventional safety, fire safety, and on-site emergency arrangements

In addition, training programs are established for initial personnel certification and maintenance of personnel training is ensured throughout. Initial and continuing certification training programs will be implemented in accordance with the principles of a systematic approach to training. Positions requiring regulatory certification are expected to be defined based on the technology needs and safety significance. Certification programs are expected to be developed as part of the design process.

13.2.2 Training Managed System Plan

All training of personnel is expected to be designed, developed, and delivered using a systematic approach to training.

A full-scope simulator, a replica of the MCR panels, is expected to be utilized to train and qualify control room Operations staff. This approach allows the operators to interface with the simulated plant system in the MCR environment.

13.2.2.1 Minimum Staff Complement

A minimum staff complement program is established to ensure sufficient numbers of qualified workers are present to meet facility licence requirements during all credible events in the BWRX-300 safety analysis. Minimum staffing numbers are expected to be determined following completion of detailed design and safety analysis. The analysis to determine the minimum staff complement considers:

- Actions required in the facility and their timing for the full range of the most resource intensive condition
- Initiating Events (IEs) and credible failures which require operator action that are considered in the Safety Analysis Report and the PSA (with HFE considerations)
- Operating strategies that define how the nuclear facility personnel respond to Anticipated Operational Occurrences (AOOs), DBAs, and emergencies
- Required interactions among facility personnel for the purpose of diagnosing, planning, communicating, coordinating, and controlling AOOs, DBAs, and emergencies
- Staffing demands required for the possible concurrent use of procedures related to AOOs, DBAs, and emergencies
- Staffing demands required to monitor indicators, displays, and alarms and to promptly and effectively operate the facility's equipment controls using procedures related to AOOs, DBAs, and emergencies
- Staffing demands required to perform tasks in field locations using procedures related to the events considered within the scope of the analysis
- Staffing demands required for the successful completion of any important human actions using procedures related to the events considered within the scope of the analysis
- Restrictions on the location of workers within the nuclear facility

The minimum staff complement requirements are validated to provide assurance that there are sufficient numbers of qualified workers available to operate the facility safely and respond to the most resource intensive conditions at all times.

The minimum staffing requirements are expected to be formalized in a procedure that describes:

- The specific number of staff to be present on-site, in the facility, and in the MCR, and the composition of the minimum staff complement with reference to specific positions or qualifications
- Modifications to minimum staff complement for different operational states and the specific number and composition of the minimum staff complement with reference to specific positions or qualifications for each operational state
- Any specific restrictions on the location of individuals in the facility

- Measures in place to monitor compliance with the minimum staff complement and to prevent any non-compliance
- Specific actions to be taken to reduce the risk to the facility in the event of non-compliance with the minimum staff complement

13.3 Implementation of the Operational Safety Program

13.3.1 Conduct of Operations

This section describes important operational issues relevant to safety throughout the lifetime of the plant and how the future licensee addresses identified issues adequately.

The future nuclear management system should set the standards for health, safety, environment, security, and quality during facility design, construction, commissioning, and operation based on a robust safety culture driven by the future licensee.

The future nuclear management system should supports a positive safety culture, in line with Institute of Nuclear Power Operations (INPO) "Principles for a Strong Nuclear Safety Culture," (Reference 13-12). This is achieved by committing workers to adhere to the nuclear management system, implementing practices that contribute to the excellence in worker performance, supporting workers in carrying out their tasks safely and successfully, and monitoring to improve the culture. The organizational structure implements the programs that make up the nuclear management system with the chief nuclear officer accountable for implementation and effectiveness of the nuclear management system. The outline of the programs and standards utilized for operating the plant is expected to be developed, with all program details and standards.

The nuclear management system should be based on a set of principles implemented in a graded approach.

13.3.2 Safety Culture

The safety culture should be established, promoted, communicated, and fostered by Senior Management through the nuclear management system. The safety culture is applicable to all activities that affect the health and safety of workers, the public, and the environment in every phase of the facility life cycle.

The safety culture is implemented, monitored, and periodically assessed through policies, programs, processes, and procedures that implement the varied administrative, maintenance, and operational aspects of facility operation. An established program summarizes internal and external processes used for oversight and assessment, tracks assessment action items, and monitors various metrics that may reveal safety culture issues (e.g., OPEX, performance trends, condition reports, regulatory inspections). The Human Performance and Performance Improvement programs also implement any expectations for understanding and promoting a strong safety culture.

13.3.3 Security Culture

The basis of a concept of security operations is set out within 006N6248, "BWRX-300 Security Assessment," (Reference 13-13) which informs the standard design.

