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BWRX-300 UK Generic Design Assessment (GDA) Chapter 21: Decommissioning and End of Life Aspects

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EXECUTIVE SUMMARY

This chapter of the Preliminary Safety Report provides an overview of the BWXR-300 Small Modular Reactor's (SMR's) approach to decommissioning and end-of-life. The content covers the information expected as part of Step 2 of a Generic Design Assessment (GDA).

It incorporates a description of the relevant operational experience informing the planning of decommissioning activities, approaches to decontamination and design for decommissioning. The key features of the BWRX-300 which enable safe decommissioning, in accordance with the principle of as-low-as-reasonably-possible and international guidelines and regulations, are described.

Given that decisions on decommissioning are expected to continuously evolve across plant licensing, construction, and operation, those areas to be addressed by a future licensee have been highlighted. A provisional approach, in the form of prompt decommissioning, has been described, alongside an anticipated timeline and overview of critical milestones. Claims and arguments relevant to GDA Step 2 have been outlined in the Appendix A. Any outstanding actions have been detailed in a Forward Action Plan (Appendix B). Appendix C gives the U.K. Regulatory Context, including an overview of the relevant U.K. legislation, standards and guidance applied to NPP decommissioning.

Based on a simplified SMR design and incorporation of International Atomic Energy Agency design for decommissioning principles, there are no anticipated showstoppers to the implementation of decommissioning a BWRX-300 site.

ACRONYMS AND ABBREVIATIONS

Acronym	Explanation
AGR	Advanced Gas-cooled Reactor
ALARP	As Low As Reasonably Practicable
BAT	Best Available Technique
BWR	Boiling Water Reactor
C&M	Care and Maintenance
DDP	Detailed Decommissioning Plan
FP	Fuel Pool
GDA	Generic Design Assessment
GDF	Geological Disposal Facility
GEH	GE Hitachi Nuclear Energy
GSR	General Safety Requirements
HAW	Higher Activity Waste
HLW	High Level Waste
IAEA	International Atomic Energy Agency
ILW	Intermediate Level Waste
LfE	Learning from Experience
LLW	Low-Level Waste
NPP	Nuclear Power Plant
NWS	Nuclear Waste Services
ONR	Office for Nuclear Regulation
OPEX	Operational Experience
PCS	Primary Containment System
PDP	Preliminary Decommissioning Plan
POCO	Post-operational Clean-out
PSLA	Plant Service Area
PSR	Preliminary Safety Report
RB	Reactor Building
RGP	Relevant Good Practice
RPV	Reactor Pressure Vessel
SAP	Safety Assessment Principle
SCCV	Steel-Plate Composite Containment Vessel
SF	Spent Fuel
SFSF	Spent Fuel Storage Facility
SMR	Small Modular Reactor
SSC	Structure, System and Component
TAG	Technical Assessment Guide

Acronym	Explanation
TB	Turbine Building
UK	United Kingdom
U.S.	United States
VLLW	Very Low-Level Waste

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

21 DECOMMISSIONING AND END OF LIFE ASPECTS

Introduction

The objective of this chapter of the Preliminary Safety Report (PSR) is to provide assurance that the design of the BWRX-300 has considered factors to ensure that the Nuclear Power Plant (NPP) can be decommissioned safely and that the risk of future decommissioning is minimised, so far as is reasonably practicable.

Scope

This chapter presents a set of principles for the facilitation of safe decommissioning, including how they have been derived from Relevant Good Practice (RGP) and Operational Experience (OPEX).

The selection and justification of a decommissioning strategy is outlined (assumed to be prompt, in line with International Atomic Energy Agency (IAEA) expectations, but not precluding deferred), alongside the anticipated timeline for key decommissioning activities.

Included is 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 challenge to ensure decommissioning risks are reduced As Low As Reasonably Practicable (ALARP). A high-level summary of why there is confidence that the BWRX-300 can be safely decommissioned is also presented, including reference to OPEX of successfully decommissioned Boiling Water Reactors (BWRs) worldwide and the availability of suitable techniques, including decontamination techniques, to support such decommissioning.

A high-level estimate of the nature and quantities of radioactive wastes anticipated to be generated during decommissioning and an assurance that these can be managed and disposed of as required is provided.

Document structure

Following on from the introduction, this PSR chapter is divided into the following sections:

- Section 21.1 – General Principles and Regulations: highlights the extent of OPEX available in the field of nuclear facility decommissioning, including BWRs, which can inform decommissioning plans.
- Section 21.2 – Decommissioning Strategy: discusses the decommissioning strategies available for selection by the future licensee, including an overview of the likely option which will be adopted for the BWRX-300 and the potential strategy timeline.
- Section 21.3 - Facilitating Decommissioning During Design and Operation: the design needs for effective and safe decommissioning are discussed, alongside how integration of these principles is, and will continue to be, facilitated by the BWRX-300 requirements management and design control processes.
- Section 21.4 – Decommissioning Plan: proposes a tentative plan for the decommissioning of the BWRX-300, indicating the stages involved and the key milestones anticipated.
- Section 21.5 – Provisions for Safety During Decommissioning: an overview of the employed design elements of BWRX-300 intended to decommissioning, provisions for safety during decommissioning, and the management strategy for waste encountered and generated during decommissioning, including waste category volume estimates and discussion of SF storage.
- Section 21.6 – Site Restoration and End-State: describes the proposed site end-state.

- Section 21.6 – Conclusions: a summary of the key content covered in this PSR chapter.
- Section 21.7 – References: a list of the supporting references used in this chapter.

Interfacing systems

The key interfacing systems with the decommissioning topic are:

- All engineered systems in the NPP must be decommissioned. The main interface is with those that will (or may) become radiologically contaminated or neutron activated.
- Facilities for the storage of both Higher Activity Waste (HAW) and Spent Fuel (SF), until such time as they can be transferred to the Geological Disposal Facility (GDF), will likely be onsite throughout decommissioning and be the final items to be decommissioned.

Interfaces with other chapters

The following chapters of the BWRX-300 PSR interface with this topic:

- Chapter 11: Management of Radioactive Waste NEDO-34174, "BWRX-300 UK GDA Chapter 11: Management of Radioactive Waste," (Reference 21-1)
- Chapter 12: Radiation Protection, NEDO-34175, "BWRX-300 UK GDA Chapter 12: Radiation Protection," (Reference 21-2)
- Chapter 26: Interim Storage of SF, NEDO-34198, "BWRX-300 UK GDA Chapter E1: Introduction," (Reference 21-3)

This document also interfaces with the Preliminary Environmental Report, particularly the following chapters:

- Chapter E1: Introduction, NEDO-34218, BWRX-300 UK GDA Chapter E1: "Introduction," (Reference 21-4)
- Chapter E4: Detailed Information about the Design, NEDO-34221, "BWRX-300 UK GDA Chapter E4: "Information About the Design," (Reference 21-5)
- Chapter E5: Radioactive Waste Management Arrangements, NEDO-34224, "BWRX-300 UK GDA Chapter E5: Radioactive Waste Management Disposals," (Reference 21-6)
- Chapter E7: Quantification of Radioactive Waste Disposals, NEDO-34224, "BWRX-300 UK GDA Chapter E7: Quantification of Radioactive Waste Disposal," (Reference 21-7

Further information of relevance to this PSR chapter is provided in the Appendices. Appendix A provides the Claims, Arguments and Evidence pertinent to this chapter, Appendix B describes the Forward Action Plan items associated with decommissioning, and Appendix C gives the U.K. Regulatory Context, including an overview of the relevant U.K. legislation, standards and guidance applied to NPP decommissioning.

21.1 General Principles and Regulations

International safety standards, regulations, and general principles, as relevant to NPP decommissioning, have been applied in the design of the BWRX-300, in addition to in the initial formulation of a decommissioning strategy and plan.

Beyond these regulations and principles, evolving best practice, informed by international OPEX, has also been considered, and, critically, will continue to be considered as the initial decommissioning plan is updated during the operational lifetime.

21.1.1 Regulatory Context

IAEA General Safety Requirements (GSR) Part 6: "Decommissioning of Facilities," (Reference 21-8) establishes a series of safety requirements for the decommissioning of nuclear facilities. These cover the following:

- The protection of people and the environment (Requirements 1-3)
- Responsibilities of associated with decommissioning (Requirements 4-6)
- Management of decommissioning (Requirement 7)
- Decommissioning strategy (Requirement 8)
- Financing of decommissioning (Requirement 9)
- Planning for decommissioning during the lifetime of the facility (Requirements 10-11)
- Conduct of decommissioning actions (Requirements 12-14)
- Completion of decommissioning actions and termination of the authorisation for decommissioning (Requirement 15)

These requirements cover actions to be taken by the relevant government, regulatory body, and site licensee, which are outside the scope of this safety case. However, the information presented in this chapter is intended to indicate how decommissioning of the BWRX-300 will comply with these regulatory requirements and additional international safety standards. The IAEA, "Format and Content of the Safety Analysis Report for Nuclear Power Plants, Specific Safety Guide" (SSG) 61, (Reference 21-9) has been used as a basis for the content included.

The IAEA safety standards (inclusive of safety fundamentals, safety requirements, and safety guides) are integrated into individual IAEA Member State national facility requirements.

21.1.2 Operational Experience of Decommissioning

Construction, and subsequent operation, of commercial NPPs began in the mid-twentieth century and has continued multi-nationally to the present day. Beyond development in the reactor designs, there has also been growing experience on how best to implement decommissioning of these facilities, as many older plants reach their operational limit or otherwise require shutdown. With each decommissioning event, lessons have been learnt to improve future reactor designs with decommissioning in mind, with the ultimate aim to improve safety, reduce risk ALARP and reduce timescales where possible.

