GE Hitachi Nuclear Energy

NEDO-34198
Revision B
July 2025

US Protective Marking: Non-Proprietary Information
UK Protective Marking: Not Protectively Marked

BWRX-300 UK Generic Design Assessment (GDA)
Chapter 26 - Interim Storage of Spent Fuel

INFORMATION NOTICE

This document does not contain proprietary information and carries the notations "US Protective Marking: Non-Proprietary Information" and "UK Protective Marking: Not Protectively Marked."

IMPORTANT NOTICE REGARDING CONTENTS OF THIS REPORT

Please Read Carefully

The design, engineering, and other information contained in this document is furnished for the purpose of obtaining the applicable Nuclear Regulatory Authority review and determination of acceptability for use for the BWRX-300 design and licensing basis information contained herein. The only undertakings of GEH with respect to information in this document are contained in the contracts between GEH and its customers or participating utilities, and nothing contained in this document shall be construed as changing those contracts. The use of this information by anyone for any purpose other than that for which it is intended is not authorized; and with respect to any unauthorized use, no representation or warranty is provided, nor any assumption of liability is to be inferred as to the completeness, accuracy, or usefulness of the information contained in this document. Furnishing this document does not convey any license, express or implied, to use any patented invention or any proprietary information of GEH, its customers or other third parties disclosed herein or any right to publish the document without prior written permission of GEH, its customers or other third parties.

UK SENSITIVE NUCLEAR INFORMATION AND US EXPORT

CONTROL INFORMATION

This document does not contain any UK Sensitive Nuclear Information (SNI) subject to protection from public disclosure as described in the Nuclear Industries Security Regulations (NISR) 2003, does not contain UK Export Controlled Information (ECI), and does not contain US Export Controlled Information (ECI) subject to the export control laws and regulations of the United States, including 10 CFR Part 810.

EXECUTIVE SUMMARY

BWRX-300 Generic Design Assessment (GDA) Preliminary Safety Report Chapter 26 presents the spent fuel (SF) management arrangements for the BWRX-300 to demonstrate that they can be developed to comply with relevant UK policy, legislation, regulations, and regulatory guidance.

Considering its place within Step 2 of a GDA, the aim is that adequate demonstration is provided that: the management of SF and irradiated in-core components (IICC) has been considered; a viable future plan for the long-term storage and disposal of the SF could be devised; and no fundamental impediments to its management have been identified.

Relevant Good Practice and UK operational experience suggests a dry cask storage approach is the most suitable for implementation in the BWRX-300 design. A generic viable option for this form of interim storage of High Heat Generating Waste is evaluated, although site-specific constraints will influence the final implementation of this design by the future licensee.

A description of the management of SF and IICC is provided, covering a description of the arisings, fuel pool operations, dry cask storage facility operations, and a high-level overview of disposal. A viable blueprint for future management, storage, and disposal of BWRX-300 SF and IICC is provided. A demonstration of disposability has been undertaken, ensuring Nuclear Waste Services is able to provide the required disposability Expert View to help support the BWRX-300 GDA.

At this stage of GDA, although subject to the outcome of the Nuclear Waste Services' Expert View, no significant obstacles have been identified and all likely issues have been identified, mitigated, or added to the Forward Action Plan for resolution in a future Step 3.

Claims and arguments relevant to GDA Step 2 objectives and scope are summarised in Appendix A. Appendix B provides a Forward Action Plan. Appendix C presents the UK legislative context.

ACRONYMS AND ABBREVIATIONS

Acronym	Explanation
ABWR	Advanced Boiling Water Reactor
AGR	Advanced Gas-cooled Reactor
ALARA	As Low As Reasonably Achievable
ALARP	As Low As Reasonably Practicable
BAT	Best Available Technique
BWR	Boiling Water Reactor
DFRP	Dry Fuel Repackaging Plant
DFS	Dry Fuel Store
FP	Fuel Pool
GDA	Generic Design Assessment
GDF	Geological Disposal Facility
GE14	General Electric 14 (fuel design)
GEH	GE Hitachi Nuclear Energy
GNF	Global Nuclear Fuels
GNF2	Global Nuclear Fuels 2 (fuel design)
GT	Gamma Thermometer
HHGW	High Heat Generating Waste
HLW	High Level Waste
IAEA	International Atomic Energy Agency
IICC	Irradiated In-Core Components
ILW	Intermediate Level Waste
KKM	Mühleberg Nuclear Power Plant (German: Kernkraftwerk Mühleberg)
LfE	Learning from Experience
LLW	Low Level Waste
LPRM	Local Power Range Monitor
LWR	Light Water Reactor
MPC	Multi-purpose Canister
NPP	Nuclear Power Plant
NRC	Nuclear Regulatory Commission (US)
NWS	Nuclear Waste Services
ONR	Office for Nuclear Regulation
OPEX	Operational Experience
PRA	Probabilistic Risk Assessment
PSR	Preliminary Safety Report
PWR	Pressurised Water Reactor

Acronym	Explanation
RB	Reactor Building
R&D	Research and Development
RGP	Relevant Good Practice
RP	Requesting Party
RPV	Reactor Pressure Vessel
SF	Spent Fuel
SZB	Sizewell B
TAG	Technical Assessment Guide
UK	United Kingdom
UKABWR	United Kingdom Advanced Boiling Water Reactor
WAC	Waste Acceptance Criteria
WPS	Waste Package Specification
WRNM	Wide Range Neutron Monitor

SYMBOLS

Symbol	Definition
GWd/MTU	Gigawatt days per metric tonne of uranium
k-eff	Neutron Multiplication Factor
kg U	Kilograms of uranium

TABLE OF CONTENTS

LIST OF TABLES

LIST OF FIGURES

REVISION SUMMARY

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

26 INTERIM STORAGE OF SPENT FUEL

Purpose

The purpose of this chapter is to present information on the Spent Fuel (SF) management arrangements for the BWRX-300 and to demonstrate that they can be developed to comply with relevant UK policy, legislation, regulations, and regulatory guidance.

The aim is to provide an adequate demonstration that: the management of SF and Irradiated In-core Components (IICCs) has been considered; a viable future plan for the long-term storage and disposal of the SF could be devised; and no fundamental impediments to its management have been identified.

Scope

This Preliminary Safety Report (PSR) chapter will present an assessment of the management, storage, and potential for disposal of SF and IICC.

A detailed worked example of each option available for storage, repackaging, and disposal of SF arising from the BWRX-300 is not provided in the PSR.

Information on the proposed approach to SF storage, repackaging, and Geological Disposal Facility (GDF) disposal will be sufficient to provide confidence that suitable SF storage, repackaging and disposal processes and techniques are available to safely manage the SF.

An outline of key SF life cycle activities and timelines is provided in order to demonstrate that there are no significant obstacles to deployment of the BWRX-300 design. Evidence of worldwide Operational Experience (OPEX) of Boiling Water Reactor (BWR) SF management shows that it is possible to develop a plan that allows its safe management.

This is considered appropriate, based on the following justifications:

- The details of SF storage and repackaging approaches, including the design of a Dry Fuel Store (DFS) and future repackaging plant, will depend significantly on site-specific issues and will need to be underpinned with detailed Best Available Technique (BAT) assessments. In the absence of this site-specific information, it is therefore not considered appropriate to develop this level of detail at this stage.
- OPEX exists in the UK to enable a relevant, UK-specific, worked-up example to be provided to demonstrate viability of the approach.
- As no Waste Acceptance Criteria (WAC) and no final Waste Package Specification (WPS) appropriate for SF disposal exists for SF to be accepted into a national UK GDF, it would be inappropriate to impose any such requirements at this stage. A high-level demonstration of approach, in line with Relevant Good Practice (RGP), is considered sufficient at this stage.
- None of the demonstrated approaches in these submissions will preclude or foreclose any other approaches and will solely demonstrate existence of a management approach and its viability.

Background and assumptions

The following points act as a key basis for the level of detail provided on the design aspects relevant to SF and SF storage presented in this chapter:

- WAC and WPS not yet established at time of assessment
- One single, worked-up example, consistent with current UK SF management and aligned with RGP will be sufficient to demonstrate viability of approach

- The approach will utilise dry cask storage as the preferred approach, drawing on UK OPEX for SF management, storage, repackaging and disposal
- IICC will be considered as part of the SF chapter. Due to the levels of activation anticipated it is assumed that these wastes will emit significant levels of radiogenic heat (>2 kW/m^3) and will therefore be classified as High-Level Waste (HLW) on production, "Basic Principles of Radioactive Wate Management," (Reference 26-1). In line with the management strategy adopted for the UK Advanced Boiling Water Reactor (UKABWR), it is assumed that these wastes will be packaged and stored to benefit from radioactive decay until they meet the Intermediate Level Waste (ILW) classification (<2 kW/m^3) (Reference 26-1). Interim storage is assumed to be in dry casks, as for SF. This approach is considered appropriate as the wastes will be routed through the Fuel Pool (FP) on removal from the reactor, and therefore align with the provisions made in the FP for cask handling and packaging. Dry casks containing HLW are assumed to be co-stored with SF casks in the DFS. Once the waste has decayed to ILW levels it will be recovered and repackaged as ILW.
- None of the approaches, techniques, or examples described will preclude or foreclose any other approaches. They exist solely to demonstrate that a viable approach to the future development of SF management, storage, repackaging, and disposal plans exists.
- The level of detail presented in the chapter shall be commensurate with the level of maturity of the relevant storage, repackaging, and disposal concepts. This is considered appropriate in the absence of site-specific information, cask vendor selection, and BAT assessments to further underpin them.