The information within the Security Assessment document and supporting annexes represents a component part of the security informed design. This supports defined security outcomes regarding response and mitigative effects based on the Design Basis Threat (DBT) and protection of the Nuclear Material (NM) and Other Radioactive Material (ORM) inventory, the safety SSCs that maintain it in a safe state, and security SSCs that protect these.

13.3.4 Fitness for Service

The fitness for service safety and control area covers activities that affect the physical condition of SSCs to ensure that they remain adequate and able to perform their intended safety functions when required. Fitness for service is addressed in established programs that include Reliability, Maintenance, AM, Chemistry Control, Periodic Inspections, and In-Service Inspections.

Programmatic requirements addressing fitness for service span the full life cycle of the facility, beginning with inclusion in facility design, decision-making and consideration during each phase (e.g., design, construction, commissioning, operation) of the facility's life. Requirements evolve as the facility ages and specific process requirements may vary based on the life cycle phase (e.g., construction versus operation).

Reliability is incorporated during facility design, and through the Reliability Program. The Reliability Program is implemented to ensure that systems function reliably in accordance with design and performance criteria. Although the Reliability Program focuses primarily on the facility operational phase, it applies to all phases of the facility life cycle. The Reliability Program includes:

- Identification and categorization of systems using a systematic process
- Identification of specific failure modes and specification of reliability targets
- Specification of minimum capability and performance level consistent with safety targets and regulatory requirements
- Provisions for incorporating information into maintenance programs
- Provisions for inspection, tests, modeling, and monitoring to assess reliability based on safety classification
- Documentation of program activities, attributes, elements, results, and administration

The facility Maintenance Program establishes a maintenance strategy based on the plant design and safety analysis, to ensure that SSCs function as designed. The facility Maintenance Program is implemented by the future licensee. A systematic approach should be used to identify the SSC maintenance activities to be performed, and the associated maintenance intervals.

The Maintenance Program will describe the processes for planning, monitoring, scheduling, and executing maintenance work activities, including those maintenance activities performed during the construction and commissioning phases. Surveillances conducted as part of the Maintenance Program, including acceptance criteria, are addressed in PSR Chapter 16.4 (Reference 13-6).

An AM Program is established to ensure the reliability and availability of the required SSC safety functions throughout the facility service life. The effects of ageing and wear are taken into consideration during the design of Safety Classified SSC. The considerations include:

- Design margin assessment that considers the known ageing and wear mechanisms causing potential degradation in operational states, including the effects of testing and maintenance
- Provisions for monitoring, testing, sampling, and inspecting SSCs to assess ageing mechanisms and identify degradation that may occur during operation as a result of ageing and wear
- Online monitoring to provide forewarning of degradation leading to failure where failure could be safety-significant

Details regarding AM design provisions are provided in PSR Chapter 3.1.12 (Reference 13-1).

Chemistry control policies and goals are established to:

- Preserve the integrity of SSCs
- Minimize the effects of chemical impurities and corrosion on SSCs

- Manage radioactive material buildup in accordance with ALARP
- Limit release of chemicals and radioactive material to the environment

Chemistry control governs the development and maintenance of chemistry procedures, specifications, and methods of control. Knowledgeable and trained staff are assigned to monitor for abnormal trends so that action can be taken to ensure operations within specified limits. Performance indicators are maintained to satisfy reporting requirements.

Periodic and in-service inspection and testing programs are established to confirm that service-induced degradation has not increased the likelihood of a failure of a barrier against the release of radioactive material. Periodic in-service inspection and testing are established for:

- Nuclear pressure boundary components
- Containment components
- Containment structures
- Safety-related structures
- Balance of Plant pressure boundary components that are Safety Classified or subject to inspection based on AM requirements

13.3.5 Nuclear Material Packaging and Transport

Processes and procedures are expected to be established that address the safe packaging, registration, and transport of nuclear substances to and from the facility in a radioactive material transportation program. The program ensures shipping packages are designed and maintained to ensure protection and containment of the quantities of nuclear material transported. In addition, package certification, testing, inspection, and maintenance aspects are addressed within the program. This program is expected to be established prior to any fuel delivery to the future site.

13.3.6 Maintenance, Surveillance, Inspection and Testing

This section provides a description of arrangements that the future licensee will have in place to identify, control, plan, execute, audit, and review maintenance, inspection, and testing practices that influence reliability of SSCs and affect nuclear safety.