Design lessons from decommissioning have been collated by various international bodies, notably the IAEA TECDOC-1657, "Design Lessons Drawn from the Decommissioning of Nuclear Facilities," (Reference 21-10), and review of on-going feedback will also inform future decommissioning decisions (including that of the BWRX-300 reactor).

Decommissioning operations have been undertaken, or are currently being undertaken, worldwide. Programmes cover a variety of reactor types and adopted decommissioning strategies, with varying end-stages and timelines. A list of fully decommissioned nuclear power reactors can be found in Appendix II of IAEAL 23-01572, “Global Status of Decommissioning of Nuclear Installations,” (Reference 21-11).

21.1.3 Boiling Water Reactors

As of 2024, over 50 BWR-type reactors have entered permanent shutdown internationally “World Statistics – Permanent Shutdown Reactors,” (Reference 21-12), with fully decommissioned BWR plants in Germany, the United States (U.S) and Japan (the JPDR demonstration reactor) (Reference 21-11).

The largest BWR nuclear reactor successfully decommissioned in the U.S. is the Big Rock Point BWR in Michigan U.S. where decommissioning commenced in 1997. Site remediation was completed in August 2006, “Power Reactor Sites – Big Rock Point,” (Reference 21-13). Smaller BWRs have been fully decommissioned in Germany; the HDR Grosswelzheim (25 MWe) achieved site clearance in 1998 and VAK Kahl (15 MWe) in 2010, “Nuclear Power Reactors in the World, Reference Data Series No. 2,” (Reference 21-14).

A subset of relevant ongoing international BWR decommissioning projects is outlined in Table 21-1.

21.1.4 General Principles of Decommissioning

There are a number of areas in which experience has informed technology and best-practice development on the approach and general principles applied to decommissioning. Knowledge and experience sharing is available via the IAEA’s “International Decommissioning Network,” (Reference 21-15) and alternative platforms. Selected topics are highlighted below.

21.1.5 Design for Decommissioning

Early reactor design and construction did not account for the need to decommission facilities. The role of this lack of consideration by complicating decommissioning, and the necessity to account for decommissioning in future reactor builds, is a consistent piece of feedback from decommissioning programmes.

This concept covers many design points, from materials choices to site layout. Some examples are listed here, with extensive detail found in IAEA-TECDOC-1657 (Reference 21-10):

- Materials selection: where possible within materials property constraints, components should be made from materials which undergo minimal activation in-service.
- Design for Post-operational Clean Out (POCO): consider ease of flushing, purging and drainage in piping system designs, design for the decontamination and dismantling of ventilation ducts, and minimise in-operation build-up of radioactive deposits.
- Waste minimisation, “Considerations for Waste Minimization at the Design Stage of Nuclear Facilities,” (Reference 21-16): incorporate Best Available Technique (BAT) to reduce amount of operational and decommissioning waste, with particular emphasis of contamination control, the reuse/recycling of materials, and reducing waste quantity through treatment (e.g., compaction, filtration, evaporation, characterisation and segregation).
- Site layout: plant design should consider ease of access to those components likely to be activated during operation (particularly any large pieces of equipment), alongside designing for segregation and shielding during decommissioning activites, and space for safe lifting and dismantling of components.
- Record keeping: documentation of design and construction history, details of operation (including any incidents), survey data, waste storage, plant modifications and

maintenance, and decommissioning procedures should be kept up-to-date, with long-term readability and maintainabilty in mind (Reference 21-16).

21.1.6 Dismantling of Large Components

Removal of large components is a notable issue encountered in ongoing, or completed, decommissioning programmes, whereby large, contaminated components in the plant are difficult to remove, and plant design does not accommodate for dismantling in situ.

Segmentation of large components has been successfully achieved in past decommissioning programmes. Significant complexity is added when components are of high activity, typical of the Reactor Pressure Vessel (RPV) and reactor internals. In cases where cutting is necessary with these components, OPEX has demonstrated the suitability of a variety of techniques for remote operation, including:

- Abrasive Water Jet Cutting (e.g., for Vermont Yankee RPV internals, (Reference 21-17)
- ‘Ice Sawing’ (freezing water to create needed stiffness for in-situ cutting e.g., for secondary steam generators in Gundremmingen, “Experience with the Dismantling of Three Secondary Steam Generators in Unit A in Gundremmingen by the ‘Ice Sawing’Technicque,” (Reference 21-18)
- Diamond Wire Sawing (e.g., RPV closure head and upper reactor internals at Vermont Yankee (Reference 21-17)

Should inappropriate methods be selected, OPEX has demonstrated that this can lead to unnecessary levels of particulate generation (a waste management complication, and hindrance to visibility within the Fuel Pool (FP) should in-pool cutting be used).

The RPV removal at Big Rock Point is an example whereby the RPV could be removed intact (suggesting the same is possible for other large components), where a sufficient opening is incorporated into design and heavy lifting equipment with sufficient weight capacity used (Reference 21-10). Full removal of RPVs from installation positions, followed by in-air sectioning in a ventilated enclosure has been completed at Neckarwestheim I (Germany), Barsebäck (Sweden) and Oskarshamn (Sweden) NPPs, “Uniper Completes Dismantling of Two RPVs in Parallel,” (Reference 21-19). IAEA TECDOC-1657 (Reference 21-10) discusses the potential benefit of modular construction for facilitating removal of large components whilst reducing operator exposure.

21.1.7 Approach to Decontamination

Decontamination prior to dismantling can significantly reduce worker radiation dose, minimise airborne contamination and yield Out of Scope waste from existing Low-Level Waste (LLW). However, the caveat is that it is only justifiable where radiation exposure during decontamination does not counteract the reduced dose in future handling/dismantling of the decontaminated components.

There has been continuous development of decontamination equipment and techniques internationally to enable effective decontamination whilst reducing risks ALARP (Reference 21-16), “Decontamination Methodologies and Approaches,” (Reference 21-20).

Choice of decontamination method is informed by radiological characterisation processes, to ensure that secondary waste generation is controlled, and the type, source and level of contamination understood. Examples of successful decontamination approaches utilised in past decommissioning activities are outlined below:

- Full System Decontamination of primary circuit and auxillary systems: Areva's Chemical Oxidation Reduction Decontamination UV (CORD® UV) and EPRI's DFD (decontamination for decommissioning) have been used across a number of plants, both during operation and during decommissioning, to signicantly reduce contamination levels in these systems, "Full System Decontamination with the HP/CORD UV for Decommissioning of the German PWR Stade, in Structural Mechanics in Reactor Technology," (Reference 21-21) and "Decontamination of Reactor Sytems and Contaminated Components for Disposal or Refurbishment: Developments and Experience with the EPRI DFD Chemical Decontamination Process," (Reference 21-22).
- Concrete Scabbling: a widely used method to remove the contaminated outer layer of concrete surfaces, often that used in buildings. It is a useful dry decontamination method for minimising airbourne contamination and waste levels when used in combination with dust collection apparatus, "Decontamination Techniques Used in Decommissioning Activities," (Reference 21-23).
- Ultra High Pressure Water Jet: used to remove contaminated layers (usually oxide layer) from structural steel components, with higher pressures giving increased contaminant removal rate and greater penetration depth. Should be used in a closed environment to contain waste water, "State of the Art Technology for Decontamination and Dismantling of Nuclear Facilities," (Reference 21-24).

21.1.8 Justification of OPEX Relevant to the Safe Decommissioning of BWRX-300

The BWRX-300 design draws on international experience of successful BWR decommissioning, utilising available and suitable techniques. The current generation of BWRs are largely expected to have been decommissioned, or be under safe-storage, by the time BWRX-300 decommissioning begins. Subsequent Learning from Experience (LfE) can feed into finalising the approach to decommissioning adopted, particularly through considerations at the design stage.

21.2 Decommissioning Strategy

The objective of the decommissioning process is to permanently retire the BWRX-300 NPP from service in a manner that will ensure the health, safety, and security of workers and public, and protect the environment.

The BWRX-300 decommissioning strategy will draw upon the lessons learned and experiences from the decommissioning of similar reactors. With the removal of fuel from the reactor, activities associated with contaminated Structures, Systems, and Components (SSCs) will be considered first when deciding on a decommissioning strategy.

21.2.1 Identification of Decommissioning Strategy Options

A decommissioning strategy will be selected early in the BWRX-300 SMR life cycle, to form the basis for planning for decommissioning and facilitate achieving the desired end-state of the decommissioning project. The following are the established strategy timing options for consideration (as recognised by "Decommissioning of Facilities, IAEA Safety Standards, General Safety Requirements," (Reference 21-25)):

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

Decommissioning strategies are evaluated in a systematic and traceable fashion so that the relative advantages and disadvantages of each strategy can be objectively compared. When determining the appropriate decommissioning strategy, the following are to be considered, as appropriate, 006N8745, "BWRX-300 Incorporation of Decommissioning in Design Considerations," (Reference 21-26):

- Public engagement
- OPEX and lessons learned
- Forms and characteristics of radioactive and non-radioactive contamination
- Integrity of containment and other SSCs over time
- Availability of decontamination, disassembly, and clean-up technologies
- Potential for recycling or re-use of equipment and materials
- Availability of knowledgeable staff
- Potential environmental impacts
- Potential worker and public radiological doses
- End-state objectives and site redevelopment plans
- Potential revenues, costs, and available funding
- Availability of waste management facilities, locations, or sites
- Interdependencies with other facilities, locations or infrastructure located at the same site

- Assurance that the BWRX-300 site is maintained in a safe configuration
- Principles of radiation protection, justification, optimisation, and application of dose limits

In-depth studies will be performed as warranted over the life cycle of the plant, to refine and solidify the recommended decommissioning strategy as part of the regular review cycle to account for the following issues, which may have relevant consequences for decommissioning (Reference 21-26):

- Changes in site conditions, or incidents and events
- Changes to the proposed decommissioning objectives
- Changes to ownership or management structure
- Advances in decommissioning technology
- Significant modifications to the facility, location, or site
- Updated schedule, cost, and funding information
- OPEX and lessons learned
- Revised regulatory requirements
- Availability of facilities, locations, or sites for the management of radioactive waste

Options for decommissioning are being considered and would be assessed as part of the finalised decommissioning plan produced by the future licensee.