Document structure

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

- Section 26.1 – Description of Spent Fuel and Irradiated In-Core Components: This section provides an overview of the nature of SF and IICC produced during BWRX-300 operation.
- Section 26.2 – Spent Fuel Management: This section provides an overview of the approach to SF management employed at a BWRX-300 site prior to interim storage.
- Section 26.3 – Spent Fuel Storage: This section provides an overview of the approach to dry cask storage, alongside reference to relevant UK OPEX.
- Section 26.4 – Spent Fuel Disposal. This section provides a high-level overview of the envisaged use of a GDF for SF disposal, aligning with assumed GDF future availability.
- Section 26.5 – Conclusions: This section is a summary of the key content of this PSR chapter.
- Section 26.6 – References: This section provides a list of supporting documents referenced in this chapter.

Interfacing systems

The key interfacing systems to be considered in this chapter are:

- Fuel pool
- Equipment storage pool
- Gantry/fuel crane
- Casking area

- Cask drying equipment

Interfaces with other chapters

The following PSR chapters interface with this topic:

- Chapter 1: Introduction and General Considerations, NEDO-34163, “BWRX-300 UK GDA Chapter 1: Introduction and Overview,” (Reference 26-2)
- Chapter 3: Safety Objectives and Design Rules for SSCs, NEDO-34164, “BWRX-300 UK GDA Chapter 3: Safety Objectives and Design Rules,” (Reference 26-3)
- Chapter 4: Reactor (Fuel and Core), NEDC-34166P, “BWRX-300 UK GDA Chapter 4: Reactor (Fuel and Core),” (Reference 26-4)
- Chapter 11: Management of Radioactive Waste, NEDO-34166, “BWRX-300 UK GDA Chapter 11: Management of Radioactive Waste,” (Reference 26-5)
- Chapter 25: Security Annex, NEDO-34197, “BWRX-300 UK GDA Chapter 25: Security Annex,” (Reference 26-6)
- Chapter 28: Safeguards Annex, NEDO-34200, “BWRX-300 UK GDA Chapter 28: Safeguards Annex,” (Reference 26-7)

The document also interfaces with the Preliminary Environmental Report, in particular the following chapters:

- Chapter E1: Introduction, NEDO-34218, “BWRX-300 UK GDA Chapter E1: “Introduction,” (Reference 26-8)
- Chapter E4: Information About the Design, NEDO-34221, “BWRX-300 UK GDA Chapter E4: “Information About the Design,” (Reference 26-9)
- Chapter E5: Radioactive Waste Management Arrangements, NEDO-34222, “BWRX-300 UK GDA Chapter E5: “Radioactive Waste Management Arrangements,” (Reference 26-10)
- Chapter E7: Radioactive Discharges, NEDO-3422, “BWRX-300 UK GDA Chapter E7: “Radioactive Discharges,” (Reference 26-11)

Further information pertinent to this chapter is presented in the Appendices. Appendix A details the Claims, Arguments and Evidence, Appendix B provides details of the Forward Actions, and Appendix C describes UK-Specific Context, including an overview of the UK legislation, standards, and guidance relevant to DFS.

26.1 Description of Spent Fuel and Irradiated In-Core Components

26.1.1 Previously Assessed Fuel – Advanced Boiling Water Reactor GE14

The Advanced Boiling Water Reactor (ABWR) GE14 fuel is particularly instructive as it has seen extensive use since its introduction in the 1990s and has already undergone assessment by the Office for Nuclear Regulation (ONR) as part of the Hitachi-GE UKABWR Generic Design Assessment (GDA), "Step 4 Assessment of Fuel & Core Design for the UK Advanced Boiling Water Reactor," (Reference 26-12). Details of the GE14 fuel design can be found in Hitachi-GE document GA91-9901-0046-00001, "UK ABWR Generic Design Assessment: Preliminary Safety Report on Reactor Core and Fuels," (Reference 26-13).

The submitted safety case made for use of GE14 fuel in the UK (see PSR Chapter 11: Reactor Core (Reference 26-14) and PSR Chapter 19: Fuel Storage and Handling (Reference 26-15), of the UKABWR Pre-construction Safety Report (PCSR)) highlighted the high level of international experience (exceeding three million fuel rods) with this fuel type and the ongoing evolution of its design.

During the GDA process, it was noted that the fuel performance and design limits were within ONR's regulatory experience and demonstrated a high quality and proven design.

26.1.1 BWRX-300

26.1.1.1 GNF2 Fuel

GNF2 fuel is proposed for use in 006N1887, "BWRX-300 Fuel Design and Qualification," and represents an evolution of the GE14 design, with over 22,000 bundles operating in BWRs (correct as of 2020 (Reference 26-16)). Differences between the GE14 and GNF2 designs are detailed in Table 2-1 of Reference 26-16. The GNF2 fuel assembly selected for use in the BWRX-300 closely resembles the GE14 fuel currently employed in BWRs present in the US, Spain, Switzerland, Germany, Sweden, and Finland.

GNF2 fuel assemblies (see Figure 26-1) consist of a fuel bundle (containing fuel rods, water rods, spacers, and tie plates) and encompassing channel (made of Zircaloy). The fuel bundle comprises 92 fuel rods and two central water rods, arranged in a 10x10 array. The fuel rods contain high density ceramic UO_2 or $(U,Gd)O_2$ undished-chamfered fuel pellets stacked within Zircaloy-2 cladding (with a thin zirconium inner barrier liner), NEDC-33941P, "GNF2 "Fuel Assembly Thermal-Mechanical Design Report," (Reference 26-17).

Full technical details of the fuel assembly are given in NEDC-34159P, "BWRX-300 UK GDA Fuel Summary Report," (Reference 26-18). Within the rest of this chapter, the terms fuel 'bundle' and 'assembly' are used interchangeably when referencing BWRX-300 fuel operations.

For GDA purposes, it is assumed that the BWRX-300 will have an average core discharge burn-up of 49.6 GWd/MTU and operate on a 12-month fuel cycle with 32 bundles replaced during each refueling outage, "BWRX-300 Plant Performance Envelope," (Reference 26-19). For a 60-year operating life, this would equate to an initial core load of 240 bundles, plus 59 reloads of 32 bundles, giving rise to a total of 2,128 SF bundles. Given a 24-month cycle would yield more SF bundles, for the purpose of conservatism a similar calculation has been conducted – in this case, there would be 2,368 bundles (based on 72 spent bundles generated every 2 years) (Reference 26-19). The spent fuel bundle numbers per cycle length information is summarised in Table 26-1.

26.1.1.2 Fuel Reliability

GNF2 (10x10) fuel experience shows a relative failed fuel rate[1] of 2.1 bundles/1000 operated, up until September 2023, NEDC-33415P, "Nordic GNF2 Operational Experience Update, Supplement 1," (Reference 26-20). Based on this failure rate and using the 12-month fuel cycle estimate of 2,128 SF bundles produced over the 60-year BWRX-300 operational lifetime, there could be, up to, an estimated five incidences of fuel failure during operation. There is no known instance of a BWR fuel failure occurring in a FP (while cold and sub-critical) over the course of 40 years of OPEX of BWR fuel storage in 006N5399, "BWRX-300 Irradiated Fuel Management Plan" (Reference 26-21).

There are multiple mechanisms associated with the in-operation failure of fuel: manufacturing defects; crud/corrosion defects; pellet-cladding interaction; and debris fretting. There are no reported cases of structural failure of GNF2 fuel resulting in debris generation. This failure mode is not considered credible for GNF2 fuel or in the BWRX-300 and therefore, it is not further discussed here. Comprehensive details on fuel failure mechanisms and the incidence levels in GNF2 are given in Appendix A of NEDC-34159P (Reference 26-18). Further, in-depth discussion of failure by debris fretting is provided in Section 4.2.16 of PSR Chapter 4: Reactor (Fuel and Core) (Reference 26-4).

Inspection experience with GNF 10x10 fuel designs is extensive, with all 10x10 fuel failure events having been investigated in poolside examinations. A summary of those GNF2 fuel inspections completed prior to September 2023 is shown in Reference 26-18 and Table A2 therein. This inspection campaign is on-going, with GNF continually tracking performance for the current generation of fuels, including leaker investigations and repairs.

The reliability of GNF2 fuel during interim storage is discussed in Section 26.3.6 of this chapter.

26.1.2 Irradiated In-Core Components

Alongside SF, IICC are expected to initially meet the HLW classification due to significant radiogenic heat production. IICC includes control rods, core monitoring instrumentation, fuel channels, guide tubes, and fuel supports.

In the BWRXR-300 design, the control rods are cruciform blades, comprised of laser-welded stainless-steel tubes attached to a central cruciform (design name: Ultra™), "ULTRA Control Rod Blades Fact Sheet," (Reference 26-22). There are 57 control blades within the core, 005N9751, "BWRX-300 General Description," (Reference 26-23). The standard tubes used will contain B_4C powder, with the potential for the leading edge (high duty regions) of blades within the control cells to use Hf rods, depending on fuel cycle requirements, NEDE-33284, "Marathon-Ultra Control Rod Assembly," (Reference 26-24).