SSCs credited in the safety analysis are identified and periodically tested (surveillance) at a frequency related to the results of reliability analysis and OPEX to ensure that they will function as required. SSC performance that is inconsistent with assumptions in the safety analysis is identified. Appropriate remedial action will be taken, which may include modification. Following potential modification to an SSC, the test requirements are re-evaluated. Defense Line 3 (Safety Class 1) SSCs credited in the deterministic safety analysis will, in the future, be addressed in the OLC of PSR Chapter 16 (Reference 13-6) and the Defense Line 2 and 4 SSCs credited in the deterministic safety analysis are addressed in a future, to be defined program, required by Chapter 16. Furthermore, additional SSCs credited in the probabilistic safety analysis will be addressed in the OLC if their failure is a significant contributor to Core Damage Frequency. This Defence-in-Depth approach provides reasonable assurance that the consequences of postulated IEs are bounded by PSR Chapter 15 (Reference 13-5) results and safety goals are met.

SSCs are maintained in accordance with a maintenance strategy that defines the frequency and type of maintenance to be performed, taking into consideration the supplier recommendations, safety analysis, periodic inspection requirements, OPEX, and service conditions. Maintenance activities will be performed in accordance with approved procedures and practices. Preventive measures are employed to eliminate structural, system, and component damage or the contamination of systems with foreign material. In addition, predictive maintenance is performed based on plant monitoring system information. Maintenance includes the repair or replacement of malfunctioning SSCs as needed to re-establish conformance with requirements.

A maintenance program is to be implemented by the future licensee to address:

- Measures, policies, methods, and procedures that ensure SSC safety functional capability is maintained in accordance with design documents and safety analysis
- Processes for planning, monitoring, scheduling, and executing work activities so SSC continue to meet design intent and remain fit for service in the presence of degrading mechanisms
- Preventive and corrective maintenance activities, record retention requirements, calibration of measuring and monitoring devices, SSC monitoring (activity optimization), outage management, work planning and scheduling, work execution, maintenance procedures, post-maintenance verification and testing, and Maintenance Program assessment
- Predictive maintenance based on plant monitoring system information
- Surveillance covering OLC, with surveillance frequencies based on a reliability analysis, a PSA, and previous OPEX. The surveillance program will show viability of inspection techniques to meet performance requirements whilst taking the ALARP principle into account
- The approach taken to develop SSC surveillance program acceptance criteria
- Assurance that the surveillance program is adequate to ensure the inclusion of all relevant aspects of the OLC
- The timeline for the development of each program with milestones for development and implementation of each program and the processes followed
- Reviews of the results of each activity against acceptance criteria, with periodic reviews to ensure the program continues to meet objectives

Multiple aspects of the surveillance, inspection, and testing program will be addressed within the OLC, to include:

- Safety Classified plant SSCs that require monitoring to ensure they remain fit for purpose and operate reliably, and within safety limits as determined by safety analysis throughout the station operating life
- How surveillance, maintenance and repair ensure OLC parameters remain within acceptable limits and systems/components are operable
- Frequencies of surveillance based on reliability analyses, including, where available, a PSA and a study of experience gained from previous surveillance results (in the absence of both, the surveillance is based on supplier recommendations)
- A system for ensuring testing is performed and confirmed within the timelines allowed

13.3.7 Core Management and Fuel Handling

The future programs and procedures that govern the operational activities associated with BWRX-300 core management regarding fuel reliability should be based on guidelines established by GEH, utilizing decades of experience with fuel from Global Nuclear Fuel (GNF). Fuel-related design aspects, including operational and transient limits, are discussed in PSR

Chapter 4 (Reference 13-2). The core/fuel management guidelines are implemented through operational methods designed to mitigate and reduce duty-related fuel performance risks.

In general, the BWRX-300 operational methods employ an approach of limiting the duration of low power periods and limiting the rate at which power is raised following prolonged operation at low power. When raising power, a combined approach is used, consisting of an unrestricted power increase to an established threshold or prior conditioned power envelope, followed by raising power to a final value at a defined, controlled, slow ramp rate.

The operational practices are based on BWR OPEX:

- An established exposure-dependent Linear Heat Generation Rate threshold, below which no power maneuvering restrictions are applied, with power increases above the threshold limited to a defined controlled rate
- Power envelopes (conditioned power) established by the maintenance of specific power conditions sustained for a defined period
- Defined power ramp rates for power increases above the more limiting of a Linear Heat Generation Rate threshold or the conditioning envelope value, performed at a defined, controlled ramp rate
- Threshold power levels established for fuel bundles or nodes with unusually long periods of low power operation (long control intervals), implemented on a case-by-case basis using industry best practices
- Control rod exercising requirements
- Barrier fuel risk mitigation
- Established threshold values for fuel with high residence time in central portions of the core

Core Monitoring is a function of the plant computer system that provides three-dimensional core power monitoring to ensure the plant operates within the power distribution design basis. Core Monitoring provides confidence that the plant is operating in conformance with specified acceptable fuel design limits. Core Monitoring obtains information from the Diverse Protection System (refer to PSR Chapter 7 (Reference 13-3)), calculates thermal power limits, and provides estimates of power distributions. These estimates are calculated by the core simulator.