21.2.2 Assumed Decommissioning Strategy for BWRX-300

A possible decommissioning strategy is presented for the BWRX-300 standard design, with further development expected at later stages of the pre-licensing process. The final decommissioning strategy, including timelines and a full decommissioning plan, will be chosen during reactor construction and operation by the future licensee, incorporating evolving best practice and site-specific considerations. An approximate timeline indicating the key decommissioning milestones for the BWRX-300 standard design is shown in Figure 21-1.

A prompt decommissioning strategy has been assumed for BWRX-300 sites, in line with IAEA preference (GSR Part 6 Requirement 8 (Reference 21-8)). However, for strategy planning purposes, a C&M period of up to 30 years has been allowed for (retaining the possibility for deferred decommissioning, if chosen, with the benefit of allowing further decay to reduce exposures), should immediate dismantling become unviable. There is an assumed dismantling and demolition period of five years. Site restoration and final surveys are assumed to take one year (Reference 21-26).

An indication of the milestones and activities anticipated in the decommissioning of the BWRX-300 site is shown in Figure 21-1. This timeline is subject to alteration during the final decommissioning planning process, with responsibility for constructing and maintaining a programme lying with the future licensee.

21.3 Facilitating Decommissioning During Design and Operation

21.3.1 Decommissioning Design Principles

OPEX demonstrates that decommissioning of reactor facilities is facilitated if considered during the design phase. Assessment of future facility decommissioning and dismantling activities at the design phase includes consideration of OPEX gained from the decommissioning of existing facilities, as well as those facilities that are in long-term safe storage. This expectation is in line with IAEA GSR Part 6 Paragraph 7.3 (Requirement 10-Planning for decommissioning) (Reference 21-8).

With respect to the BWRX-300, design lessons from decommissioning have been collated by various international bodies, notably the IAEA TECDOC-1657, “Design Lessons Drawn from the Decommissioning of Nuclear Facilities,” (Reference 21-10) and review of on-going feedback will also inform future decommissioning decisions. International OPEX has demonstrated that this form of reactor can be readily decommissioned, in compliance with regulations and safety principles, and most likely will be subject to enhanced methods in comparison to ongoing and completed decommissioning projects. The small modular design, materials choices, SSC design, and benefit of improved decontamination and dismantling techniques are expected to further reduce decommissioning complexity.

Furthermore, OPEX evaluations are a required part of the design process and documented within the design basis records of each system. The GEH design control process, as described in Section 21.3.3, details the process by which operating experience is incorporated into the design features of the BWRX-300.

Based on this consideration of OPEX, and incorporating the IAEA provisions for safety during decommissioning (SSG-61, Section 3.21.8 (Reference 21-9)) and the requirements detailed in IAEA GSR Part 6 (Reference 21-8), the following are key considerations for future plant decommissioning and dismantling activities:

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

Evidence supporting how these considerations for decommissioning have been incorporated into the BWRX-300 design is documented in Section 21.5.1, which details elements of the design which facilitate safety during decommissioning.

21.3.2 Record Keeping

International decommissioning experience identifies that operational records should be retained to meet the needs of future decommissioning. Decommissioning related documentation will be managed and maintained by the licensee in accordance with paragraph 7.7 of GSR Part 6 (Reference 21-8). These records will contain historical information that may be required in the future in order to update the Decommissioning Plans and facilitate successful decommissioning. Records pertinent to decommissioning will be kept in the storage medium in standard use. All records will be assembled and maintained in accordance with the document and record management process and governance (Reference 21-26).

21.3.3 Requirements Management and Design Control Processes Relevant to Decommissioning

21.3.3.1 Requirements Management

Decommissioning considerations are integrated into the design process through a structured Requirements Management Plan (RMP), 005N9036, “BWRX-300 Requirements Management Plan” (Reference 21-27). This plan for the BWRX-300 outlines a structured approach to requirements traceability, which helps ensure that all design activities are aligned with the overarching objectives, including decommissioning. By maintaining a clear traceability framework, the RMP allows for the identification and documentation of decommissioning requirements early in the design phase, ensuring that these considerations are integrated as the design matures. Below is a narrative detailing how the flow-down of decommissioning considerations is facilitated through the RMP, incorporating relevant design criteria:

- Source Requirements Identification: Decommissioning considerations are initially captured as part of the source requirements. These requirements are identified through stakeholder input, regulatory guidelines, and industry best practices. The RMP ensures that decommissioning requirements are clearly defined from the outset, forming a crucial part of the project's objectives.

- Requirements Traceability: The RMP establishes a traceability framework that tracks these decommissioning requirements throughout the design phases. This traceability ensures that all design decisions align with the initial decommissioning considerations, facilitating consistent integration as the project progresses.

- Graded Approach: Within the RMP, the graded approach allows for varying levels of detail and focus depending on the phase of the design process. During early design phases, broad decommissioning goals are considered, while later phases incorporate more specific and detailed decommissioning requirements. This ensures that the requirements are neither too vague nor overly detailed, maintaining efficiency and clarity in the design process.

- Standard Design and Baseline Phases: The multi-phase design process includes Baseline 0 through Baseline 3. Initially, the Standard Design developed through Baseline 2 incorporates general decommissioning considerations applicable to all projects. As the design enters Baseline 3, site-specific decommissioning requirements are addressed, ensuring that these considerations are tailored to individual project needs and constraints.

- Decision Hierarchy: The decision hierarchy within the RMP involves various levels of review and approval to ensure decommissioning considerations are appropriately weighted in design decisions, as discussed further in Section 21.3.3.3 below. This hierarchy involves project managers, design engineers, and stakeholders, who review and validate that decommissioning requirements are being met and integrated effectively at each stage.

- Feedback and Continuous Improvement: Throughout the design process, there is a mechanism for feedback and continuous improvement. This allows for revisiting decommissioning considerations as new insights or technologies emerge, ensuring that the design remains aligned with evolving decommissioning standards and practices.

21.3.3.2 Integrated Design Engineering Process

The integrated design engineering process, as detailed in 006N3139, "BWRX-300 Design Plan" (Reference 21-28), demonstrates the relationship between requirements and the generation of design. This process begins with the collection of top-level requirement sources, encompassing product specifications, regulatory standards, and owner requirements, as well as iterative safety analyses derived from Human Factors Assessments and Deterministic/Probabilistic Safety Analyses.

These requirements are systematically decomposed and linked to lower-level requirements, which pertain to plant, system, and component specifications. The gathered requirements and associated information/data are channelled into Design Data Generation, where engineering disciplines converge in collaborative, data-centric workspaces. Utilising 3D modelling and system simulation tools, engineers explore system design options, optimise solutions, and evaluate various aspects such as feature and function choices, performance margins, and cost-effectiveness.

System design options include OPEX, past BWR designs, and a Research and Development program. At the conclusion of each design phase, design reviews and the generation of design documents are conducted. These comprehensive design reviews, involving all relevant disciplines, focus on scrutinising system requirements and the design itself. Outcomes from these reviews may drive modifications to either the requirements or the design.

For example:

- The plant design incorporates features that facilitate future decommissioning per REGDOC-2.5.2 (Reference 21-29), Section 7.24 Decommissioning, in 006N4173, "BWRX-300 Composite Design" (Reference 21-30) requirement [48008], Section 4.4.1, including:
  - Material selection that minimises the potential for radioactive waste generation and facilities decontamination
  - Plant layout that facilitates access for decommissioning or dismantling activities
  - Potential requirements for storage of radioactive waste that may arise due to construction of additional new facilities or expansion of existing facilities
- As the design matures, decommissioning considerations may be communicated to plant designers that is based on plant requirements, common requirements, system requirements, or component level requirements described in the following plant design documents, but not limited to:
  - 006N5081, "BWRX-300 As Low As Reasonably Achievable Design Criteria for Standard Design" (Reference 21-31)
  - 006N7859, "BWRX-300 Plant Layout Design Criteria" (Reference 21-32)
  - 006N2829, "BWRX-300 Human Factors Engineering Design Requirements" (Reference 21-33)

21.3.3.3 Design Reviews

Design reviews are conducted per GEH Common Procedure CP-03-100, "Design Control," (Reference 21-34) at various phases of the design plan, 006N3139, "BWRX-300 Design Plan" (Reference 21-28), with design control progressing as the design matures. Design verification is performed prior to releasing a design for procurement, manufacture, construction, or use by another design organisation. Design verification is performed to ensure that the design complies with the customer requirements, technical requirements, regulatory requirements, and codes and standards. Verification ensures that appropriate design methods are used, that Design Inputs are correctly incorporated into the design, and that the Design Output is reasonable when compared with the Design Input.

There are three acceptable methods for performing design verification and more than one method can be used:

1. Design Review by individuals or cross-discipline teams provides assurance that the final design is correct, and in compliance with requirements, by reviewing Design Inputs, compliance to requirements, assumptions, design methods, and computer programs.
2. Alternate calculation by alternative methods are required for verification of calculations unless Pre-verified Computer Programs are used within their applicable defined limits.
3. Qualification tests demonstrate adequacy of performance under conditions that simulate the most adverse design conditions. Where the test is intended to verify only specific design features, the other features of the design are verified by other approved verification methods.