Burnable poisons within the core include B_4C and Hf from the control rods, alongside Gd present within select fuel pellets (as Gd_2O_3 in solid solution with UO_2). It is expected that, as part of viable fuel shuffling, across the multiple cycles prior to fuel discharge, all the Gd present in fuel rods is burnt out. This is substantiated by hot excess reactivity calculations and described in Section 4.2.1.1 of PSR Chapter 4: Reactor (Fuel and Core) (Reference 26-4).

The core also contains monitoring apparatus; namely, the Wide Range Neutron Monitor (WRNM), Local Power Range Monitor (LPRM) and Gamma Thermometers (GTs). There are 13 vertical LPRM/GT strings (four LPRM and eight GTs on each) and 10 WRNM fixed neutron detectors in the core. Full details on the BWRX-300 in-core monitoring equipment are given in Chapter 4 and Chapter 7 of the PSR. It is anticipated that the LPRMs will be broadly in line with those employed in BWR4-type reactor cores. The GTs are K-type thermocouples, which

[1] Fuel failures indicated here are in-service failures and are not considered Anticipated Operational Occurrences (AOOs). Fuel failures are not quantified as AOOs as they are the consequence of faults, rather than faults themselves (for example fuel clad fretting as a result of particle ingress into the fuel bundle).

have an extensive history of in-core monitoring, comprising of a combination of Ni-Cr and Ni-Al alloys, with a Al_2O_3 coating and stainless-steel sheath, "Gamma Thermometer Datasheet," (Reference 26-25). Further details on the BWRX-300 core monitoring instrumentation can be found in Subsections 7.3.2 and 7.3.3 of NEDO-34169, "BWRX-300 PSR Chapter 7: Instrumentation and Control," (Reference 26-26).

26.2 Spent Fuel Management

26.2.1 Spent Fuel Life Cycle

The life cycle of SF will cover the following phases of operation: defueling, transfer to the FP, short-term wet storage in the FP, casking, delivery, transport across site, long-term dry storage, repackaging, and disposal.

26.2.2 Description of Relevant Fuel Route Operations

The Refuelling and Servicing Equipment System Design Description, 006N5377, "System Design Description: BWRX-300 Refueling and Servicing Equipment," (Reference 26-27), details the steps undertaken by reactor operators and provides the apparatus for the removal and storage of SF assemblies alongside the storage of new fuel. A more detailed description of the BWRX-300 fuel storage and handling system can be found in Section 9A.1 of NEDO-34171, "BWRX-300 PSR Chapter 9A: Auxiliary Systems," (Reference 26-28). Those aspects of relevance to SF and IICC management and storage are addressed in this section, and they have been designed to broadly align with UK expectations.

26.2.3 Spent Fuel and IICC Fuel Pool Operations

The Reactor Building (RB) FP contains the FP, equipment pool, reactor cavity pool, new inspection stand, channel handling boom, and space for receiving/handling new fuel 008N0988, "BWRX-300 Power Block General Arrangement Drawings," (Reference 26-29). In the unlikely event a Fuel Handling Accident occurs, visual inspections are undertaken to establish if there is any structural damage (Reference 26-21). If any damage is identified, the bundle will be managed according to damage severity. Failed fuel is treated according to the description given in Section 26.2.5.

The FP is arranged to house a combination of new fuel, SF, and IICC, and organised in a manner to allow for safe storage and operations in the FP. Items segregated include control rods, fuel supports, fuel channels, guide tubes, water level instruments, startup sources, GTs, LPRMs, and WRNM dry tubes (Reference 26-27). The FP is adjacent to the reactor cavity, with isolation possible via a removable gate. The deep pit in the FP provides storage for SF and space for loading a SF cask. There are two fuel preparation machines within the FP and the refuelling platform spans the FP. The refuelling platform is used to transfer fuel underwater between fuel storage racks and the reactor core to ensure operator shielding.

26.2.3.1 Fuel Storage Racks

Fuel storage racks housed within the pool can contain a mixture of both new and SF, with the aim of providing safe, effective, and traceable storage. They will have a minimum storage capacity of approximately 600 fuel assemblies. The FP thermal management was designed with safety margin, with bounding thermal calculations demonstrating that up to 660 fuel assemblies could be safely accommodated, 007N0022, "BWRX-300 Spent Fuel Pool Decay Heat," (Reference 26-30). Water shall cover the entire active fuel height (normally up to more than 3 metres above the top of active fuel), with natural convection allowed through the rack and fuel to remove decay heat under both normal and abnormal conditions. The Fuel Pool Cooling and Cleanup System is responsible for providing continuous cooling of the water volume in the fuel pool to remove decay heat from SF, such that SF is kept cool and submerged until relocated for permanent storage, 006N7941, "System Design Description: BWRX-300 Fuel Pool Cooling and Cleanup," (Reference 26-31). There is no known incident of BWR fuel failure occurring during FP storage following reactor discharge (Reference 26-21).

The fuel racks are designed to hold an entire reload of fresh fuel and up to 8 years of SF. The racks are sized for 36 fresh bundles, 240 off-loaded bundles, 40 first cycle bundles, and seven sets of 36 bundles, for a total of 568 (Reference 26-21). This accounts for 8 years of operation, plus new fuel and one full core offload of fuel assemblies.

26.2.3.2 Control Rod and Defective Fuel Storage Racks

The control rod and defective fuel storage rack is a double row rack which can accommodate control rods, defective fuel rod storage containers, and control rod guide tubes. A total of 10 cylinders can be stored, eight in the double row rack, with two centred at each end of the rack (Reference 26-27).

26.2.4 Spent Fuel and IICC Casking

The multiple components of the chosen dry storage system, such as the Multi-Purpose Canisters (MPCs), on-site transfer overpacks, and storage overpacks, and supporting systems, such as welding, fuel and cask drying, and cooling and monitoring systems, are set up on-site.

Upon sufficient cooling in the FP, SF is loaded into canisters. SF from the reactor is loaded into the at-reactor FP for an initial cooling period of approximately 7-8 years. The implemented cooling time will be a decision for the licensee and will likely be dictated by the thermal limits of the selected casking technology. A casking campaign takes place while the reactor is online, with the frequency of campaigns determined by the future site licensee and dependence on the cask vendor selected (Reference 26-21).

As the pool reaches capacity, or as part of the SF management policy after this initial cooling period, the fuel assemblies are to be retrieved, dried, and eventually emplaced in dry storage casks. A typical process for this is described below. Nonetheless, the general cask preparation process (casking, draining, welding, moisture removal, and helium backfill) is consistent across operating commercial designs.

This casking process takes place within the pool, where an empty canister is moved into the designated area (i.e., the SF cask pit). The selected assemblies are moved under water from the SF racks into the canister. Once filled, a video recording of the canister serial number is obtained, and a cap placed and bolted onto the canister. The SF canister can then be moved to the cask pad on the refuelling floor, where it is evacuated of water, vacuum dried, and a cap is welded on using remote welding techniques to minimise worker dose. The canister is backfilled and slightly pressurised with an inert gas. The sealed canister is leak tested and a non-destructive examination of the weld is undertaken (Reference 26-21).

The sealed and shielded canister is moved to a location for decontamination until external surfaces almost reach free release status. It is then able to be loaded into a transport shielded cask for further transport and storage. Casking order is determined by the age of the fuel, exposure, and decay heat.

IICC are also stored in the FP to manage decay heat. An instrument handling tool is used for removing and installing the fixed in-core detector assemblies, in addition to the WRNM sources and dry tubes. The instrument handling tool consists of the frame, air cylinder, and slide mechanism. The tool is handled by one of the refuelling platform auxiliary hoists and the terminal stud of the hoist cable threads into the stud on top of the handling tool (Reference 26-27).

There is an option, based on OPEX, for separate treatment of LPRM waste, whereby the ‘hot’ and ‘cold’ sections of the mounting tube are separated in the FP gate area after removal from the core. This allows the lower, ‘cold’ end to be further sectioned and disposed of as Low-Level Waste (LLW), whilst the ‘hot’ end is isolated and stored in a FP rack to allow cooling prior to casking. The vertical instrument cutter tool is used to cut in-core instruments and dry tubes, which are then placed into a holding can. The tool is placed in a vacated cell locations within the core, increasing efficiency and allowing immediate cutting after instrument/tube removal. The can is moved to the FP after cutting (Reference 26-27). IICC are treated as distinct waste forms to SF and thus will be casked separately, to allow for its future management and disposal as ILW.

26.2.5 Management of Failed Fuel Bundles

Failed bundles are usually first identified during operation via the detection of increased coolant radiological activity. Gamma radiation levels are measured in the plant's Off-gas System, with spikes indicating potential fuel failures (Reference 26-21). Routine sampling for isotopic concentrations is also used.

The location of failed bundles within the reactor is determined using power suppression testing (i.e., flux tilting). The assembly with a leaking fuel rod is established by a sipping technique and placed in a fuel preparation machine elevator in the FP for a more detailed examination and, if required, repair. The general inspection and repair sequence is to separate the channel from the fuel bundle, identify the problem, disassemble the bundle, and replace failed or damaged components as required. Upon shutdown, the failed fuel bundle is removed from the core following the conventual refuelling procedure. Placement of the bundle into the FP allows visual inspection to identify the failed rod and determine cause of failure, NEDC-33940P, "BWRX-300 GNF2 Fuel Assembly: Mechanical Design Report," (Reference 26-32).