The Core Monitoring function acquires real-time reactor data from site plant data acquisition systems as necessary to define the reactor state for use by the core simulator. Core Monitoring can calculate the accumulated thermal and electrical energy produced by the plant from the beginning of an operating cycle. The Core Monitoring function is described further in PSR Chapter 4 (Reference 13-2).

13.3.8 Ageing Management and Long-Term Operation

AM processes and plans ensure the reliability and availability of required safety functions of SSCs throughout the service life of the facility (Lifecycle Management Plans). Periodic inspection or in-service inspection programs, as they relate to BWRX-300 aspects, are expected to be incorporated directly into AM Programs.

AM is addressed during the design process; the design provisions for AM are discussed in more detail in PSR Chapter 3 (Reference 13-1). Consideration is given to the feedback of OPEX, and a systematic approach is taken during the design phase to understand the effect of ageing of SSCs to evaluate design features for ageing prevention, monitoring, and mitigation. Mechanical, thermal, chemical, electrical, physical, biological, and radiation aspects are taken into consideration. SSCs determined to have shorter service lives than the nominal design life are identified, with AM strategies provided in the design documentation. The components that are identified with service lives less than the nominal design life also have replacement plans defined in the plant Maintenance Program, with associated monitoring requirements and provisions to permit their removal and replacement.

Ageing effects under design basis conditions, including transient and postulated IEs are also considered in equipment qualification programs.

SSC design information which takes into consideration the adverse effects of ageing establishes the baseline for the test data required to be collected and documented for AM Program monitoring and evaluation requirements.

Design documents also identify any special manufacturing or construction processes that are to be applied to prevent, mitigate, or eliminate known ageing mechanisms. These provisions are necessary for specification in procurement documents.

The AM Program and processes are used to detect, assess, and manage deterioration of SSCs as a result of ageing effects such as irradiation, corrosion, erosion, fatigue, and other material degradation.

The AM Program includes description of the following elements:

- Organizational arrangements
- Data collection and record keeping
- Screening and selection process for AM
- Evaluations for AM
- Condition assessments
- SSC specific AM plans
- Management of obsolescence
- Interfaces with other supporting programs
- Implementation of SSC specific AM plans
- Review and improvement processes for the AM Program

13.3.9 Control of Modifications

Future revision of the conduct of operations will address the identification method for designing, planning, executing, controlling, testing, auditing, reviewing, and documenting modifications to the plant from construction onwards, consistent with the guidance provided in IAEA SSG-61, “Format and Content of the Safety Analysis Report for Nuclear Power Plants,” (Reference 13-14). The control of modifications throughout the design development, including site specific Pre-construction Safety Report, is described with PSR Chapter 17 (Reference 13-7). The modification control process covers all safety-significant changes (permanent and temporary) made to SSCs, OLCs, plant procedures, and process software. The design and safety analyses are incorporated into the purchasing, construction, commissioning, operating, and maintenance documentation such that the as-built configuration of the facility is aligned with the design and safety analysis. Design authority configuration requirements, including the responsibilities and authority of organizations whose functions affect the configuration of the facility, including activities such as design, maintenance, construction, licensing, and procurement, are controlled through its Configuration Management System. A series of programs, including engineering change control, design management, and software, ensures plant configuration is controlled in a manner that is analysed to be safe. Control of modifications and configuration management during construction and commissioning phases is discussed further in PSR Chapter 14 (Reference 13-4).

The information provided in PSR Chapter 14 (Reference 13-4) includes descriptions of:

- The modification control process for maintaining the design basis, taking into account new information, OPEX, safety analyses, resolution of safety issues, or correction of deficiencies
- How design changes are assessed, addressed, and accurately reflected in the safety analyses or record prior to implementation

The plant modification control process covers:

- Changes made to plant systems and components, OLCs, plant procedures and process software, taking into account the safety significance of the proposed modifications to allow them to be graded
- Changes to task performance requirements (task step alterations, expected outcomes, procedure level), personnel job role responsibilities or the future licensee
- Records retention, and, where necessary, revision of documentation, procedures, instructions, and drawings to reflect the changes

13.3.10 Configuration Management

Configuration management, for construction onwards, is incorporated into operating and maintenance so that the as-built configuration of the facility aligns with the design and safety analysis. Configuration management is applied in a graded approach.

Configuration management during the construction and commissioning phases is also described further in PSR Chapter 14 (Reference 13-4).