Design verifications are performed at each level of the requirement flow path, from plant level to system level to component level. This approach ensures issues are identified and addressed early in the design.

21.4 Decommissioning Plan

The BWRX-300 Decommissioning Plan (under development) will outline the overarching decommissioning sequence and describe the design features to facilitate this. It will then be developed into the UK-specific Preliminary Decommissioning Plan (PDP), which will be prepared during the next stage of design development (see the Forward Action Plan item PSR21-154, provided in Appendix B).

The Decommissioning Plan and subsequent UK-specific PDP will describe the overall sequencing of the decommissioning and disassembly of the BWRX-300, which will be facilitated by ensuring that the design incorporates appropriate decommissioning considerations, as described in Section 21.3.1. Details on how these considerations have been incorporated into the BWRX-300 design are provided in Section 21.5.

Based on the examination of decommissioning strategy available, the provisional stages comprising the decommissioning plan are laid out below. The timings and actions described are tentative, and subject to continuous revision in-line with future licensee decisions and evolving best practice.

21.4.1 Decommissioning Stages

The IAEA have established a sequence for decommissioning common to all nuclear facilities, based on programmes to date, “Approaches for Decommissioning Strategies and Lessons Learned So Far,” (Reference 21-35) and “A Taxonomy for the Decommissioning of Nuclear Facilities,” (Reference 21-36):

A. Preparation for decommissioning – begins approximately 5 years prior to reactor shutdown and involves finalising the decommissioning plan and obtaining regulatory approvals

B. Post-shutdown Operations (Transition) – those operations critical to allowing safe decommissioning activities are carried out, including removal of SF from the reactor, and placing systems into an inactive, safe state

C. Care and Maintenance (*optional*) – site placed in a deferred decommissioning state, where essential monitoring and maintenance work is conducted, and reactor status is continually recorded, but safety is generally maintained by passive means

D. Dismantling and decontamination – dismantling is the reduction in size of components to ease removal and containment, whilst decontamination is the removal (either fully or partially) of radioactive substances or material from component surfaces

E. Demolition – the process of removing the on-site buildings, SSC without the need for nuclear-specific practices

F. Site Clean-up – also known as remediation, constitutes the removal of contaminated soil and other materials from within the site boundary, with the aim of allowing the determined land ‘end state’ to be reached

G. Termination of Authorisation for Decommissioning (Site End State) – the point when a site is ready for its next intended use

21.4.1.1 Transition

The Transition phase will commence immediately following the end of commercial operations and be complete once the physical, operational, and administrative Transition is confirmed.

Some of the key Transition activities are defueling and dewatering, radiation surveys, and removal of hazardous material. This process includes POCO, whereby tasks are carried out immediately after shut-down with the aim to reduce the hazards/risks of later decommissioning.

The Transition and POCO goals and objectives include, but are not limited to (Reference 21-26):

- Defuel and removal of fluids, gases, hazardous materials, and other substances (e.g., resins) from the systems
- Continue to safely, and securely, store nuclear substances, such as irradiated fuel, until it can be removed to dry storage
- Maintain the site in a safe and stable condition while creating no new hazards
- Reduce the footprint of the station in preparation for the next phase of decommissioning
- Protect workers, the public, and the environment from residual radioactive sources and hazardous materials remaining at the BWRX-300 site and maintain exposures ALARP

As Transition activities progress, systems that are no longer required to support the operation of the plant will be placed into an inactive safe state. That is, they will be de-energised, drained of gas or fluids and isolated from operational systems. The operational, inspection, and maintenance requirements (as needed) for each system (or groups of related systems) will be identified and documented. Thereafter, the following activities will take place, as necessary:

1. Reviewing and revising operational programs to ensure requirements for the remaining phases of decommissioning are met. Examples include, but are not limited to, environmental monitoring, radiation protection, emergency response, and fire protection. Plans and protocols, developed during the detailed planning phase, for monitoring the following would be submitted to the Nuclear Decommissioning Authority:
    i. Work hazards during decommissioning
    ii. Personnel dosimetry
    iii. Environmental emissions and effluents
    iv. Materials, sites, and structures to be cleared from regulatory control
2. Developing a safety assessment framework to manage the nuclear and reactor safety aspects of Transition activities
3. Engaging stakeholders in Transition planning activities

In support of Transition to decommissioning, a safety assessment is also completed, including controls and approvals, to facilitate the shutdown and Transition of the station. The objectives for the safety assessment include:

1. Demonstrate that applicable regulatory requirements are met throughout Transition
2. Demonstrate, through systematic hazard analyses, that the risks posed by hazards due to both Transition activities and for accident conditions are understood and managed
3. Identify necessary mitigating measures, limit controls and conditions to meet safety criteria throughout Transition
4. Quantify the hazard reduction to be achieved through Transition activities

21.4.1.2 Care and Maintenance

Should a Care and Maintenance (C&M) period be entered (although, given prompt decommissioning is the assumed strategy, C&M is not anticipated), the duration will be detailed as part of the final decommissioning plan prior to shutdown. The period allows for some decay of residual activation and fission products that remain in the station's systems prior to commencing dismantling and demolition activities.

During C&M, SSCs that would remain necessary to support continued operations (i.e., in an active safe state) will be modified or reconfigured as necessary to meet operational demands.

Programs will continue to support station operations and will be revised as necessary, similar to what is performed for Transition, as identified above. Programs and procedures will be adapted to meet regulatory requirements, while remaining commensurate with the complexity and risks of the C&M operations, and any revisions to these programs and procedures will require acceptance by regulators, where applicable.

During the Care and Maintenance phase, the future site licensee will be expected to perform continuous monitoring and surveillance of the facility to ensure worker, public, and environmental safety is maintained.

An adequate level of security will be provided during C&M. Fire and radiation alarms are also to be monitored and maintained. Adequate plant staff will be available during this period to support the maintenance, inspection, and surveillance programs, as necessary.

21.4.1.3 Dismantling, Decontamination, Demolition and Site Restoration

Dismantling, decontamination, and demolition work begins after the necessary licensing for decommissioning is approved. Dismantling work has six conceptual steps:

1. Prepare the buildings and site
2. Decontaminate and dismantle nuclear systems
3. Dismantle non-nuclear systems
4. Demolish buildings
5. Manage and dispose of the waste
6. Restore the site

Work in the above steps may occur in parallel. Remedial action to support surveys for radioactive and other hazardous materials will be performed throughout the dismantling work, up to the final survey.

21.4.1.4 Dismantling

Dismantling activities are anticipated to involve the following (Reference 21-26):

1. Construct temporary facilities, modify existing storage facilities, and erect and place scaffolding in and around components to be dismantled to support the dismantling and decontamination activities. These may include a cutting station (as needed for large components), additional change rooms and contaminated laundry facilities for increased work force, establishment of laydown areas to facilitate equipment removal, upgrading roads to facilitate hauling and transportation, and modifications to the Reactor Building (RB) to facilitate access of large/heavy equipment.
2. Modify containment to support segmentation activities and prepare rigging for segmentation and extraction of heavy components.
3. Conduct decontamination of components and piping systems as required to control (minimise) worker exposure. Remove, package, and dispose of all piping and components that are no longer essential to support dismantling operations. It is anticipated that radioactive corrosion products on inner surfaces of piping and components will not have decayed to levels that will permit unrestricted use or allow conventional removal. These systems and components are surveyed as they are

removed and disposed of in accordance with the radiological clearance levels that have been developed.

4. Remove systems and associated components as they become non-essential, such as those related to decommissioning activities or worker health and safety. Remove systems (piping and components) utilising contamination control practices, remote tooling, and packaging in approved transportation containers for disposal.
5. Remove reactor vessel and reactor internals. Package in LLW or Intermediate-Level Waste (ILW) containers (as appropriate).
6. Remove activated and accessible contaminated concrete. Remove those portions of the associated enclosures necessary for access and component extraction.
7. Remove all remaining LLW and ILW along with any remaining hazardous materials. Material removed in the decontamination and dismantling of the nuclear unit will be routed to an on-site central processing area. Material that meets clearance criteria will be released for unrestricted disposition (e.g., as scrap, recycled or general disposal). Contaminated material will be characterised and packaged for controlled disposal at the long-term disposal facilities for respective LLW and ILW.
8. Remove remaining components, equipment including tools used in dismantling and demolition, and plant services in support of the area release survey(s).
9. Conduct final radiation surveys to ensure that all radioactive materials in excess of permissible residual levels have been remediated.

Dismantling work performed on contaminated nuclear systems will be conducted in a manner that will minimise the spread of contamination. Appropriate contamination control techniques, including the use of portal monitoring systems at controlled egress points, temporary enclosures, local ventilation, Personal Protective Equipment, and contamination monitoring, will be used when the work is performed.

21.4.1.5 Demolition

Once contaminated systems, structures and non-nuclear systems have been dismantled and final surveys have confirmed that the remaining structures are below the radioactive and hazardous materials release limit (Out of Scope), demolition activities may begin.

Efficient removal of the contaminated materials and verification that residual radionuclide concentrations are below regulatory limits will result in substantial damage to many of the structures. Remaining buildings and above ground structures will be removed using conventional demolition techniques.

Remaining structures (including buildings that were not contaminated and temporary structures) will be demolished by general demolition crews. Waste blocks will be sized so that they can be handled and moved by the available technologies. Foundation and exterior walls will be removed to the nominal one-metre removal depth below grade whenever possible. At-grade foundation slabs exceeding one meter in thickness will be abandoned in place and covered with a one-metre-thick layer of backfill, as needed. Concrete rubble and clean fill produced by demolition activities may be used to backfill voids. Suitable materials will be used for filling, otherwise the rubble will be trucked off-site for disposal as construction debris.