Visual inspections are performed with a periscope or underwater colour television camera, permitting an assessment of the bundle integrity and check for the presence of debris. If it is not obvious from the visual inspection, fuel assemblies are checked to establish which rods have failed. To do this the upper tie plate is removed and fuel rods are individually withdrawn. They are passed through a flaw detection eddy current/ultrasonic testing device that interrogates the fuel rods for cladding defects and water on the inside surface. A visual inspection of the rod identified as failed characterises the damaged region and the cause of failure. (Reference 26-32)

Note that poolside inspection of irradiated BWR fuel is safely accomplished by requiring approximately 2 m of water between the top of the fuel bundle and the surface of the water in the FP. The fuel prep elevator is equipped with a chain stop to assure the required water shielding is maintained while a bundle is undergoing inspection.

If needed, there is an option of removing damaged fuel rods for replacement (by either new fuel or dummy rods) and separate storage. General practice, however, is that damaged bundles are stored in normal SF rack locations in the FP (Section 26.2.3), as leakage and further deterioration when not at power is understood to be negligible, "Behaviour of Spent Power Reactor Fuel during Storage," (Reference 26-33). Failed fuel bundles can stay in the FP indefinitely without any further special controls (Reference 26-21). SF casks may contain several damaged fuel bundles; however, the specific loading is dependent on the cask design and the vendor specifications.

26.2.6 Spent Fuel and IICC Delivery

Canister movement from the FP to the DFS is performed under guidance of the Health Physics and Security functions. The RB's polar crane is used to lift the cask to the refuel floor and subsequently to the truck bay. There the transfer cask is picked up by the hauler and moved to the DFS.

The lift height and maximum drop height for a loaded cask handled by the RB polar crane are described in detail within Section 15.7.12 in PSR Chapter 15 (Reference 26-34). These are based on an assumed set of cask dimensions and weights, during normal fuel route operations. Laden casks have a significantly reduced maximum drop height due to the subterranean nature of the RB design.

26.2.7 Transport to Dry Storage Facility

A detailed description of transport to the DFS is not provided, as that will depend on both site-specific matters and cask vendor selection being completed, but an indicative outline is given below. The prepared transfer cask is moved from the RB to the DFS building. This requires the use of a transporter (to move the transfer cask into and out of the RB) and Cask

Transporter (to transfer canister/cask assemblies from the transporter, once outside of RB, to the interim storage building). Cask transporters are already in use in the UK in Sizewell B (SZB), as well as multiple international Nuclear Power Plants (NPPs), and are assumed to be the primary transport mechanism for road-based movement of casks into the interim storage building. Upon arrival at the DFS, the MPC is then removed from the transfer cask and inserted into the storage overpack for long-term storage.

26.3 Spent Fuel Storage

26.3.1 Spent Fuel and IICC Storage Approach

This section will outline the long-term dry storage approach, deploying "business as usual" arguments in line with RGP and UK-centric OPEX. It will align with current Light Water Reactor (LWR) SF management arrangements at other UK NPPs, demonstrating that systems can be designed and operated using BAT, such as to maintain risks As Low As Reasonably Practicable (ALARP) and fulfil optimisation requirements As Low As Reasonably Achievable (ALARA).

A detailed description of transport across site is not provided, as that is both a site-specific matter and dependent on cask vendor selection, but a general outline is given. Detailed fault studies associated with the operation of the DFS are also not provided as they will depend on the cask storage technology and vendor selected, but a high-level outline is given here and in Appendix C of Chapter E5: Radioactive Waste Management Arrangements, NEDO-34222, "BWRX-300 UK GDA Chapter E5: "Radioactive Waste Management Arrangements," (Reference 26-10).

The preferred storage approach assumed for this assessment is dry storage of the SF and IICC in dry casks. The IICC will be segregated in dry casks, separate from the stored SF, and repackaged once the radiogenic heat production has decreased to levels below which the IICC can be disposed as ILW.

The approach outlined below will be manufacturer and technology-agnostic but will aim to demonstrate the viability of the approach. In the absence of site-specific information and cask vendor selection, required to underpin the future BAT assessment, a high-level approach is described. It is in broad agreement with UK expectations and approaches already undertaken in the UK. This is commensurate with the level of detail available to underpin the assessment.

The FP is able to accommodate 8 years' worth of SF assemblies, therefore a DFS will not need to accommodate any SF until the ninth year after initial reactor operations.

26.3.2 Relevant United Kingdom Spent Fuel Storage Experience

Modern UK RGP for the management and storage of SF is dry cask storage, with high integrity storage casks stored indoors in a secured containment facility to further ensure security and atmospheric control. This is evidenced by recent applications for DFS at SZB and Hinkley Point C. These most recent applications for DFS permits and their construction demonstrate that this approach is in line with current RGP and is in fact the preferred approach for the long-term storage of SF and IICC.

UK experience with dry storage of spent nuclear fuel dates back to the Wylfa dry storage vault for Magnox SF in the 1970s, "Written evidence submitted by Magnox Limited (FNP 53)," (Reference 26-35). The Wylfa dry store, constructed by GEC ALSTHOM, was the design on which further dry storage vaults were based, including the Independent SF Storage Installation at Fort St. Vrain in the US and the DFSs at Paks NPP in Hungary. After successful operation over the 45-year lifetime of the reactor, all fuel was moved to Sellafield for reprocessing, Wylfa Site: Environmental Management Plan," (Reference 26-36). Extensive Research and Development (R&D) was carried out in this period on dry storage of non-dismantled Advanced Gas-cooled Reactor (AGR) fuel, "AGR Fuel Storage Atmospheres," (Reference 26-37). Currently, storage of dismantled AGR fuel pins is a topic of intense R&D in the UK, as a contingency to the current SF pond storage of the AGR fuel inventory.

The UK has operated a dry storage cask facility since 2017 at the SZB power plant for storage of SF from its Pressurised Water Reactor (PWR). Dry storage casks, consisting of stainless steel 316L MPCs loaded into concrete-metal overpacks, are housed in a warehouse-like building, capable of holding the SF arisings from the reactor's operational lifetime. The cask system has a design lifetime of 100 years. Each canister used is capable of accommodating up to 24 fuel assemblies. Due to corrosion concerns, an updated canister with double steel walls was used. The SF is expected to be stored in this arrangement until a GDF becomes available, “Assessment of Sizewell B Dry Fuel Store Post Operational Safety Case NP/SC 7575, EC 338898-1,” (Reference 26-38), and “Assessment of Category 1 paper, NP/SC 7575, SZB EC 338898, Dry Fuel Store (FDS) Safety Case,” (Reference 26-39).

26.3.3 BWRX-300 Dry Storage

A Dry Storage Building would be constructed onsite to house the dry storage casks produced during BWRX-300 operation. Current SF cask maximum capacity is 89 BWR fuel bundles (Reference 26-21), although capacity varies dependent on the chosen cask. Based on the spent bundle quantity yielded by a 12-month refuelling cycle, the equivalent to 32 bundles would be replaced per year. This would mean an 89-bundle capacity cask would be filled and transported to the dry storage building approximately every 2.8 years. For the total of 2,128 bundles per unit lifetime, this would correspond to 24 casks of SF requiring dry storage. 100 years of storage will be assumed. Given there is capacity for 8 years’ worth of SF in the FP, a DFS would need to be online after approximately 9 years of reactor operation.

To reduce cask weight for road-based transport to the NPP site, the storage casks are to be transported as steel ‘shells’, unfilled by concrete. Upon receival at the site, a covered cask preparation area in the vicinity of the DFS will be used to fill the shells with concrete to provide additional shielding. The filled storage-overpack casks can then be stored in the DFS until required.

The exact building specifications would be a decision for the future licensee. To provide security and allow for environmental monitoring (e.g., internal temperature and humidity), the building will be covered. It will be designed to accommodate at least 25 casks (24 SF casks and 1 IICC cask) yielded over the 60-year operational lifetime. It will include consideration of cask transporter movement and ease of loading during operations. It will further account for the necessary environmental controls for managing canister heat loads distributed within the structure. This includes adequate roof insulation and careful design of the canister loading distribution within the building.

26.3.4 Storage of Failed Fuel

Storage of failed fuel will align with the chosen cask storage system, accounting for compatibility between specific failed fuel canister geometries and the dry storage cask. Arrangements for any additional containment will be judged on a case-by-case basis, dependant on the extent of failure. For GNF2 fuel, it is not anticipated that fuel failure will be severe enough to warrant extraordinary casking measures. However, there is the capacity for failed fuel pins to be separated from assemblies for separate disposal if necessary (according with the process described in Section 26.2.5. Severely failed fuel can be casked into dry storage canisters specifically designed for this purpose, supplied by the chosen cask vendor. These canister types are already in use internationally and currently being considered for use at SZB. Following the casking, failed fuel will be treated analogously to typical SF with respect to interim storage and eventual repackaging for long-term storage.

