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24.1 INTRODUCTION

The Fundamental Purpose of the Generic Design Assessment (GDA) Safety, Security and Environment Case (SSEC) is to demonstrate that the Generic Small Modular Reactor (SMR)-300 can be constructed, commissioned, operated, and decommissioned on a generic site in the UK to fulfil the future licensee's legal duties to be safe, secure and protect people and the environment, as defined in Part A Chapter 1 [1].

The Fundamental Purpose is achieved through the Fundamental Objective of the Preliminary Safety Report (PSR), which is to summarise the safety standards and criteria, safety management and organisation, claims, arguments and intended evidence to demonstrate that the generic SMR-300 design risks to people are likely to be tolerable and As Low as Reasonably Practicable (ALARP) [1].

Part B Chapter 24 of the PSR presents the Claims, Arguments and intended Evidence (CAE) for the design of Fuel Transport and Storage Structures, Systems and Components (SSCs) that underpin the design of the generic SMR-300.

24.1.1 Purpose and Scope

The Overarching SSEC Claims are presented in Part A Chapter 3 [2].

This Chapter (Part B Chapter 24) links to the overarching claim through Claims 2.2 and 2.3:

Claim 2.2: The design of the systems and associated processes have been developed taking cognisance of relevant good practice and substantiated to achieve their safety and non-safety functional requirements.

Claim 2.3: The design and safety assessment of the generic Holtec SMR-300 considers the entire reactor lifecycle.

As set out in Part A Chapter 3 [2], Claim 2.2 is further decomposed across several engineering disciplines which are responsible for development of the design of relevant SSCs.

This chapter presents SSCs associated with the fuel route design for the generic SMR-300 and therefore directly supports Claim 2.2.17.

Claim 2.2.17: SSCs which support operational fuel activities are designed to ensure safety functions and measures are delivered and radiation exposure and release of radioactivity are minimised ALARP.

As set out in Part A Chapter 3 [2], Claim 2.3 is further decomposed across several disciplines which support the development of through-life management arrangements and use of Relevant Good Practice (RGP).

This chapter presents the overall management of spent fuel aspects for the generic SMR-300 and therefore directly supports Claim 2.3.4.

Claim 2.3.4: Spent fuel will be safely managed throughout the entire reactor lifecycle.

The scope of this chapter covers the Fuel Transport and Storage SSCs and operations as set out in Sub-chapter 24.2.

Further discussion on how the Level 3 claims are broken down into Level 4 claims and how the Level 4 claims are met is provided in Sub-chapter 24.3.

Sub-chapter 24.4 covers the codes and standards associated with Fuel Transport and Storage, the Design of SSCs, Safety Assessment and Quality, Manufacturing, Installation and Examination Inspection Maintenance and Testing (EIMT).

Sub-chapter 24.5 covers the Spent Fuel Management Lifecycle, including Spent Fuel Management Strategy and Nuclear Liabilities Regulation (NLR) compliance.

Finally, Sub-chapter 24.6 provides a technical summary of how the claims for this Chapter have been achieved, together with a summary of key contributions to the overall ALARP case and to demonstrating BAT. No GDA commitments are raised in this Chapter.

The Fuel Transport and Storage operations within the scope of this chapter include:

- **New fuel transport and storage:** from its receipt and inspection in the Radiologically Controlled Area (RCA) of the Reactor Auxiliary Building (RAB), encompassing its transport, handling and storage, and transfer to the Spent Fuel Pool (SFP), to its loading in the reactor core.
- **Spent fuel transfer and storage:** from its unloading from the reactor core, encompassing its transfer, handling and storage, to the point it is retrieved from the SFP for a dry storage campaign.
- **Spent fuel transport and storage (dry storage campaign operations):** from the retrieval of spent fuel from the SFP, encompassing its transfer, processing[1] and subsequent transport across site using a Multi-Purpose Canister (MPC), through MPC downloading and sealing in a Vertical Ventilated Storage Module (VVM) of the Holtec International Storage Module Underground Maximum Capacity (HI-STORM UMAX) system, to its interim storage at the on-site Independent Spent Fuel Storage Installation (ISFSI).

The scope includes transport and storage of any damaged fuel and of Non-Fuel Waste (NFW). Types of NFW include redundant in-core[2] and ex-core components that are stored within the SFSR prior to packaging for interim storage. They are high dose rate activated metallic components. Damaged fuel and NFW are both managed in a similar manner to spent fuel, through storage in the SFP followed by transfer for processing for dry storage in the UMAX

[1] Not formally in scope, but a high level description is included for completeness in this Chapter.