Piping that exceeds the release criteria will be removed and disposed of appropriately. Clean metal piping will be considered scrap or will be recycled. Road and parking areas with asphalt or concrete surfacing are broken up and the rubble used for backfilling on-site if needed or disposed of appropriately.

21.4.2 Plan Management and Maintenance

The decommissioning plan is an evolving document anticipated to undergo continual revision throughout the reactor lifetime. As part of efforts to ensure information is retained till the onset of decommissioning activities, the expectation is that the site licensee will prepare a site-specific PDP and submit it to the regulators for acceptance as early as possible in the lifecycle of the facility or the conduct of licensed activity. The site-specific PDP will be developed from the UK-specific PDP described previously in Section 21.4. The PDP should include:

- Detailed description of the location of the facility (maps, geographic information, surrounding environment details, land used and municipality information)
- Purpose and description of facility (components and systems, building types and services, hazardous materials list, quantity and types of materials and design features to facilitate decontamination and dismantling)
- Post-operational conditions (summary of planned removal of materials, expected nature and extent of contamination, identification of separate planning envelopes)
- Decommissioning strategy (end-state objectives, strategy overview, rationale)
- Work breakdown structure (steps, schedule)
- Radiological monitoring and survey commitments
- Cost and financial guarantee details
- Public consultation plan

The licensee will review and update the PDP and submit it to the regulators as required by regulation or as requested by regulators. The PDP should be updated in light of the following considerations:

- Changes in site conditions, or incidents and events with relevant consequences for decommissioning
- Changes to the proposed decommissioning objectives
- Changes to ownership or management structure
- Advances in decommissioning technology
- Significant modifications to the facility
- Updated schedule, cost and funding information
- Operational experience and lessons learned
- Revised regulatory requirements
- Availability of a facility for the management of irradiated fuel and radioactive waste

Prior to executing decommissioning activities, the licensee will prepare and submit a Detailed Decommissioning Plan (DDP) to the regulators for acceptance. The DDP refines and adds details to the PDP. The DDP will: document the final end-state objectives; include a description of any institutional controls; provide a decommissioning strategy; include a waste management plan; consider potential environmental effects; include a cost estimate and financial guarantee arrangements. Once accepted by regulators, the DDP will be incorporated into a license authorising decommissioning.

GEH is currently developing a Decommissioning Plan that will serve as a basis for owners to use in the development of their site-specific PDP and DDP. This document will capture and address the decommissioning regulations and requirements for the region.

21.5 Provisions for Safety During Decommissioning

21.5.1 BWRX-300 Design Elements to Facilitate Decommissioning

Design features relevant to decommissioning, in-keeping with the design principles, are discussed in the sub-sections below (based on information provided in (Reference 21-26)):

21.5.1.1 Site Plot Plan

The BWRX-300 design has been optimised for constructability, with a focus on simplifying safety systems and fewer pools of water incorporated. A compact design, which minimises below ground excavation and buried utilities where possible, may be beneficial for dismantling the facility during decommissioning. Space allocation is considered to accommodate construction activities (i.e., access area around the plant, areas for laydown, etc.), thereby facilitating the ability to decommission and dismantle the plant once a license for decommissioning is granted (Reference 21-26).

21.5.1.2 Modular Construction

A modularisation strategy will be used to design and construct the BWRX-300 at the designated site. The module and skid assemblies are intended to be built off-site, transported to the site, and erected on-site. This modularisation strategy will provide guidance in selection of disassembly methods employed during decommissioning (Reference 21-26).

21.5.1.3 Control of Materials During Design

Specific guidance has been exercised for plant systems materials to minimise corrosion products during plant operation is a design requirement, which provides restrictions regarding the use of cobalt-based alloys and cobalt in stainless steel and nickel-base alloys (refer to Section 3.13, 006N5956, “Materials and Process Controls,” (Reference 21-37)). These restrictions reduce personnel dose exposure during plant operation and decommissioning activities (Reference 21-26).

Materials choices consider minimising corrosion products, including protecting non-stainless-steel equipment with a non-corrosive layer to aid decontamination.

21.5.1.4 Simplified Boiling Water Reactor Design

Compared to existing BWR designs, the BWRX-300 design is much simpler. This simplified design will have the concurrent effect of enabling and aiding the decommissioning process. Aspects of BWRX-300 plant simplification most relevant to facilitating decommissioning are:

- Natural circulation of reactor coolant – by utilising a tall vessel, natural circulation drives steam/water flow in the BWRX-300 reactor core, thus the need for primary coolant pumps and the associated piping is removed. This in turn reduces the amount of components exposed to high radiation levels in the core region, 005N971, “BWRX-300 General Description,” (Reference 21-38).
- Direct-cycle – there are fewer and smaller highly contaminated pieces associated with the nuclear steam supply system for a direct-cycle plant (Reference 21-38).
- Primary Containment System (PCS) – use of a dry containment system has eliminated the need for a suppression pool (Reference 21-38), which reduces the quantity of potentially contaminated water. The Steel-Plate Composite Containment Vessel (SCCV) integrated into the embedded RB and designed to be as small as feasible, reducing back-fill requirements. The Isolation Condenser System is not contained within the PCS, which also reduces the number of components and areas susceptible to contamination.
- Plant and SCC design – as part of the BWRX-300 ‘simplicity’ and ‘design for constructability’ construction philosophy, SSCs are simpler in design and fewer in

number compared to existing LWRs, resulting in a reduction in building materials. Additionally, features are included in the plant design to facilitate construction and decommissioning activities, including sequenced installation and removal, staging, disposal, handling, storing, monitoring, and security of SSCs.

- Reactivity control – control rods are the only method of reactivity control under normal operation and are designed to reduce the amount of stainless steel within the high fluence RPV region. This gives a reduction in activated material and decommissioning waste volume. Elimination of the Recirculation System provides for simplified dismantling and a significant reduction in contaminated decommissioning waste.
- Accessibility of RPV internals – the BWRX-300 RPV has bolted internal components designed to be removed (vice welded in place), which simplifies the handling of contaminated materials (Reference 21-38). Disconnect-able control rod drives and instrumentation tubes from the reactor vessel lower head were designed to support efficient disassembly for decommissioning.
- Skid-mounted radwaste process systems – the radwaste process systems are skid-mounted and located in the Radwaste Building to allow truck access, and system skid loading and unloading, and skid removal during dismantling. This approach differs from many existing plant designs, which require extensive demolition to remove system components. This allows for easier access, loading, and unloading, both during operational lifetime and during decommissioning
- System Isolations – as identified in system Piping and Instrumentation Diagrams, isolation valves are incorporated at locations required to enable key systems required for decommissioning (Fuel Pool Cooling and Cleanup, Compressed Air, Liquid Radioactive Waste, Solid Radioactive Waste) to continue operation while non-key systems (power production related) are disconnected and dismantled, ensuring processing and cooling of spent fuel, air systems for cutting and layup, demineralised water systems for flushing, hydraulic cutting, decontaminating and processing liquid/solid radwaste volumes during POCO.
- Embedded piping – the BWRX-300 principal design requirements exclude most embedded piping unless it is required due to being in a base mat or a penetration in a wall. Embedded piping is limited to U50 (Equipment and Floor Drain System) floor drains in the Balance of Plant grade elevation slabs. These instances are required by design to vertically transfer wastewater from the concrete floor of each elevation to subsequent levels below, until reaching the associated sumps. Existing methods of cutting out and storing as radiological waste will be utilised during dismantling. No contaminated piping exists buried in the ground, only cooling water and conduits. The plant layout document, 006N7859 (Reference 21-32), Sections 6.1.13 and 6.1.14 provide details for these requirements. The original primary design of the plant is to exclude the issues from older BWR, PWR, and CANDU plants that have contaminated the ground water, soil, or concrete. The limiting buried or embedded piping also reduces the decommissioning costs and schedule.
- Chimney design – since the BWRX-300 utilises a taller ‘chimney’ design, it is assumed that some RPV components (e.g., moisture separator, steam dryer, etc.), located a greater distance from the neutron flux, will be less irradiated than similar components in previous designs, such as the Advanced Boiling Water Reactor. This will reduce the volume of decommissioning HAW.

21.5.1.5 Site Building Accessibility

BWRX-300 site design considers the requirement for ease of component removal once decommissioning commences. This includes the installation of entry doors across the estate suitable for enabling large equipment removal, and numerous cubicles with equipment hatches inside the buildings. The RB hatch is sized such that components (i.e., fuel pool and cleaning system heat exchangers) can fit through. Air locks in the SCCV wall shall be sized to accommodate removal of the most limited or largest components, and space shall be provided around equipment locations inside the SCCV (initially to allow maintenance activities, but these features will also help facilitate decommissioning). Floor hatches at each level of containment are sized to remove the equipment in the levels below.

The Plant Service Area (PLSA) was designed to support decommissioning. 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 Control Building and Turbine Building. The functions performed in the Reactor Auxiliary Bay include new fuel and spent fuel cask transit, equipment ingress and egress to the RB, and removal of plant components during dismantling. Additionally, a decontamination facility was incorporated into the Plant Services Area design, which provides a central location to equipment locations to receive, decontaminate, temporarily store, and ultimately transport plant components during disassembly. The RB, Turbine Building (TB), and Radwaste Building are designed for removal of the most limiting large equipment, consisting of entry doors from the outside and numerous cubicles with equipment hatches inside the buildings.

21.5.1.6 Allowances for RPV Disassembly

Space was incorporated into the BWRX-300 design for installation of a temporary vessel support structure below the vessel lower head, such to support the lower portion of the vessel once the vessel support skirt is cut free. This contributes to an efficient method of core component separation and removal.