26.3.5 Integration of Fault Analysis into Dry Storage Designs

The specific operation and execution of SF interim storage using DFS is vendor-specific. However, fault analysis has been conducted on these systems to demonstrate incorporation of potential faults into DFS design.

A comprehensive probabilistic risk assessment (PRA) was produced by the US Nuclear Regulatory Commission (USNRC) “U.S. NRC Level 3 Probabilistic Risk Assessment Project Volume 7: Dry Cask Storage PRA” (Reference 26-40), building on work undertaken since 2006, “A Pilot Probabilistic Risk Assessment of a Dry Cask Storage System at a Nuclear Power Plant,” (Reference 26-41) and “Probabilistic Risk Assessment of Bolted Storage Casks:

Updated Quantification and Analysis Report," (Reference 26-42). This PRA extensively covers a wide range of faults possible during SF casking and dry storage, including, but not limited to the dropping of fuel assembly; the dropping of cask; tipping and damage from external hazards; cask overpressure; criticality; damaged fuel; incomplete drying; cask overpressure during drying. The report is based on a reference site and vendor but illustrates many of the risks common to all dry storage operations, including those already undertaken by GEH BWR plants in the US.

Risk analysis reports (both overarching and vendor-specific) would inform cask storage optioneering, indicating that the considered vendors will have taken into account fault studies during dry storage system design.

26.3.6 GNF2 Fuel Reliability During Dry Storage

An evaluation of GNF2 SF cladding under interim dry storage conditions was conducted by GNF, undertaken at the off-site interim dry storage facility of Mühleberg Nuclear Power Plant (KKM) in Switzerland (a GE-designed BWR Type 4 plant), 003N0801, "GNF2 Spent Fuel Interim Dry Storage Integrity Evaluation," (Reference 26-43).

GNF findings indicate that the SF integrity is maintained in this fuel design for 40 years when peak-pellet exposure is limited to 80 GWd/MTU and fuel temperature remains below a defined bounding limit (T_{max}= 350˚C upon exit from FP, then reducing with time) (Reference 26-43). This is based on evaluation of failure by creep, catastrophic crack propagation, delayed hydride cracking, and stress corrosion cracking. It also applies only to predicting the behaviour of intact (non-failed) fuel bundles, held in static, normal dry storage conditions and is conservative based on the use of extrapolation from short-term experiments (Reference 26-43). The storage solution employed by KKM is equivalent to the suggested approach described for the BWRX-300 site, although exact loading configuration, cask design, and evaluation processes are vendor specific. Reloads within the BWRX-300 will conform to a peak pellet burn-up of 70 GWd/MTU (within the upper-bound established in the GEH Dry Storage Integrity Evaluation) (Reference 26-16). The decay heat calculations performed as part of the GNF2 dry storage integrity study demonstrate that SF temperatures are expected to remain below T_{max}= 350˚C upon exit from the FP (Reference 26-43).

This work was undertaken in the context of supporting US licencing applications to the USNRC for the design and operation of US-based DFS. The use of "40 years" by GEH as a bench-mark dry storage duration to qualify GNF2 fuel is based on US DFS licensing cycles of 40 years. This is not indicative of an upper time limit on fuel integrity during dry storage. In fact, the authors note that conservatisms were introduced at each step of their analysis, leading to a significant layering of conservatisms in their approach.

Confidence in longer durations has been established with international OPEX and will continue to evolve based on ongoing UK dry storage experience. In the UK, operational facilities will reach (and assess viability of) longer storage times prior to DFS operation on a BWRX-300 site. It is indicative to note that dry cask storage technologies deployed at SZB are designed for 100-year operational lifetime.

26.3.7 In-storage Monitoring

During the storage lifetime, monitoring of the temperature difference across the MPC bottom and top, using a Temperature Difference Monitoring System, is used as a direct monitor of temperature, and an indirect monitor of containment. Changes in the temperature difference may indicate loss of helium fill gas due to a possible breach in containment, from modelling studies. The Health Physics function would monitor the top vent ports of the concrete-steel overpacks, and in long-term storage would be used to detect the presence of radiological species. If the radiological risk is considered sufficiently low, eddy current inspection equipment could then be used to detect breakthrough of the sacrificial layer and containment.

26.3.8 Demonstration of Suitability of Approach

There is significant UK and worldwide OPEX in the use of dry storage facilities for SF. In the US alone, USNRC licensing has been granted to in excess of 40 licensees, based on five DFS designs (as of October 2020, see Appendices N and O of Reference 26-44). The preferred storage approach described vendor-agnostically in this chapter is one option, based on commercially available technology and currently implemented processes (specifically the DFS at SZB). The final design decision will be the responsibility of future licensees.

26.4 Spent Fuel Disposal

26.1.1 Spent Fuel and IICC Disposal Approach

In-keeping with current UK Government policy, it is anticipated that, following a period of on-site dry storage, currently assumed to be 100 years, BWRX-300 SF and IICC will be repackaged and sent to a GDF. The implementation of High Heat Generating Waste (HHGW) disposal and the GDF is outside the scope of the PSR. However, a high-level overview of how the interim storage and repackaging of SF will be compatible with envisioned GDF best practice is provided.

26.4.1 Final Repackaging

At the end of the storage lifetime, a Dry Fuel Repackaging Plant (DFRP) is to be constructed on-site to allow fuel to be transferred to containers suitable for disposal prior to transport to final disposal. Appropriate record keeping will ensure suitable documentation is maintained regarding the fuel history and dry storage cask loadings to aid the subsequent repackaging and disposal. The plant may require an on-site interim storage area for repackaged SF prior to transport to GDF depending on future Nuclear Waste Services (NWS) requirements.

IICC will also be repackaged following interim storage. Having undergone decay and cooling within the SF storage casks, the expectation is that, where radiogenic heat has reduced to <2 kW/m^3, they will meet the criteria to be classified as ILW (Reference 26-1). Thus, repackaged IICC will be in accordance with NWS criteria for “Waste Package Specification and Guidance Documentation: Specification for Waste Packages Containing Low Heat Generating Waste,” (Reference 26-45).

26.4.2 Disposal

Any necessary repackaging of waste forms following interim storage shall comply with future GDF WAC and transport specifications and should be deployed by future licensees once final disposal operations become more relevant. It is expected that repackaging will be based on ALARP/BAT considerations at the time of design and will conform to current NWS expectations stipulating periodic reviews and record-keeping. The repackaging of interim storage SF will leave the original interim storage canisters as ILW/LLW (dependent on storage duration) which will be dealt with in accordance with procedures described in NEDO-34166, (Reference 26-5).

Given SF is HHGW, transport packages will be of the Type B design category described in the International Atomic Energy Agency (IAEA) Transport Regulations of the safe Transport of Radioactive Material,” (Reference 26-46). The exact waste package for transport and disposal will be vendor specific, though current understanding in this area points to repackaging into final disposal containers adhering to one of two possible design variants, which are outlined in WPS/240/02, “Waste Package Specification and Guidance Documentation: Specification for High Heat Generating Waste Precursor Product,” (Reference 26-47). On site, prior to loading into transport containers, the waste packages will be checked for any signs of structural degradation.

26.4.3 Demonstration of Disposability

NWS undertakes assessments to determine a preliminary disposability assessment of wastes arising from the operation and decommissioning of new reactors as part of the GDA. This NWS Expert View was undertaken as part of Step 2, NEDC-34230P, “BWRX-300 UK GDA Disposability Expert View,” (Reference 26-48). The Requesting Party (RP) engaged constructively with NWS to establish the scope and detail of the Expert View. A supporting document, NEDC-34229P, “BWRX-300 UK GDA Demonstration of Disposability,” (Reference 26-49) was produced to help NWS establish their Expert View (Reference 26-48). The analysis in the Expert View found no major impediments to the disposability of BWRX-300 SF, in alignment with other LWR fuel for disposal in the UK and NWS confirmed that a disposability case could be made for the Higher Activity Wastes and spent fuel from the BWRX-300.

NWS advised that current LWR fuel for disposal in a UK GDF is spent PWR fuel from SZB and that differences between spent PWR and BWR fuel would need further accommodating with future development. These relate mainly to the size of the SF assemblies and the SF inventory source term.

NWS therefore identified a number of risks that required addressing, including one risk classed as 'Medium' relating to the disposability of SF. This risk requires more significant attention because it has a potentially larger effect on GDF development. The risk relates to the assessment of post-closure criticality aspects of the GDF safety case. At present, the case does not take adequate account of the inclusion of SF from BWRX-300. GEH have acknowledged that this gap will require addressing as part of the development of a disposability case for BWRX-300 SF and have identified an appropriate forward action (FAP-PER5-374) as part of NEDO-34222, "BWRX-300 UK GDA Chapter E5: "Radioactive Waste Management Arrangements," (Reference 26-10), arranging for stakeholder engagement between a future developer/operator with NWS to support BWR SF post-closure criticality safety assessment.

This is in line with the developments undertaken for spent GE14 fuel from the ABWR GDA and NWS' expert view outlined therein.

26.4.4 Current GDF Programme and Emplacement

The nature of a UK GDF is yet to be determined, but the design requirements are outlined in "Geological Disposal: Generic Disposal System Specification," (Reference 26-50). Transport to, and emplacement in, the GDF will be facilitated by RGP at the time a GDF has been made available and will be in line with the relevant transport regulations.