Configuration management is not a stand-alone program. Configuration management plans are developed and integrated within the future nuclear management system (e.g., assessment, problem identification and resolution, training). From conception to the end of operations, configuration management ensures that data generated during design, construction, and commissioning reflects the design basis, and that specified requirements are kept current in the design and as-built documentation.

The design basis and requirements for the BWRX-300, including safety analysis, are established, documented and are traceable to the respective SSCs. Impact of design changes are assessed, addressed, and when applicable, reflected in the safety analyses. Subsequent changes to the physical and operational configuration are maintained in accordance with design requirements and configuration information throughout the operational life cycle. Where SSC requirements exceed functional design requirements (safety margin versus design margin), the process ensures that the safety margin is maintained for subsequent modifications. Physical assessments of SSC configuration are conducted as part of facility management.

Configuration information, the types and sources of configuration information, and associated documentation are controlled and maintained, with the status of changes identifiable. The facility design basis is maintained following turnover and commissioning to reflect new information, OPEX, safety analyses, and the resolution of safety issues or deficiency corrections.

Configuration deviations, when identified, are managed through the problem identification and resolution processes. Deviations are immediately controlled (if required), documented, evaluated for significance, and the underlying cause assessed. If deviations are deemed to be systemic, further action is taken to prevent future recurrence. Problem resolutions are reviewed for effectiveness.

Configuration management objectives and concepts are addressed in the respective training programs, with the necessary links between configuration management and the training programs established and maintained.

13.3.11 Engineering Change Control

Engineering change control is an integrated management process that ensures the physical and operational configuration and documentation continue to conform to the design and licensing basis requirements.

Facility configuration is maintained from initial fuel load to the end of operating life through established programmatic configuration and change control processes.

The change control process makes certain that safety limits, design basis, licensing basis, and normal operating margins are controlled under engineering change control, ensuring the facility is operated well within conditions analyzed to be safe.

13.3.12 Design Management

The Design Management Program will specify requirements for the following two areas:

- Management of prescribed activities appropriate for execution and control of required design, design support, and documentation for nuclear facilities and organizations i.e., maintaining design integrity
- Processes for creating or modifying documentation required for controlling design bases and design outputs

13.3.13 Program for the Feedback of Operating Experience

This section describes the program implemented for the feedback of OPEX. The OPEX Program ensures operational events and incidents occurring at the facility and other relevant facilities are captured or identified, recorded, notified, investigated internally, and used to incorporate lessons learned for the operation of the facility.

Relevant OPEX is considered for the BWRX-300 during design, construction, commissioning, operation, maintenance, and decommissioning. The requesting party, GEH, establishes provisions for the incorporation of OPEX through Integrated Management Systems. The OPEX comes from a variety of sources, including direct input, GEH/GNF experience from operating the BWR and Advanced Boiling Water Reactor fleet, INPO, Electric Power Research Institute, Boiling Water Reactor Owners' Group, U.S. Department of Energy, U.S. Nuclear Regulatory Commission, Canada Deuterium Uranium Owners Group, and Canadian Nuclear Safety Commission. OPEX associated with the construction and commissioning phases is discussed in PSR Chapter 14 (Reference 13-4).

Industry OPEX information is routinely made available to, or distributed by, GEH design and modifications personnel. An OPEX process will be established by the future licensee for evaluating, integrating, accessing, and sharing OPEX information. The OPEX process should address implementation of OPEX feedback during design activities and its continuance throughout the entire life cycle of the facility, to include how events are identified, recorded, investigated, and reported, as well as how findings from the events are used to enhance safety performance (i.e., a corrective action program).

13.3.14 Documents and Records

This section addresses the programmatic provisions for the management of records relevant to the operation of the facility over its lifetime.

Document and records program management is the responsibility of the future licensee.

Records management encompasses the control of documents and records with requirements addressed in the Controlled Document Management Program. The process for the control of documents should include the development, validation, and approval of safety-related documents. Documents should be available for use at the location where the work is to be performed. Changes to documents should be documented and tracked. The future process for the control of records ensures that records are readable, complete, identifiable, traceable, retrievable, preserved, and retained as necessary.

The program should ensure that controlled documents include:

- Unique identification
- Defined format and presentation
- Identification of status
- Review for adequacy and approval
- Availability for use at the location where the work is performed or where the document is required for reference
- Prompt removal of obsolete documents for use.

Records shall be:

- Readable
- Complete
- Identifiable
- Traceable to the related items and work
- Retrievable
- Preserved
- Retained as specified

Document management for the BWRX-300 is controlled under the QA Program during design. The QA Program requirements during design of the BWRX-300 are established in NEDO-11209-A, "GE Hitachi Nuclear Energy Quality Assurance Program Description" (Reference 13-15). NEDO-11209-A includes requirements during design, addressing:

- Procurement document control
- Instructions, procedures, and drawings
- Document control
- Control of QA records

Document and records management is discussed with respect to the QA Program in PSR Chapter 17 (Reference 13-7). Documents and records management during construction and commissioning is discussed in PSR Chapter 14 (Reference 13-4).