[2] Examples of NFW for light water reactors include burnable poison rod assemblies, thimble plug devices, control rod assemblies, axial power shaping rods, wet annular burnable absorber, Rod Cluster Control Assemblies (RCCA), neutron source assemblies, water displacement guide tube plugs, instrument tube tie rods, vibration suppressor inserts, and components of these devices.

system. See Sub-sections 24.2.4.20 and 24.2.4.21 for further information on the management of NFW and damaged fuel.

A GDA scope change proposal paper 'Reduction in GDA Scope for the HI-STORM UMAX System' [3] was raised in Step 2 by the RP to rationalise submissions within the Spent Fuel Management topic area. The change reduced the level of detail of the UMAX System under assessment at GDA, but retained the aspects of both the fuel route and dry storage system within scope to enable a fundamental assessment by the regulators. These include the atypically small SFP, which constrains storage capacity for spent fuel more than at a typical Pressurised Water Reactor (PWR); and novel design features of the UMAX system[3].

More detailed consideration during Step 2, reported in the Holtec SMR-300 GDA Spent Fuel Management Strategy [4], revealed that the SMR-300 SFP design has a greater potential storage capacity than had originally been realised, along with potential means to increase storage capacity (e.g. by direct loading of new fuel into the reactor core from a submerged MPC, and/ or inclusion of additional rows of storage cells), and also to increase margin to SFP capacity (e.g. by using alternative MPC loading strategies).

A review of the UMAX system design during Step 2 identified design features of interest, based on continuous development of Holtec's proprietary dry storage technology over a period of decades, taking into account learning, international good practice and Operating Experience (OPEX), contribute to significantly improved safety, security, environmental performance/ sustainability and efficiency [4].

The Spent Fuel Management Strategy developed during Step 2 establishes a compliant baseline strategy for the long-term management of spent fuel, damaged fuel and NFW, along with retained alternative credible options based on UK expectations, as risk mitigations for novel aspects subject to further SSEC substantiation.

This establishes a 'design envelope' for spent fuel management for the generic SMR-300, demonstrating that the design is capable of being permitted by the regulators. This approach provides a high level of confidence that a robust, fully compliant site-specific Spent Fuel Management Strategy can be developed at the Pre-Construction Safety Report (PCSR) stage.

A decision was consequently made to limit the scope of assessment of the UMAX system during Step 2 of GDA. to primarily a Nuclear Liabilities Regulation (NLR) review [3]. Assessment scope on the UMAX system for interfacing topics is therefore deferred to subsequent phases, as is typical for consideration of specific dry storage systems during GDA Steps 1 and 2.

Excluded from the Part B Chapter 24 scope are:

- Site-specific considerations.
- Detailed plant design aspects e.g. dedicated Instrumentation & Control (I&C).
- Spent fuel processing for dry storage within the Fuel Handling Area (FHA) of the RAB RCA, using the Forced Helium Dehydration (FHD) system.

[3] [REDACTED]

- Spent fuel management operations following the end of the interim storage phase.

Notwithstanding this, high level strategic assumptions for spent fuel management following the interim storage phase are made in this Chapter, consistent with a fully compliant baseline strategy for the generic SMR-300 that adequately considers UK policy and strategic frameworks.

A master list of definitions and abbreviations relevant to all PSR Chapters can be found in Part A Chapter 2 [5].

24.1.2 Assumptions

Assumptions which relate to this topic have been formally captured using the Commitments, Assumptions and Requirements (CAR) process [6]. Further details of this process are provided in Part A Chapter 4 Lifecycle Management of Safety and Quality Assurance [7] (MSQA).

The identified assumptions are set out below, assumption headings are as they appear in the CAR Register:

Installation of Dry Fuel Storage Solution

- The UMAX system will be installed during the power station construction phase.

Spent Fuel Cooling and Processing Campaigns

- Fuel will be stored in the Spent Fuel Storage Racks (SFSR) within the SFP for a minimum of [REDACTED] years to allow for sufficient cooling prior to loading into an MPC for a dry fuel storage campaign.
- Dry fuel storage campaigns will commence after [REDACTED] and will occur after every refuelling cycle thereafter, until all spent fuel has been transferred to the UMAX system at the ISFSI; NFW will be subject to a dry storage campaign upon accumulation of sufficient waste within the SFSR to fill a Non-Fuel Waste Canister (NFWC).

Disposal Routes and Timing

- Spent fuel and NFW will be interim stored in the UMAX system for 100 years and then repackaged into Nuclear Waste Services (NWS) approved canisters/ containers at a shielded repackaging plant, prior to disposal to the Geological Disposal Facility (GDF).

24.1.3 Interfaces with other SSEC Chapters

The Fuel Transport and Storage Chapter interfaces with the following PSR chapters.