Based on current projections, NWS is leading the GDF programme with anticipated readiness for ILW by the 2050s and HLW/SF from 2075. The current UK emplacement plans for the 24 GW of new nuclear builds are not finalised yet, with disposal and emplacement dates still unclear. Therefore, 100-year on-site storage prior to final repackaging, transport, emplacement, and disposal is considered appropriate.

26.5 Conclusions

UK RGP and OPEX suggests a dry cask storage approach is the most suitable for implementation in the BWRX-300 design. A generic viable option for this form of interim storage of HHGW has been evaluated, although site-specific constraints will influence the final implementation of this design by the licensee.

A description of the management of SF and IICC has been provided, covering a description of the arisings, FP operations, dry cask storage facility operations, and a high-level overview of disposal. A viable blueprint for a future plan for the management, storage, and disposal of BWRX-300 SF and IICC has been provided. A demonstration of disposability has been undertaken, ensuring NWS was able to provide the required disposability Expert View.

No significant obstacles were identified by NWS in their Expert View and all identified issues have been mitigated or added to the project Forward Action Plan for resolution in a future Step 3.

Table 26-1: Total Number of SF Bundles Generated After 60 Years Operation

Fuel Cycle Length	Number of Bundles
12-month cycle	2128
24-month cycle	2368

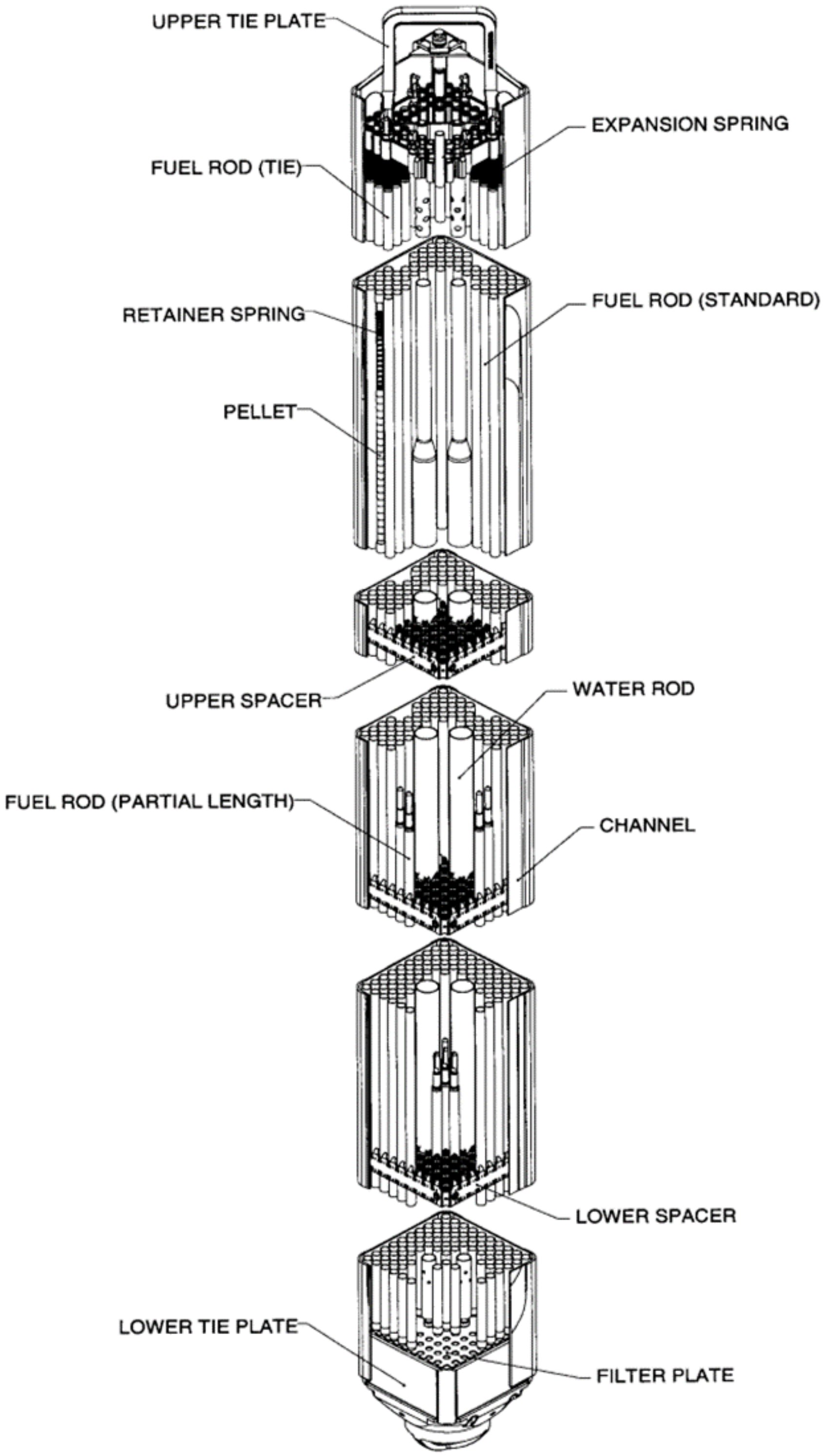

Figure 26-1: GNF2 Fuel Assembly (Reference 26-16)

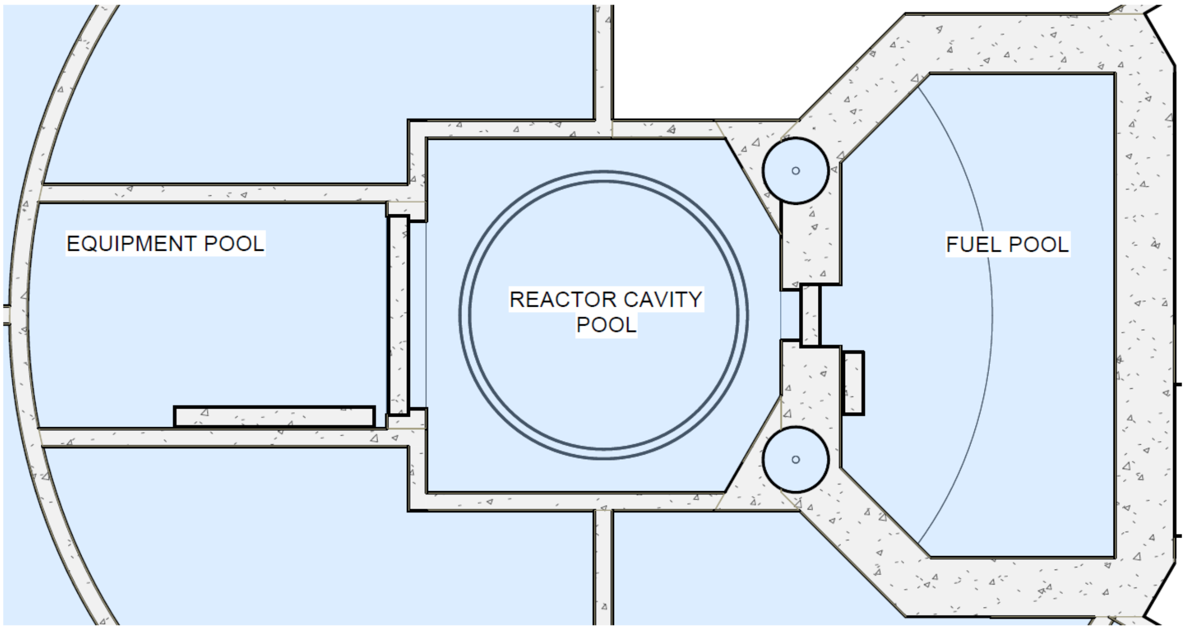

Figure 26-2: Arrangement of Fuel, Reactor Cavity and Equipment Pools

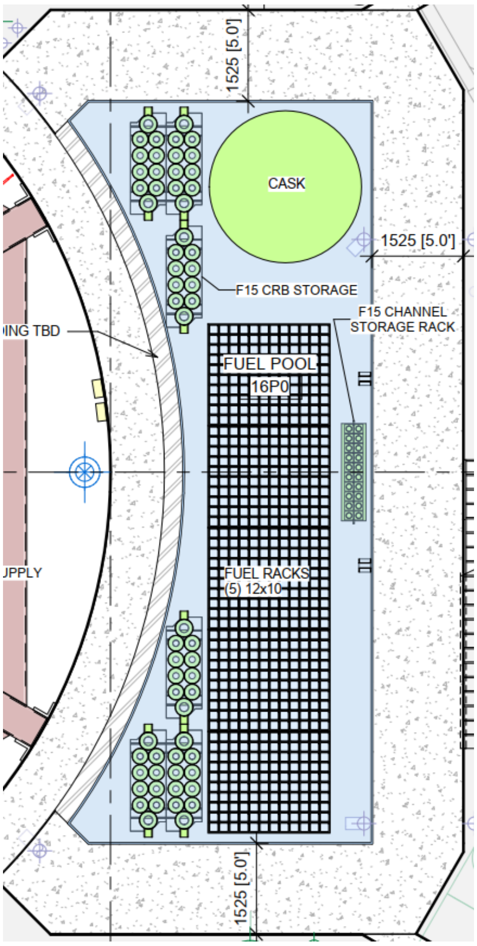

Figure 26-2: Fuel Pool Arrangement (CRB: Control Rod Blades), adapted from 005N1730, "BWRX-300 Reactor Building General Arrangement Drawings" (Reference 26-51)

26.6 References

26-1 "Basic Principles of Radioactive Waste Management", Revision 2, ONR, EA, SEPA, NRW, February 2015.