21.5.1.7 Reduction in Waste Yields

The BWRX-300 will give lower lifetime waste yields compared to current generation LWRs and a lower end-of-life decommissioning volume. This is due to use of improvements from the current fleet of BWRs, including the following design features/considerations:

- The use of improved GNF2 fuel, as described in NEDC-34159P, “BWRX-300 Fuel Summary Report,” (Reference 21-39) is anticipated to result in a lower incidence of fuel cladding failures that will positively impact on all downstream source term values, helping further drive a reduction in decommissioning waste.
- Quantifiable increased use of stainless steel throughout the design – this is anticipated to result in reduced corrosion and erosion particulate generation throughout the plant, and lower reactivity in decommissioning waste. This will both reduce wet solid waste volumes and result in fewer particles undergoing irradiation in the core, reducing the overall radioactivity of the filter backwash sludges produced. Reduced presence of particles in the RPV is also complementary to reduced fuel clad failures.
- Reduced cobalt inventory – the BWRX-300 material selection strategy presented in (“BWRX-300 Materials and Process Controls,” (Reference 21-37) focusses on reducing cobalt inventory wherever practicable throughout the plant design.
- Enhanced water chemistry regime – this is anticipated to result in further reduction of corrosion and erosion particulate as above, and minimisation of cobalt deposition on coolant facing surfaces. Reactor water quality and related circuit cleanliness is further enhanced by an integrated water chemistry regime comprising Hydrogen Water Chemistry, proprietary On-Line NobleChem™ and GE Zinc Injection Passivation. The water chemistry regime is anticipated to result in further reduction of corrosion and erosion particulate, leading to reduced volumes of filter backwash sludge, and minimisation of cobalt deposition on coolant facing surfaces which will have a beneficial impact on decommissioning waste volumes.

- BWRX-300 utilises a 'skid mounted' approach to provide a series of effluent treatment sub-systems that can be configured to cope with an array of effluent scenarios arising from upstream plant events. This provides significant mitigation against upstream faults and errors that could result in the generation of effluents that are beyond the normal plant parameters (e.g., increased fission products due to a fuel clad failure, ingress of foreign materials such as organics, oils, greases from a foreign materials intrusion, or chloride ingress from a main condenser tube leak). This approach is consistent with that applied at the Tokyo Electric Power Company Fukushima-Daiichi site as part of the Advanced Liquid Processing System, as presented in "Overview of the Multi-nuclide Removal Equipment (ALPS) at Fukushima Daiichi Nuclear Power Station," which provided a substantial amount of operating experience to support design development of the LWM system.
- Aqueous Effluent Filter Modules - The backwash-able fine filters utilised in the CFD, FPC and LWM (including RWST) systems comprise a vessel housing the filter modules. The vessels are backwashed to remove adherent sludge, and this is flushed to the SWM sludge storage tank. Over time, the filter media itself will degrade, and underlying differential pressure across the filter will rise, ultimately resulting in a requirement for the filter modules to be replaced. Backwash-able filters are anticipated to last for more than ten years if operated under an appropriate regime. Due to the highly efficient backwash process, the filter modules themselves are anticipated to contain very little radioactive waste and are assumed to arise as LLW.

Furthermore, a 12-month fuel cycle has been shown to be more efficient compared to 18 or 24-months – producing fewer used fuel assemblies. The focus on design-to-cost and a reduced construction schedule duration gives a construction method which yields less material for decommissioning purposes (Reference 21-38).

21.5.2 Provisions for Safety During Decommissioning

Decommissioning considerations made in order to reduce risks in the design of the BWRX-300, include the following aspects (Reference 21-26):

1. Shielding design considers protection during maintenance, inspection, decommissioning, and operations.
2. The RB, TB, and Radwaste Building are designed for large equipment removal, consisting of entry doors from the outside and numerous cubicles with equipment hatches inside the buildings.
3. The radwaste process systems are skid-mounted and located in the Radwaste Building to allow truck access, and system skid loading and unloading.

BWRX-300 design features that facilitate decommissioning by maintaining low occupational exposures are detailed in Sections 12.1.1 and 12.3 of PSR Chapter 12: Radiation Protection (Reference 21-2). Examples are (Reference 21-26):

- Provisions for draining, flushing, and decontaminating equipment and piping
- Design of equipment to minimise the buildup of radioactive material and to facilitate flushing of piping systems
- Shielding that provides protection during maintenance, inspection, and operations, that may facilitate decommissioning
- Provision for adequate space for utilisation of movable shielding
- Separation of more highly radioactive equipment from less radioactive equipment

- Provision for separate shielded compartments for adjacent items of radioactive equipment
- Provision for access hatches for the installation or removal of plant components
- Provision for the Reactor Water Cleanup System, Shutdown Cooling System, and the condensate demineraliser to minimise crud buildup

BWRX-300 design objectives that minimise radioactive contamination include (Reference 21-26) (further details are given in Section 12.3 of PSR Chapter 12 (Reference 21-2)):

- Provide containment in areas where leaks and spills are most likely to occur.
- Provide leak detection capability for prompt detection of leakage from SSC.
- Use leak detection methods (e.g., instrumentation, automated samplers) capable of early detection of leaks in areas where it is difficult (inaccessible) to conduct regular inspections (such as the FP and buried, embedded, or subterranean piping) to avoid release of contamination. BWRX-300 active liquid waste is kept within the Radwaste building in tanks located in cubicles.
- Facilitate decommissioning by minimising embedded piping, sumps, or buried equipment.
- Design the plant to facilitate the removal or replacement of equipment or components during facility operation or decommissioning.
- Minimise the generation of radioactive contamination and waste during operation and decommissioning by reducing the volume of components and structures that become contaminated during plant operation.

21.5.3 Decommissioning Wastes

Hazardous wastes generated during the dismantling and site clearance periods of the decommissioning will likely be limited to hazardous materials originally used as building materials. Volumes of these wastes are likely to be small, since very few hazardous materials are expected to be used in the construction of the plant. Dry active waste such as combustibles (paper, cloth, wood, filter cartridges) could also be generated in the removal of plant systems.

The bulk of the non-hazardous waste materials generated during decommissioning will be produced during the dismantling and site clearance periods of the decommissioning, although some are likely to be produced during the Transition from operations period.

In-keeping with U.K. government policy, the future availability of a GDF is assumed for the storage of HAW. Hazardous wastes will be packaged for transport and disposal according to the Waste Acceptance Criteria of Nuclear Waste Services Limited (NWS). All hazardous wastes, including nonradioactive hazardous wastes, will be transferred to an appropriate, licenced waste management facility for storage or disposal at approved disposal facilities. Waste manifests will be prepared and submitted to NWS. Mixed waste (i.e., radioactive waste mixed with clean waste that is also hazardous) will be transferred to an appropriate long-term disposal facility once it has been made available.

21.5.3.1 Waste Management

Prior to dismantling and demolition, the licensee will be responsible for detailing the approach to dismantling the facility and disposal of the resulting waste. This will include development of a waste management plan, applying the waste management hierarchy. Minimisation of waste generation is discussed in PSR Chapter 11: Management of Radioactive Wastes (Reference 21-1). Aspects of waste management to be considered during decommissioning are given below (Reference 21-26):

- The plan for both the short-term and, where possible, the long-term, for managing all decommissioning waste.
- Procedures for processing radioactive waste such as resins, filter media, metallic and non-metallic waste generated during the dismantling work.
- Determination of the transport and disposal container requirements for radioactive materials and hazardous wastes including the requirements for shielding and stabilisation of the waste.
- Procurement and testing of the transportation and disposal containers for radioactive materials and hazardous waste.
- Preparation of the detailed procedures for the packaging, removal and disposal of radioactive materials, hazardous waste, and construction debris.
- Assess/investigate decontamination methods such as chemical cleaning, electro polishing, mechanical abrasion or melting. These decontamination methods may be used to decontaminate scrap metal if the reduction in the volume of the scrap is sufficient to justify further processing. Depending on the efficiencies achieved, metals will be considered as either radioactive wastes for controlled disposal, lightly contaminated (or activated) for consideration for re-use within the controlled nuclear environment or metals that are decontaminated to levels below the clearance levels will be released for recycling in the open market.

21.5.3.2 Waste Estimates

Table 21-2 gives the estimated amounts of High Level (HLW), ILW, LLW and Very Low-Level Waste (VLLW).

Further development of enhanced estimates of the nature and volumes of radioactive and hazardous waste generated by decommissioning is required.

21.5.4 Long-term Irradiated Fuel Storage

For SF storage, assumed as the interim solution prior to moving the BWRX-300 fuel to a GDF or alternative long-term disposal facility, the BWRX-300 design incorporates equipment and facilities necessary to load used nuclear fuel into containers for transfer to a licensed on-site independent SF storage installation (Reference 21-26). An overview of the possible design approaches for this facility is laid out in Section 26.3 of PSR Chapter 26: Interim Storage of Spent Fuel (Reference 21-3). Criticality safety requirements are considered throughout the life cycle of the fuel. As part of the decommissioning plan, the last several fuel cycles of irradiated fuel will remain in the FP until they can be transported to an interim storage facility. SF is an operational waste stream (Reference 21-26).

The last cycle of irradiated fuel prior to plant shutdown will remain in the FP for up to 10 years (Reference 21-26). Full decommissioning activities (POCO and decontamination, followed by system dismantling) can only commence after all used fuel is removed from the FP. Factors that impact the SF storage period include:

1. Burn-up time for fuel bundles
2. FP capacity for SF (will be approximately 600 fuel bundles, ONR “Safety Assessment Principles for Nuclear Facilities,” (Reference 21-40))
3. Type of canister used - impacts the allowed minimum wet time, as there should be consideration for both thermal rejection to the environment and radiation shielding

21.6 Site Restoration and End-state

Whilst the final decision on the site end-state will be agreed upon following confirmation of the site characteristics and consultation with the relevant future stakeholders, the current proposal is for site restoration to an industrial end-state (commonly known as “brownfield”).