Part B Chapter 1 Reactor Coolant System and Engineered Safety Features [8] in relation to the Reactor Coolant System (RCS) and engineered safety features, and the interface between the SSCs of the fuel route and the RCS.

Part B Chapter 2 Reactor [9] in relation to the fuel characteristics, the fuel cycle and core management.

Part B Chapter 4 Control and Instrumentation Systems [10], in relation to I&C principles, methodologies and approaches applicable to the fuel route e.g., controls over the movement of cranes over the SFP.

Part B Chapter 5 [11], in relation to required services e.g., auxiliary systems involved in fuel transport and storage such as the fuel bridge, Polar Crane, etc.; and the Spent Fuel Pool Cooling System (SFC), which cools the water in the SFP, and controls its chemistry, clarity and radioactivity levels.

Part B Chapter 9 Description of Operational Aspects/Conduct of Operations [12], in relation to the fuel route operational procedures and arrangements.

Part B Chapter 10 Radiological Protection [13], in relation to radiological protection aspects, including criticality control, radiological protection and contamination control e.g., within the SFP and the RAB FHA.

Part B Chapter 13 Radioactive Waste Management [14], in relation to the arrangements for the safe management of radioactive wastes, for example for NFW, and for the management of secondary wastes generated e.g. SFC resins and filters.

Part B Chapter 14 Safety/Design Basis Accident Analysis [15], in relation to the methodology and principles for analysis of potential Design Basis Accidents (DBA) and the categorisation and classification approach for SSCs.

Part B Chapter 15 BDBA, Severe Accidents Analysis and Emergency Preparedness [16], in relation to the methodology, principles and approach to analysis of Beyond Design Basis Accidents (BDBA)/ Severe Accident Analysis (SAA) relevant to fuel transport and storage, including strategies and guidelines to mitigate Beyond Design Basis (BDB) external events from natural phenomena.

Part B Chapter 17 Human Factors [17], in relation to the methodology, principles and approach to human factors considerations for workers involved in operations.

Part B Chapter 19 Mechanical Engineering [18], in relation to the application of mechanical engineering analyses, for example to the analysis of dynamic loads.

Part B Chapter 20 Civil Engineering [19], in relation to the methodology, principles and approach to civil engineering of relevant structures, such as the Containment Structure (CS) and Containment Enclosure Structure (CES) which house and protect the SFP, and the New Fuel Vault (NFV) and SFP which house and protect the New Fuel Storage Rack (NFSR) and SFSRs respectively.

Part B Chapter 21 External Hazards [20], in relation to the aircraft impact safety case strategy and acceptance criteria for maintaining the integrity and cooling of the SFP.

Part B Chapter 23 Reactor Chemistry [21], in relation to the methodology, principles and approach to control of water quality in the SFP (chemistry, clarity and radioactivity levels), to ensure a controlled environment for the storage of Fuel Assemblies (FAs) and NFW components.

Part B Chapter 25 [22], in relation to arrangements and requirements for construction and commissioning of the fuel route.

Part B Chapter 26 Decommissioning Approach [23], in relation to design-for-decommissioning considerations, decommissioning options for SSCs, and any requirements on the functionality of SSCs to support or enable decommissioning activities.

The Fuel Route PSR also interfaces with wider topic areas within the SSEC:

- Preliminary Environment Report (PER) Chapter 1 Radioactive Waste Management Arrangements [24], in relation to arrangements for the management of spent fuel, any damaged fuel and NFW.
- PER Chapter 4 Conventional Impact Assessment [25], for example in relation to sustainability considerations.
- PER Chapter 5 Monitoring and Sampling [26], in relation to details regarding monitoring and sampling arrangements.
- PER Chapter 6 Demonstration of Best Available Techniques [27], in relation to the demonstration of BAT.
- Submission to NWS for an Expert View on packaging arrangements and disposability of spent fuel, any damaged fuel and NFW arising from the generic SMR-300 [28].
- Generic Security Report [29] and Preliminary Safeguards Report [30], in relation to security and safeguards arrangements for nuclear fuel.

Where such matters are relevant to the design basis of SSCs involved in fuel transport and storage are referred to in this chapter. An example is how safety and environmental factors are taken into account in a holistic, integrated manner in terms of how they contribute to ALARP and BAT, taking into consideration all relevant factors, alongside learning-from-experience, good practice and OPEX.

Security-by-design and safeguards-by-design principles are also taken into account in the generic SMR-300 design. For example, the optimised, integrated fuel route design for the SMR-300, with a small SFP located entirely within the CS/ CES, and utilising Holtec's proprietary proven dry fuel storage system, which is designed for safe, secure interim storage of spent fuel and NFW for 100 years plus, contributes to security considerations.