13.3.15 Reactor Outages

This section addresses the programmatic aspects of the conduct of periodic reactor shutdowns (outages).

The current reference cycle is based on a nominal 12-month fuel cycle. Different fuel cycle durations can be supported depending on the overall fuel reload strategy to be deployed on a cycle/multi-cycle specific basis.

Outage analysis does not address forced outages, but surveillance and maintenance activities that require the plant to be shut down are minimized to the extent possible, largely by enhanced system reliability achieved through design simplicity.

The Work Management Program provides for the implementation of processes and procedures for the planning, scheduling, and execution of maintenance activities. Work planning is conducted at both the overall plant and individual job levels. The Outage Management Program establishes the criteria followed to confirm that planned outage and emergent work is completed satisfactorily.

In addition to procedures for routine outage maintenance activities, a forced outage plan is required for emergent conditions.

Outage plans are reviewed for nuclear safety, with work groups reviewing the plans within their area of responsibility and with specific consideration given to:

- Impact on operating units and systems
- Application of controls during infrequently performed tests and evolution to ensure the plant is maintained within the design basis
- Contingency plans for alternate measures to maintain safe shutdown
- Routine review to capture changes from the original plan impact assessment
- Outage OPEX

The cumulative effect of plant equipment taken out of service is taken into consideration to ensure there are no adverse effects on the performance of safety functions when planning and scheduling outage work. In addition, plans to remove equipment from service during an outage include measures to deal with the possible consequences of an event occurring while the equipment is out of service. Clear statements are made to identify when equipment is being taken out of service, to include the duration and impact of removing the equipment from service. The Outage Management Program is expected to be produced by the future licensee.

13.4 Plant Procedures and Guidelines

This section addresses the relevant documents used by plant staff to ensure that procedures and guidelines for normal operation, AOOs, and accident conditions are followed in the intended manner. Procedure development is a technical element of the BWRX-300 HFE program. The procedure development process is described in PSR Chapter 18 (Reference 13-8).

13.4.1 Administrative Procedures

This section describes the administrative procedures that outline the essential elements of the administrative programs used by the future licensee to ensure the safe management of the plant. The processes to develop, approve, revise, and implement the procedures are described along with a list of the relevant procedures.

Administrative procedures contain adequate programmatic controls to provide an effective interface between organizational elements. This includes contractors or organizations providing support to the facility licensee.

Procedure maintenance and control of procedure updates should be performed in accordance with the future licensee's QA Program processes.

13.4.2 Operating Procedures

The facility is operated, monitored, and maintained within the safe operating envelope and in accordance with procedures that are consistent with the design. Operating procedures are established to provide for the safe conduct of BWRX-300 normal operations. Normal operation is specified in the OLCs (see PSR Chapter 16 (Reference 13-6)).

Normal, abnormal, unplanned, and Emergency Operating Procedures (EOPs) are validated to be accurate and usable without any human error traps and are verified to be consistent with the safe operating envelope.

Plant operations are performed in accordance with procedures, with use and adherence directions provided for the worker. Temporary procedures may be issued when existing permanent procedures are not applicable to the work being performed. Temporary procedures are periodically reviewed for applicability and cancelled when no longer required.

Operating procedures address:

- Normal operation
- Abnormal operation
- Emergency operation
- Refueling and outage planning
- Alarm response
- Maintenance, inspection, test, and surveillance
- Beyond design basis and severe accidents

13.4.3 Procedures and Guidelines for Operating the Plant During Accidents

13.4.3.1 Emergency Operating Procedures

This section describes the approach followed to develop the EOPs, and procedure development that supports the operator when responding to anticipated and unanticipated events. Further details on emergency response can be found in NEDO-34191, "BWRX-300 UK GDA Ch. 19: Emergency Preparedness and Response," (Reference 13-16).

Emergency procedures are available for non-routine and emergency conditions that require immediate action. Emergency conditions addressed include unexpected radiological and non-radiological hazards, excessive emission of radiological and non-radiological liquid or gaseous effluent, fires, loss of preferred power, and natural disasters. Emergency procedures are kept in prominent, easily accessible locations. Emergency procedures are exercised in practice drills to ensure that requirements are met.

EOPs implement the strategies and measures employed in the integrated accident management plan and prevent or reduce the likelihood of escalation of an accident with the aim that it is avoided, accident progression is terminated, and fission product releases are kept to a minimum. The EOPs contain a set of information, instructions, and actions designed to prevent or reduce the likelihood of escalation of an accident, mitigate its consequences, and bring the reactor to a safe and stable state.