The design of the BWRX-300 standard plant includes considerations for removal of residual hazards, materials, and structures post-decommissioning, including minimising the resulting wastes needed to be disposed of, to facilitate a brownfield end-state that has the potential to be developed for new industrial uses.

By the end of the Dismantling and Demolition and Site Restoration period, the site will be free of industrial and radiological hazards. All radioactive contamination in excess of the established clearance levels and all other hazardous materials will have been removed from the site. It is expected that the clearance level used for the clean-up of the facility will not require institutional controls after the release from regulatory control. All of the station systems will have been dismantled and all of the buildings above the final finished grade demolished. Subsurface structures will have been drained and de-energised. These subsurface structures will also have been surveyed for contamination, decontaminated, if required, and dismantled to a nominal depth of one meter below grade (consistent with international practices), backfilled with clean concrete rubble and/or soil and graded over. The remaining facility will have been backfilled to prevent future subsidence and restored to a state suitable for industrial use.

Upon completed site clearance, and confirmation that all decommissioning tasks have been completed, the licensee will submit the application to delicense the site. At this point, it will no longer be subject to regulatory control measures.

21.7 Conclusions

The BWRX-300 design accounts for the decommissioning process, incorporating best practice and allowing for the development of new technologies. A provisional decommissioning timeline has been provided, covering the potential for both immediate and deferred strategies and indicating the likely milestones encountered as part of the process.

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

The responsibility for planning decommissioning will lie with the future site licensee. However, at this stage, a viable case has been made that this plan can be implemented, based on the BWRX-300 design and worldwide OPEX.

Table 21-1: International BWR Decommissioning Experience

BWR	Shutdown Date	Decommissioning Progress
Vermont Yankee (U.S.) (Reference 21-42) (Reference 21-17)	December 2014	Fuel removed from reactor January 2015 and all SF stored in on-site dry storage facility. Entered accelerated decommissioning with projected finish 2026. Segmentation, packaging, and disposal of Reactor Pressure Vessel (RPV) and reactor internals completed in late 2022.
Pilgrim (U.S.) (Reference 21-43) (Reference 21-44)	May 2019	Fuel removed and placed in (Fuel Pool) FP in June 2019. Undergoing active decommissioning, with estimating completion by 2063. SF was held in FP longer than anticipated following lack of GDF construction; all SF housed in the FP, which was beyond initial design considerations. Loading into dry storage casks and movement to Spent Fuel Storage Facility (SFSF) completed in 2021.
Oyster Creek (U.S.) (Reference 21-45)	September 2018	Fuel removed from reactor September 2018. Active decommissioning – dismantling and demolition underway. Partial site release scheduled for 2029 (excluding SFSF), license termination by 2035.
Duane Arnold Energy Centre (U.S.) (Reference 21-46)	August 2020	Fuel removed from reactor October 2020, and SF placed in on-site dry storage facility by end of April 2022. Facility under ‘safe storage’ until 2075, when full decommissioning will commence.
Caorso (Italy) (Reference 21-47) (Reference 21-48)	July 1990	Site under safe storage, with ongoing decommissioning processes since 2014. Dismantling has begun (off-gas building and turbine dismantled), and a waste route is being built between the RB and the TB. SF shipped to France for reprocessing. Expected completion by 2031.
Isar-1 (Germany) (Reference 21-49)	August 2011	Fuel removal complete and SF moved from FP into on-site dry storage. In dismantling phase of active decommissioning (prompt decommissioning employed), with license termination expected in 2036
Hamaoka 1 and 2 (Japan) (Reference 21-50)	January 2009	Decommissioning started November 2009. SF removed from FP by 2015. Dismantling of peripheral equipment undertaken using remotely operated large band saw.

BWR	Shutdown Date	Decommissioning Progress
Barsebäck 1 and 2 (Sweden) (Reference 21-51)	November 1999 and May 2005	All SF transported to Swedish interim storage facility (SKB) by 2006. RPV and adjacent systems partially decontaminated with high pressure hosing. All RPV internals segmented and packed by 2019 – stored in onsite interim storage facility. Dismantling is being undertaken as joint project with Oskarshamn reactors to improve work package efficiency and maximise control and safety. Demolition phase expected in early 2030s.

Legend:

“Power Reactor Sites – Vermont Yankee Nuclear Power Station,” (Reference 21-42)

“Vermont Yankee Decommissioning,” (Reference 21-17)

“Power Reactor Sites – Pilgrim Nuclear Power Stations,” (Reference 21-43)

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“Power Reactor Sites – Oyster Creek Nuclear Generating Stations,” (Reference 21-45)

“Power Reactor Sites – Duane Arnold Energy Center,” (Reference 21-46)

Caorso Nuclear Power Plant Decommissioning,” (Reference 21-47)

Caorso Nuclear Power Plant – Decommissioning Project,” (Reference 21-48)

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“The Power of Hitachi’s Large Band Saw Demonstrated at Hamaoka Nuclear Power Station Demolition Site,” (Reference 21-50)

“Dismantling and Demolition – How are Dismantle Barsebacksverket, Uniper,” (Reference 21-51)

Table 21-2: Estimate for the Decommissioning Activated Material Mass of Solid Waste (Reference 21-26)

Waste Category	Mass (tonnes)	Assumed Density (tonnes/m^3)	Volume of Waste (m^3)
HLW	900*	7.8 (value for steel)	115.4
ILW	650	1.165 (value for spent resin as bounding for wet solid waste)	557.9
LLW and VLLW	8,200	1.0 (generic value for heterogeneous wastes)	8,200

* Value does not include SF

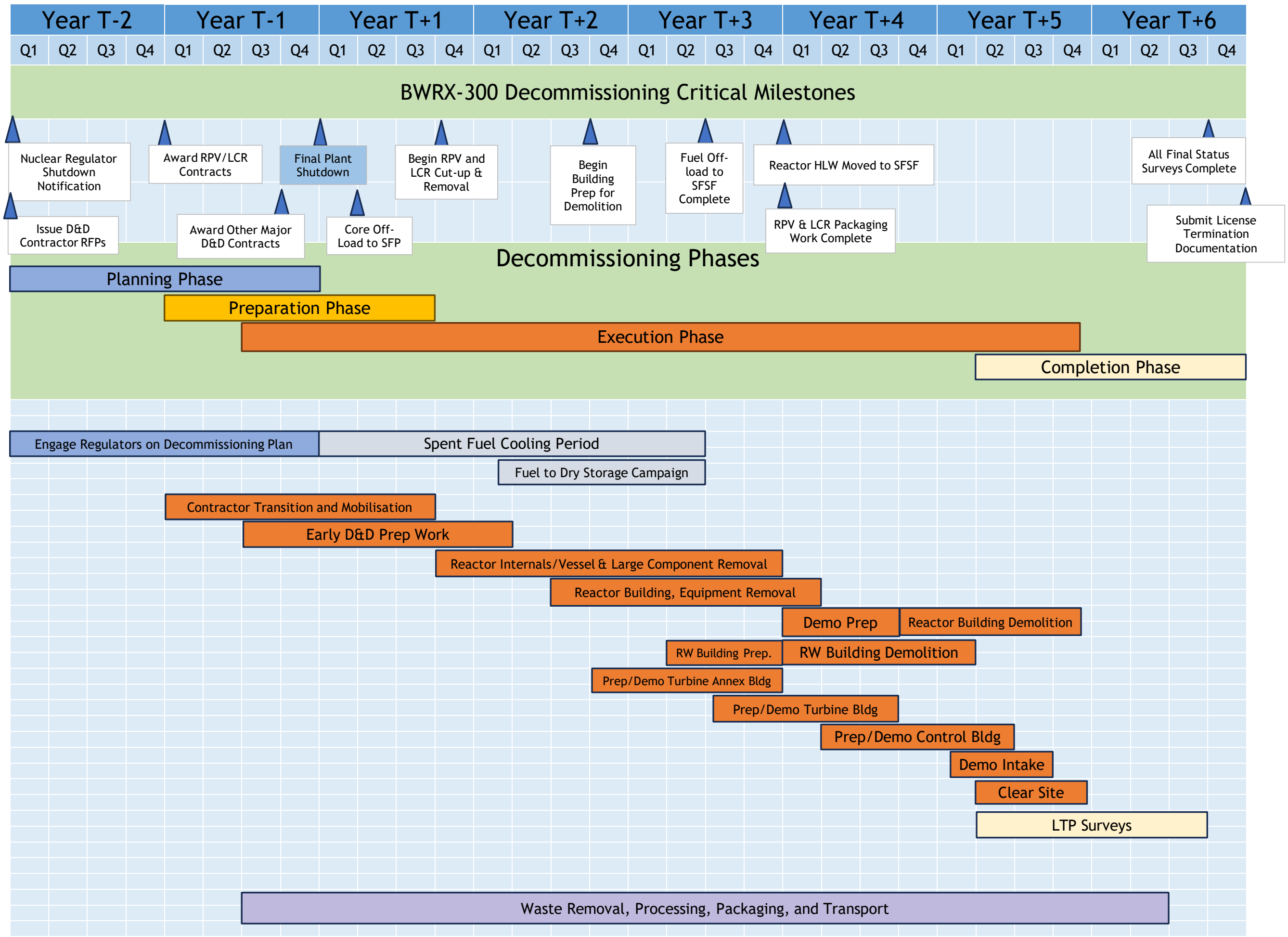

Figure 21-1: BWRX-300 Decommissioning Critical Milestones (Reference 21-26)

Legend:

LCR = Large Component Removal

D&D = Dismantling and Demolition

Bldg = Building

RW = Radwaste

Demo = Demolition

LTP = License Termination Plan

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APPENDIX A CLAIMS, ARGUMENTS AND EVIDENCE

The Office for Nuclear Regulation (ONR) SAPs 2014 (SC.4) (Reference 21-40) indicate the expectation that safety claims be clearly supported with appropriate arguments and evidence.