26-2 NEDO-34163, "BWRX-300 UK GDA Chapter 1: Introduction and Overview," Rev B, GE-Hitachi Nuclear Energy, Americas, LLC.

26-3 NEDO-34164, "BWRX-300 UK GDA Chapter 3: Safety Objectives and Design Rules," Rev B GE-Hitachi Nuclear Energy, Americas, LLC.

26-4 NEDC-34166P, "BWRX-300 UK GDA Chapter 4: Reactor (Fuel and Core)," Rev B, GE-Hitachi Nuclear Energy, Americas, LLC.

26-5 NEDO-34166, "BWRX-300 UK GDA Chapter 11: Management of Radioactive Waste," Rev B, GE-Hitachi Nuclear Energy, Americas, LLC.

26-6 NEDO-34197, "BWRX-300 UK GDA Chapter 25: Security Annex," Rev B, GE-Hitachi Nuclear Energy, Americas, LLC.

26-7 NEDO-34200, "BWRX-300 UK GDA Chapter 28: Safeguards Annex," Rev B, GE-Hitachi Nuclear Energy, Americas, LLC.

26-8 NEDO-34218, "BWRX-300 UK GDA Chapter E1: Introduction," Rev B, GE-Hitachi Nuclear Energy, Americas, LLC.

26-9 NEDO-34221, "BWRX-300 UK GDA Chapter E4: Information About the Design," Rev B, GE-Hitachi Nuclear Energy, Americas, LLC.

26-10 NEDO-34222, "BWRX-300 UK GDA Chapter E5: Radioactive Waste Management Arrangements," Rev B, GE-Hitachi Nuclear Energy, Americas, LLC.

26-11 NEDO-3422, "BWRX-300 UK GDA Chapter E7: Radioactive Discharges," Rev B, GE-Hitachi Nuclear Energy, Americas, LLC.

26-12 ONR-NR-AR-17-019, "Step 4 Assessment of Fuel & Core Design for the UK Advanced Boiling Water Reactor," ONR, December 2017.

26-13 GA91-9901-0046-00001 (XE-GD-0156) "UK ABWR Generic Design Assessment: Preliminary Safety Report on Reactor Core and Fuels," Rev B, Hitachi-GE Nuclear Energy, 2014.

26-14 GA91-9101-0101-11000, "UK ABWR Generic Design Assessment - Generic PCSR Chapter 11: Reactor Core," Hitachi-GE Nuclear Energy, Revision C, 2017.

26-15 GA91-9101-0101-19000, "UK AWBR Generic Design Assessment - Generic PCSR Chapter 19: Fuel Storage and Handling," Hitachi-GE Nuclear Energy, 2017.

26-16 006N1887, "BWRX-300 Fuel Design and Qualification," Revision A, GNF, August 2020.

26-17 NEDC-33941P, "GNF2 Fuel Assembly Thermal-Mechanical Design Report," Revision 2, GE-Hitachi Nuclear Energy, Americas, LLC., June 2023.

26-18 NEDC-34159P, "BWRX-300 UK GDA Fuel Summary Report," Rev A, GE-Hitachi Nuclear Energy, Americas, LLC.

26-19 DBR-0057741, "BWRX-300 Plant Performance Envelope," Revision 0, GE-Hitachi Nuclear Energy, Americas, LLC., April 2022.

26-20 NEDC-33415P, "Nordic GNF2 Operational Experience Update, Supplement 1," Rev 4, GNF/ENUSA, November 2023.

26-21 006N5399, "BWRX-300 Irradiated Fuel Management Plan," Revision C, GE-Hitachi Nuclear Energy, Americas, LLC., 2021.

26-22 ULTRA Control Rod Blades Fact Sheet, GE-Hitachi Nuclear Energy, Americas, LLC., 2011.

26-23 005N9751, "BWRX-300 General Description," Revision F, GE-Hitachi Nuclear Energy, Americas, LLC., December 2023.

26-24 NEDE-33284, "Marathon-Ultra Control Rod Assembly," Supplement 1P-A, Revision 1, GE-Hitachi Nuclear Energy, Americas, LLC., March 2012.

26-25 Gamma Thermometer Datasheet, Reuter-Stokes, 2023 – Access at: Gamma-Thermometers-Datasheet-cva.pdf

26-26 NEDO-34169, "BWRX-300 UK GDA Chapter 7: Instrumentation and Control," Rev A, GE-Hitachi Nuclear Energy, Americas, LLC.

26-27 006N5377, "System Design Description: BWRX-300 Refueling and Servicing Equipment," Rev 3, GE-Hitachi Nuclear Energy, Americas, LLC., May 2024.

26-28 NEDO-34171, "BWRX-300 UK GDA Ch 9A: Auxiliary Systems," Rev A, GE-Hitachi Nuclear Energy, Americas, LLC.

26-29 008N0988, "BWRX-300 Power Block General Arrangement Drawings," Revision 2, GE-Hitachi Nuclear Energy, Americas, LLC., 2023

26-30 007N0022, "BWRX-300 Spent Fuel Pool Decay Heat," GE-Hitachi Nuclear Energy, Americas, LLC., 2022.

26-31 006N7941, "System Design Description: BWRX-300 Fuel Pool Cooling and Cleanup," Rev 2, GE-Hitachi Nuclear Energy, Americas, LLC., March 2024.

26-32 NEDC-33940P, "BWRX-300 GNF2 Fuel Assembly: Mechanical Design Report," Revision 0, GE-Hitachi Nuclear Energy, Americas, LLC., September 2022.

26-33 IAEA-TECDOC-1862, "Behaviour of Spent Power Reactor Fuel during Storage," IAEA, 2019.

26-34 NEDO-34185, "BWRX-300 UK GDA Chapter 15.7: Safety Analysis – Internal Hazards," Rev A, GE-Hitachi Nuclear Energy, Americas LLC.

26-35 "Written evidence submitted by Magnox Limited (FNP 53)", Magnox Ltd, https://committees.parliament.uk/writtenevidence/65293/pdf/, March 2016.

26-36 "Wylfa Site: Environmental Management Plan," Magnox Ltd, DP/EMP/ENG/001 Issue 10, 2021.

26-37 NDA DRP Lot 4 WP 15 (2), "AGR Fuel Storage Atmospheres," Babcock International Group, 2011.

26-38 ONR-OPF-PAR-16-010, "Assessment of Sizewell B Dry Fuel Store Post Operational Safety Case NP/SC 7575, EC 338898-1," ONR, October 2016.

26-39 ONR-SZB-PAR-13-020, "ONR's assessment of Category 1 paper, NP/SC 7575, SZB EC 338898, Dry Fuel Store (DFS) safety case," Revision 1, ONR, July 2013.

26-40 "Level 3 Probabilistic Risk Assessment Project - Volume 7: Dry Cask Storage PRA," B. W. F. Gonzalez, U.S.NRC, 2024.

26-41 NUREG-1864 "A Pilot Probabilistic Risk Assessment of a Dry Cask Storage System at a Nuclear Power Plant," U.S. NRC, 2007.

26-42 EPRI-1009691, "Probabilistic Risk Assessment of Bolted Storage Casks: Updated Quantification and Analysis Report," EPRI, 2004.

26-43 003N0801, "GNF2 Spent Fuel Interim Dry Storage Integrity Evaluation," GNF, Revision 0, December 2015.

26-44 NUREG-1350, "2020-2021: Information Digest," Volume 32, U.S. NRC, October 2020.

26-45 WPS/220/01, "Waste Package Specification and Guidance Documentation: Specification for Waste Packages Containing Low Heat Generating Waste," Radioactive Waste Management, March 2020.

26-46 "IAEA Regulations for the safe Transport of Radioactive Material," IAEA Safety Standards Series, No. SSR-6 (Rev.1), IAEA, 2018.

26-47 WPS/240/02, "Waste Package Specification and Guidance Documentation: Specification for High Heat Generating Waste Precursor Product," NWS, October 2024.

26-48 NEDC-34230P, "BWRX-300 UK GDA Disposability Expert View (NWS)," Rev A, NWS, GE-Hitachi Nuclear Energy, Americas LLC.

26-49 NEDC-34229P, "BWRX-300 UK GDA Demonstration of Disposability," Rev A, GE-Hitachi Nuclear Energy, Americas LLC.

26-50 DSSC/401/01, "Geological Disposal: Generic Disposal System Specification," RWM, December 2016.

26-51 005N1730, "BWRX-300 Reactor Building General Arrangement Drawings," Rev G, GE-Hitachi Nuclear Energy, Americas, LLC.

26-52 NEDC-34140P, "BWRX-300 UK GDA Safety Case Development Strategy," Rev A, GE-Hitachi Nuclear Energy, Americas, LLC.

26-53 NEDO-34184, "BWRX-300 UK GDA Chapter 15.6: Probabilistic Safety Assessment," Rev B, GE-Hitachi Nuclear Energy, Americas LLC.