All EOPs are developed in accordance with a systematic procedure development plan that considers HFE principles in both the actions required by the procedure and the design of the procedure itself.

13.4.3.2 Guidelines for Accident Management

This section describes the programmatic approach followed to develop accident management procedures and guidelines, including EOPs, Emergency Mitigating Equipment (EME) and Severe Accident Management Guidelines (SAMGs).

Accident management includes multiple components such as equipment and instrumentation, procedures and guidelines, and organizational accountabilities, and it interfaces with many programs established for a reactor facility. An adequate accident management plan ensures the ability to respond to any credible accident in order to prevent the escalation of the accident, mitigate the consequences of the accident, and achieve a long-term stable state after the accident. Integrated accident management planning consists of a cohesive set of plans and arrangements undertaken to ensure:

- Safety systems and the available SSCs can be used to control the reactivity, cool the fuel, and contain the radioactive materials, such that damage to the reactor vessel and harm to workers, public, and environment is prevented or mitigated
- Personnel with responsibilities for accident management are adequately prepared to utilize the available resources, procedures, and guidelines to perform effective accident management actions and, when deemed necessary, to call for and interact with the emergency response teams

EOPs, EME guidelines, and SAMGs are developed and implemented to facilitate a licensee’s capability to manage the AOOs, DBAs, and BDBAs, including DECs and severe accidents (see PSR Chapter 15 (Reference 13-5)).

The process of accident management planning will define and describe the following requirements:

- Specific goals of accident management
- Requirements of accident management
- Equipment and instrumentation
- Procedures and guidelines
- Organizational accountabilities

A timeline with milestones for the development, validation, and implementation of all operating procedures, EOPs, EME guidelines, and SAMGs for accident management is expected to be provided by the future licensee.

13.5 Nuclear Safety and Nuclear Security Interfaces

13.5.1 General Nuclear Safety and Security

The plans for physical protection of the facility are described in separate, restricted documents. This section addresses how safety measures and nuclear security measures are designed and integrated.

The future licensee is responsible for managing the implementation of safety requirements and security requirements, with the primary objective of minimizing risk, through programs and processes established to ensure close cooperation between safety managers and security managers. The safety and security measures are designed and implemented through programs and processes in a complementary manner that do not compromise each other. Mechanisms are established within the programs to resolve any potential conflicts and to manage the safety-security interfaces.

13.5.2 Security Interface

The basis of a concept of security operations is set out within the Security Assessment document (Reference 13-13) which informs the standard design.

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 concern response and mitigative effects based on the DBT and protection of the NM and ORM inventory, the safety SSCs that maintain the inventory in a safe state, and security SSCs that protect these.

PSR Chapter 25 (Reference 13-9) provides further information.

13.5.3 Safeguards Interface

PSR Chapter 28 (Reference 13-10), demonstrates that the BWRX-300 design and generic nuclear material safeguards arrangements can comply with the international treaty obligations, relevant domestic policy, legislation, regulations, and regulatory guidance.

The report considers:

- Safeguards design information
- Safeguards operational information
- Nuclear fuel life cycle
- Provision of access and assistance to Safeguards Inspectors
- Demonstration that a feasible program for nuclear material accountancy can be achieved
- Safeguards' interface with safety, security and waste management for an integrated Environment, Safety, Security and Safeguards case.

13.6 References

13-1 NEDO-34165, “BWRX-300 UK GDA Ch. 3: Safety Objectives and Design Rules for SSCs,” Rev B, GE-Hitachi Nuclear Energy Americas, LLC.

13-2 NEDC-34166P, “BWRX-300 UK GDA Ch. 4: Reactor,” Rev B, GE-Hitachi Nuclear Energy Americas, LLC.

13-3 NEDO-34169, “BWRX-300 UK GDA Ch. 7: Instrumentation and Control,” Rev B, GE-Hitachi Nuclear Energy Americas, LLC.

13-4 NEDO-34177, BWRX-300 UK GDA Ch. 14: Construction and Commissioning,” Rev B, GE-Hitachi Nuclear Energy Americas, LLC.

13-5 NEDO-34178, “BWRX-300 UK GDA Ch. 15: Safety Analysis,” Rev B, GE-Hitachi Nuclear Energy Americas, LLC.

13-6 NEDO-34188, “BWRX-300 UK GDA Ch. 16: Operational Limits and Conditions of Safe Operation,” Rev B, GE-Hitachi Nuclear Energy Americas, LLC.

13-7 NEDO-34189, “BWRX-300 UK GDA Ch. 17: Management for Safety and Quality Assurance,” Rev B, GE-Hitachi Nuclear Energy Americas, LLC.