The claims relevant to Chapter 21 are:

2.1 The functions of systems and structures have been derived and substantiated using design safety principles and taking into account RGP and OPEX, and processes are in place to maintain these through-life.

2.4 Safety risks have been reduced ALARP.

To support these claims the following sub-claims will be addressed:

2.1.3 The system design has been undertaken in accordance with relevant design codes and standards RGP and design safety principles and taking account of OPEX to support reducing risks ALARP

2.1.6 The BWRX-300 will be designed so that it can be decommissioned safely, using current available technologies, and with minimal impact on the environment and people.

2.4.1 RGP has been taken into account across all disciplines.

2.4.2 OPEX and LfE has been taken into account across all disciplines.

Table A-1: CAE Route Map

Chapter 21 Claim	Chapter 21 Sub-claim	Chapter 21 Arguments	Evidence
2.1 The functions of systems and structures have been derived and substantiated using design safety principles and taking into account RGP and OPEX, and processes are in place to maintain these through-life.	2.1.3 The system design has been undertaken in accordance with relevant design codes and standards RGP and design safety principles and taking account of OPEX to support reducing risks ALARP	Design for decommissioning has accounted for international OPEX, which incorporates risk-reduction methods and RGP.	21.5.1 BWRX-300 Design Elements to Facilitate Decommissioning
		Steps will be taken to minimise radioactive contamination ALARP.	21.5.2 Provisions for Safety During Decommissioning
	2.1.6 The BWRX will be designed so that it can be decommissioned safely, using current available technologies, and with minimal impact on the environment and people	BWRX-300 design is based on RP OPEX, BAT and ALARP principles, including with respect to decommissioning. This is aided by the modular design, material choices, SSCs design and continually evolving knowledge of international RGP of decontamination and dismantling methods.	21.5.1 BWRX-300 Design Elements to Facilitate Decommissioning Document “BWRX-300 Incorporation of Decommissioning in Design Considerations,” 006N8745 (Reference 21-26)
2.4 Safety risks have been reduced ALARP	2.4.1 RGP has been taken into account across all disciplines	Design for decommissioning is based on significant RGP, and the expectation is an evolving decommissioning plan will account for improvements in RGP by the time a decommissioning is underway.	21.5.1 BWRX-300 Design Elements to Facilitate Decommissioning 21.1.2 Operational Experience of Decommissioning Document “BWRX-300 Incorporation of Decommissioning in Design Considerations,” 006N8745 (Reference 21-26)
	2.4.2 OPEX and LfE has been taken into account across all disciplines	Planning and design for decommissioning is based on significant LfE and OPEX, both of which are expected to have expanded by decommissioning implementation for a BWRX-300 site.	21.1.2 Operational Experience of Decommissioning

APPENDIX B FORWARD ACTIONS

Table B-1: Forward Action Plan

Action ID	Finding	Forward Actions	Delivery Phase
PSR21-153	Enhanced estimates of the nature and volumes of radioactive and hazardous waste generating by decommissioning are yet to be established by GEH.	A more comprehensive breakdown of waste volumes will be calculated once the site is decided, with these estimates understood to evolve during operation, in-keeping with recording requirements.	For Site License Application
PSR21-154	Although a high-level overview exists, a complete decommissioning plan has not yet been produced for the BWRX-300 NPP.	Produce UK-specific Preliminary Decommissioning Plan for BWRX-300	For PCSR/PCER

APPENDIX C UNITED KINGDOM-SPECIFIC CONTEXT

C.1 UK Regulatory Environment

The ONR had established Safety Assessment Principles (SAPs) for Nuclear Facilities, to inform and direct specialist inspectors in the assessment of facility safety cases (Reference 21-40).

The NS-TAST-GD-026 ONR “Technical Assessment (TAG) Guide: Decommissioning,” (Reference 21-52)) supports the inspector in determining whether the principles laid out in the SAP for Nuclear Facilities have been applied.

For the Requesting Party, it acts as a source of guidance on decommissioning good practice, with compliance to TAG expectations indicating compliance with the legislation, license conditions and RGP described below.

The “Regulating Duties to Reduce Risks to ALARP” TAG (NS-TAST-GD-005) states that the IAEA Safety Standards (Reference 21-8) and Western European Nuclear Regulators Association “Safety Reference Levels for Existing Reactors,” (Reference 21-53) and Safety Objectives are considered RGP in the field of nuclear safety.

C.2 United Kingdom Legislation

Many aspects of U.K. legislation are relevant to NPP decommissioning. The key pieces are highlighted, but operators will comply with all legislation and regulations arising during decommissioning activity.

- Nuclear Installations Act 1965
- Energy Act 2013
- Health and Safety at Work Act 1974
- Ionising Radiations Regulations 2017
- Radiation (Emergency Preparednes and Public Information) Regulations 2019
- Construction (Design and Management) Regulations 2015
- Environmental Permitting Regulations 2016

C.3 Relevant United Kingdom License Conditions

The following License Conditions have been identified to be of particular relevance to decommissioning and end-of-life aspects (as listed in ONR TAG: Decommissioning):

- 4 Control of nuclear matter
- 6 Documents, records, authories and certificates
- 15 Periodic review
- 17 Management systems
- 18 Radiological protection
- 22 Modifications
- 23 Operating rules
- 25 Operational records
- 28 Examination, inspection, maintenance and testing
- 32 Accumulation of radioactive waste

- 33 Disposal of radioactive waste
- 34 Leakage and escape of radioactive material and radioactive waste
- 35 Decommissioning
- 35 Organisational capability

C.4 United Kingdom Decommissioning Design Principles

The ONR TAG decommissioning principles (Reference 21-52), were derived from the same source (IAEA) as the decommissioning design principles outlined in Section 21.3.1 of this chapter. As such, UK regulatory design for decommissioning enforces principles are well aligned with those discussed in that section.

With respect to decommissioning strategy, the UK government stance is that prompt decommissioning is preferred (mirroring that of the IAEA).

C.5 United Kingdom Operational Experience

There is currently no U.K. OPEX in decommissioning BWRs, however, there is significant national experience with other reactor types which can be drawn upon. The U.K. fleet of Advanced Gas-cooled Reactors (AGRs) are undergoing various stages of decommissioning activity, initially under a deferred decommissioning strategy. Utilising a safe enclosure approach (deferred dismantling for approximately 100 years) following fuel removal stemmed from a lack of options for safe graphite disposal and the benefits of allowing radioactive decay for reducing worker dose.

Due to ongoing assessment of the risk levels of these sites, the decommissioning strategy for the eleven U.K. Magnox reactor sites has moved to one of ‘continuous decommissioning’, whereby priority, earlier, decommissioning is planned for certain sites/facilities in accordance with risk, SMS/TS/A-SR/002/3, “Site Restoration: Magnox Reactor Dismantling Timing and Sequencing Strategy,” (Reference 21-54). The reduction of intolerable risks is prioritised, with activities for this end spanning across the sites where achievable and necessary, NDA, SG/2021/48, “Strategy” (Reference 21-55).

More modern designs bring with them a reduced need for extended decay of radiation levels, and there is now a shift to prompt decommissioning for newer U.K. NPPs.

An overview of completed and ongoing decommissioning activities in the U.K. is given in the table below.

The complexity and variation in decommissioning, decontamination and disposal needs arising from U.K. NPPs and experimental reactors has prompted extensive Research, Development, and Innovation on this topic, “NDA Areas of Research Interest,” (Reference 21-41). It is expected that by the time of BWRX-300 decommissioning, the knowledge base will have been further expanded and the future site licensee can employ techniques already utilised in U.K. decommissioning activities.

Table C-1: UK Decommissioning Activities

Site	Reactor Type	Shutdown Date	Decommissioning Status
Magnox Fleet (Reference 21-55)	Magnox	1989 - 2015	All sites have been defueled. Varying status across sites, with all held in a care and maintenance period, in combination with priority-driven active dismantling efforts. Currently, Trawsfynydd is the lead site for decommissioning during to significant degradation.
Dounreay Fast Reactor	Fast Reactor	March 1977	Defueling of reactor complete, with fuel transferred to Sellafield
Winfrith (Reference 21-56)	Prototype Steam Generating Heavy Water Reactor (SGHWR)	September 1990	Fuel sent to Sellafield. Ongoing decommissioning activities since 1991, including preparation for reactor core dismantling
Sellafield (Windscale) (Reference 21-57)	WAGR	April 1981	Reactor core dismantled on a ‘top-down’ basis. Delays caused by unsupported insulation collapsing during RPV removal, and an inappropriate ventilation system.
Dungeness B (Reference 21-58)	AGR	June 2021	Defueling phase underway
Hunterston B (Reference 21-59)	AGR	November 2021 and January 2022	Defueling phase underway
Hinkley Point B (Reference 21-59)	AGR	July 2022 and August 2022	Defueling phase awaited

Legend:

“Work Underway to Remove Nuclear Reactor Core in Dorset; (Reference 21-56)

“The Windscale Advanced Gas Cooled Reactor (WAGR) Decommissioning Project: A Close Out Report for WAGR Decommissioning Campaigns 1 to 10,” (Reference 21-57)

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