26-54 "Safety Assessment Principles for Nuclear Facilities", Revision 1, ONR, 2014 Editions, January 2020.

26-55 "Safety Aspects Specific to Storage of Spent Nuclear Fuel," ONR, 2021.

26-56 "Storage of Spent Nuclear Fuel, IAEA Safety Standard Series," IAEA, No.SSG-15 Rev 1), 2020.

26-57 "Waste and Spent Fuel Storage Safety Reference Levels," WENRA, January 2024.

26-58 "Interim Storage of Higher Activity Waste Packages - Integrated approach," Nuclear Decommissioning Authority, December 2021.

APPENDIX A CLAIMS, ARGUMENTS AND EVIDENCE

A.1 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 NEDC-34140P, "BWRX-300 UK GDA Safety Case Development Strategy," (SCDS) (Reference 26-52) 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 2 sub-claims that this chapter demonstrates compliance against are identified within the SCDS (Reference 26-52) and are as follows:

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 as low as reasonably practicable.

The Level 3 sub-claims that this chapter demonstrates compliance against are identified within the SCDS (Reference 26-52) and are as follows:

2.1.1 The safety functions (Design Basis) have been derived for the system/structure through a robust analysis, based upon RGP.

2.1.2 The design of the system/structure has been substantiated to achieve the safety functions in all relevant operating modes.

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.

2.1.4 System/structure performance will be validated by suitable testing throughout manufacturing, construction, and commissioning.

2.1.5 Ageing and degradation mechanisms will be identified and assessed in the design. Suitable examination, inspection, maintenance, and testing will be specified to maintain systems/structures fit-for-purpose through-life.

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.

2.4.1 RGP has been taken into account across all disciplines.

2.4.2 OPEX and Learning from Experience (LfE) has been taken into account across all disciplines.

2.4.3 Optioneering (all reasonably practicable measures have been implemented to reduce risk).

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.

A.2 Risk Reduction As Low As Reasonably Practicable (ALARP)

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:
  - RGP 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 26-52)

Probabilistic safety aspects of the ALARP argument are addressed within NEDO-34184, “BWRX-300 PSR Chapter 15.6: Probabilistic Safety Assessment,” (Reference 26-53).

Table A-1: CAE Route Map

Level 26 Claim	Level 26 Sub-claim	Chapter 26 Arguments	Sub Section and/or reports that evidence the arguments
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.	2.1.1 The safety functions (Design Basis) have been derived for the system/structure through a robust analysis, based upon RGP.	FP operations and DFS operation are based on significant OPEX and are aligned with UK RGP and ONR TAG 81.	26.2.2 Description of Relevant Fuel Route Operations 26.3.3 BWRX-300 Dry Storage
	2.1.2 The design of the system/structure has been substantiated to achieve the safety functions in all relevant operating modes.	The FP is capable of housing fuel from 8 years of operation, plus new fuel and one full core offload of fuel assemblies.	26.2.3 Spent Fuel and IICC Fuel Pool Operations Document “BWRX-300 Refuelling and Servicing Equipment” 006N5377 (Reference 26-27)
	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.	FP operations and DFS operation are based on significant OPEX and are aligned with UK RGP and ONR TAG 81.	26.2.2 Description of Relevant Fuel Route Operations 26.3.3 BWRX-300 Dry Storage
	2.1.4 System/structure performance will be validated by suitable testing throughout manufacturing, construction, and commissioning.	Future validation and fault studies will be undertaken prior to construction of the DFS.	26.3.5 Integration of Fault Analysis into Dry Storage Designs
	2.1.5 Ageing and degradation mechanisms will be identified and assessed in the design. Suitable examination, inspection, maintenance, and testing will be specified to	SF will be monitored during operation and storage, including detection of failure of containment.	26.3.7 In-storage Monitoring

Level 26 Claim	Level 26 Sub-claim	Chapter 26 Arguments	Sub Section and/or reports that evidence the arguments
	maintain systems/structures fit-for-purpose through-life.	GNF2 fuel assemblies have been designed to minimise degradation during operation and evaluated for interim storage of up to 40 years.	26.1.1.1 GNF2 fuel 26.3.6 GNF2 Fuel Reliability During Dry Storage Document “GNF2 Spent Fuel Interim Dry Storage Integrity Evaluation” 003N0801 (Reference 26-43) Document “BWRX-300 Fuel Design and Qualification,” 006N1887 (Reference 26-16)
	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.	Measures relating to safe SF decommissioning have been accounted for in the proposed DFS.	26.1.1 Spent Fuel and IICC Disposal Approach
2.4 Safety risks have been reduced as low as reasonably practicable	2.4.1 RGP has been taken into account across all disciplines.	The approach to SF handling and DFS is aligned to UK-specific RGP supported by further worldwide OPEX.	26.2.2 Description of Relevant Fuel Route Operations 26.3.2 Relevant United Kingdom Spent Fuel Storage Experience
	2.4.2 OPEX and LfE has been taken into account across all disciplines.	The design aspects for SF in BWRX-300 incorporate international OPEX and LfE.	26.3.2 Relevant United Kingdom Spent Fuel Storage Experience
	2.4.3 Optioneering (all reasonably practicable measures have been implemented to reduce risk).	Licensee will need to perform a full optioneering assessment of the cask vendors during vendor selection.	26.3.1 Spent Fuel and IICC Storage Approach 26.3.5 Integration of Fault Analysis into Dry Storage Designs

APPENDIX B FORWARD ACTIONS

Table B-1: Forward Action Plan

Fap No.	Finding	Forward Action Plan Item	Delivery Phase
PSR26-155	An option for the DFS is provided in this step of the GDA, but further development is site-specific and may be subject to change dependent on evolving BAT by the time construction of a storage facility is relevant.	A future developer/operator shall establish a comprehensive, site-specific plan for the DFS, including numbers of units, design, layout, on-site transport processes, procedures of the constructed storage facility and any other site-specific details, as required.	Before Site License Application, Environmental Permit Applications and/or BL3 Design Phase
PSR26-156	SF source term information is outside the scope of Step 2 of the GDA.	A future developer/operator shall derive a 'Realistic Model' End User Source Term for BWRX-300 spent fuel waste streams to provide an estimated spent fuel inventory under normal operating conditions, to allow strategic decision-making for provision of appropriate management and storage arrangements aligned with UK relevant good practice and relevant regulatory requirements. .	For PCSR/PCER
PSR26-157	IICC will be casked in the FP and dry stored to decay until the radiogenic heat output drops to ILW levels (<2 kW/m^3).	A future developer/operator shall establish an understanding of the casking timelines for IICC and their radiogenic heat production evolution to ensure demonstration of their disposability as ILW.	For PCSR/PCER
PSR26-158	No cask vendor or technology has been selected yet by the RP.	A future developer/operator shall undertake optioneering to complete a comprehensive vendor selection study, including full fault studies, to decide on the cask technology to be deployed.	Before Site License Application, Environmental Permit Applications and/or BL3 Design Phase

APPENDIX C UNITED KINGDOM-SPECIFIC-CONTEXT

C.1 Safety Assessment Principles

The ONR has established "Safety Assessment Principles (SAP) for Nuclear Facilities," (Reference 26-54) which are applied by specialist inspectors in the assessment of facility safety cases put forward by licensees.

C.2 Guidance

SAPs are further underpinned by topic-specific Technical Assessment Guides (TAG). The safe storage of SF is underpinned by TAG 081, "Safety Aspects Specific to Storage of Spent Nuclear Fuel," (Reference 26-55).

TAG 081 outlines the necessary approach to enable the safe functioning of a SF storage facility, highlights relevant nuclear site licence conditions, lists relevant UK legislation, and identifies international RGP. If the licensee meets the expectations set out in the TAG, they are understood to comply with the requirements detailed out in the legislation, licence conditions, and RGP highlighted in the following sections.

C.3 Legislation

No specific UK legislation exists governing the safe storage of SF, but a list of relevant legislation includes:

- Ionising Radiations Regulations 2017 (IRR17)
- Management of Health & Safety at Work Regulations 1999 (MHWSR)
- Nuclear Industries Security Regulations 2003 (NISR)
- The Nuclear Safeguards (EU Exit) Regulations 2018
- The Energy Act 2008

C.4 Relevant License Conditions

The following licence conditions are identified as being relevant to the storage of SF at a nuclear licenced site.

- 4 Restrictions on nuclear matter on the site
- 5 Consignment of nuclear matter
- 6 Documents, records, authorities and certificates
- 15 Periodic review
- 23 Operating rules
- 25 Operational records
- 28 Examination, inspection, maintenance and testing
- 32 Accumulation of radioactive waste
- 34 Leakage and escape of radioactive material and radioactive waste
- 35 Decommissioning

C.5 Relevant Good Practice

The IAEA's SSG-15 is identified as RGP for the storage of SF, "Storage of Spent Nuclear Fuel," (Reference 26-56), as are the WENRA's Waste and Spent Fuel Storage Safety Reference Levels," (Reference 26-57).

Additionally, the Nuclear Decommissioning Authority's Industry Guidance on "Interim Storage of Higher Activity Waste packages – Integrated Approach," (Reference 26-58) and the NWS WPS 240 on Higher Activity Wastes (Reference 26-47) were consulted.