13-8 NEDO-34190, “BWRX-300 UK GDA Ch. 18: Human Factor Engineering,” Rev B, GE-Hitachi Nuclear Energy Americas, LLC.

13-9 NEDO-34197 “BWRX-300 UK GDA Ch. 25: Security,” Rev B, GE-Hitachi Nuclear Energy Americas, LLC.

13-10 NEDO-34200 “BWRX-300 UK GDA Ch. 28: Safeguards,” Rev B, GE-Hitachi Nuclear Energy Americas, LLC.

13-11 IAEA Specific Safety Guide No. SSG-72, “The Operating Organization for Nuclear Power Plants,” IAEA, 2022.

13-12 Principle for a Strong Nuclear Safety Culture, INPO, 2004.

13-13 006N6248, “BWRX-300 Security Assessment,” GE-Hitachi Nuclear Energy Americas, LLC.

13-14 Format and Content of the Safety Analysis Report for Nuclear Power Plants, Specific Safety Guide, IAEA, SSG-61, 2021.

13-15 NEDO-11209-A, “GE Hitachi Nuclear Energy Quality Assurance Program Description,” GE-Hitachi Nuclear Energy Americas, LLC.

13-16 NEDO-34191, “BWRX-300 UK GDA Ch. 19: Emergency Preparedness and Response,” Rev B, GE-Hitachi Nuclear Energy Americas, LLC.

13-17 NEDO-34140, “BWRX-300 UK GDA Safety Case Development Strategy,” Rev B, GE-Hitachi Nuclear Energy Americas, LLC.

APPENDIX A CLAIMS, ARGUMENTS AND EVIDENCE

Claims, Argument, Evidence (CAE)

The 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

The GDA CAE structure is defined within NEDO-34140, "BWRX-300 UK GDA Safety Case Development Strategy," (SCDS) (Reference 13-17) 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 environmental, safety, security and safeguard 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.

The Level 3 sub-claims that this chapter demonstrates compliance against are identified within the SCDS (Reference 13-17) and are as follows:

2.2.1 Appropriate MSQA procedures controlling documentation production are in place.

2.2.4 Future arrangements can be developed to support an operational facility including normal and emergency arrangements.

In order to facilitate compliance, demonstration against the above Level 3 sub-claims, this PSR chapter has derived a suite of arguments that comprehensively explain how their applicable Level 3 sub-claims are met (see Table A-1 below).

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 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
  - OPEX has been taken into account within the design process
  - All reasonably practicable options to reduce risk have been incorporated within the design
- It supports its applicable Level 3 sub-claims, defined within the SCDS (Reference 13-17)

Probabilistic safety aspects of the ALARP argument are addressed within PSR Chapter 15 (Reference 13-5).

Table A-1: Conduct of Operations Claims and Arguments

Level 3 Chapter Claim	Chapter 13 Argument	Sections and/or reports that evidence the arguments
2.2 The BWRX-300 has been developed in accordance with approved procedures, with appropriate governance and assurance arrangements by a competent and clearly defined organization.		
2.2.1 Appropriate MSQA procedures controlling documentation production are in place.	Procedures controlling the creating, receiving, classifying, controlling, storing, retrieving, updating, revising, and deleting of documents, records, and reports relevant to the operation of the facility can be developed.	13.3.14 Documents and Records
2.2.4 Future arrangements can be developed to support an operational facility including normal and emergency arrangements.	Operating procedures can be developed to provide a guide for operator response suitable to normal, emergency, and severe accident conditions.	13.4 Plant Procedures and Guidelines
	Operating limits and conditions can be identified to ensure the plant is operated safely at all times.	3 Safety Objectives and Design Basis Rules for Structures, Systems and Components
	The maintenance, surveillance, inspection, and testing, and ageing and degradation procedures can be developed to ensure the requirement of operating limits and conditions is effective.	13.3.6 Maintenance, Surveillance, Inspection and Testing 13.3.8 Ageing Management and Long-Term Operation

APPENDIX B FORWARD ACTIONS

Table B-1: Conduct of Operations Forward Actions

FAP No.	Finding	Forward Actions	Delivery Phase
PSR13-27	Detail of the duty holder/licensee arrangements are currently unknown and are not specific. Therefore, at this stage the scope of this chapter is limited to a summary of the operational philosophies developed for the BWRX-300 design. For example, the conduct of operations and the approach to defining a minimum staffing level will be described, however, staffing numbers will not be defined and are outside the scope.	Detail to be added for conduct of operations once future duty holder/licensee arrangements are known.	Before Site License Application, Environmental Permit Applications and/or BL3 Design Phase