The generic SMR-300 design also makes a significant contribution towards sustainability. The design of the generic SMR-300 fuel route, alongside implementation of an optimised, integrated fuel management strategy across the entire lifecycle, makes a significant contribution in supporting application of the waste hierarchy. This is achieved by minimising the generation of wastes (including secondary radioactive wastes), minimising the use of construction materials and resources (including water and energy), and by maximising opportunities for materials recycling and recovery.

Holtec's dry storage system is also specifically designed to facilitate ease of retrieval and future management of spent fuel and NFW following the interim storage phase, enabling consignment to the most appropriate routes available in the future. This avoids the foreclosure of long-term management options that could contribute to sustainability, such as reprocessing for spent fuel, and the diversion of suitable NFW to potential alternative waste routes.

24.2 FUEL TRANSPORT AND STORAGE DESCRIPTION

This sub-chapter describes fuel handling operations and the Fuel Transport and Storage SSCs.

The following documentation has been used to provide a description of the generic SMR-300 fuel route and the dry storage system operations:

- SMR-160 Fuel Handling and Storage [31].
- Final Safety Analysis Report (FSAR) on the HI-STORM UMAX Canister Storage System [32].
- FSAR on the HI-STORM FW MPC Storage System [33].
- SMR-300 Purchase Specification - Fuel Handling Bridge Crane [34].
- SMR-300 Purchase Specification - Polar Crane [35].
- SMR-300 Purchase Specification - Reactor Auxiliary Building Crane [36].
- System Design Description (SDD) for Spent Fuel Pool Cooling System [37].

Documentation for the fuel route is still in development. This is typical at this stage of development (and is reflected in fuel route submissions from other RPs at Steps 1 and 2 of the GDA). As a consequence, the majority of the referenced documents for the fuel route are not within the Design Reference Point (DRP) [38]. SDDs are available for certain auxiliary systems that are key interfaces to the fuel route.

SMR-160 Fuel Handling and Storage [31], a key reference used in this Chapter for the fuel route, will be updated for the SMR-300 design during the next stage of development. The general principles, approaches and methodologies are not expected to change materially. Proposed changes for the SMR-300 are noted in relevant sub-chapters. Salient fuel route parameters set out in this Chapter are generally from SMR-300 references e.g. the capacity of fuel storage racks. A Spent Fuel Management Procedure will also need to be produced during the site-specific development phase, which is a standard requirement.

By contrast, detailed FSAR documentation is available for the UMAX system, which are incorporated within the DRP. This suite of documentation goes well beyond the level of detail that Requesting Parties (RP) typically provide at this stage of development. This reflects the fact that Holtec's proprietary UMAX dry storage technology is a proven, mature system that has been in-service at power station sites since 2015. It is licensed by the US Nuclear Regulatory Commission (NRC) for generic use in the US, including for potential deployment at a consolidated ISFSI proposed for the HI-STORE project in New Mexico. A variation of Holtec International's HI-STORM technology has been licensed in the UK for dry fuel storage at Sizewell B.

A planned update to the UMAX FSAR for SMR-300 fuel in the MPC-37 configuration for the generic SMR-300 will be available for the next stage of development, noting that the current assessments set out there-in for standard PWR fuel in an MPC-37 are expected to be bounding for SMR-300 fuel and configuration.

24.2.1 Fuel Design Data

Key design parameters concerning the fuel and fuel cycle are provided in Table 1, taken from the SMR-300 Plant Overview [39]. Key design parameters related to fuel transport and storage are provided in Table 2.

The design parameters set out in Table 1 and Table 2 are subject to change as the design evolves.

Table 1: Design Parameters for the Fuel and Fuel Cycle

Parameter	Value/ Description	Notes or Units
Fuel assembly model	Framatome GAIA 17 x 17	Manufactured by Framatome
Fuel material	UO_2	
Fuel enrichment	4.9 / 5.0%	% avg / % max
Refuelling cycle	[REDACTED]	Months
Refuelling outage	[REDACTED]	Days
Reactor design life	80	Years
Neutron energy spectrum	Thermal	
Fuel cladding material	M5	
Weight of one fuel unit	See DWG-15131, SMR-300 Fuel Assembly Design Interface Drawing [40]	
Main reactivity control	Soluble boron and Rod Cluster Control Assemblies (RCCAs)	
Soluble burnable absorber	Gd_2O_3	Integral to fuel format
Standard plant configuration	2 x units	
Core-average discharge burn-up	[REDACTED]	GWd/MTU
Maximum assembly average burn-up	[REDACTED]	GWd/MTU
Peak rod-average burn-up	[REDACTED]	GWd/MTU

A nominal [REDACTED] cycle length is indicated for the generic SMR-300 based on an equilibrium fuel cycle; an operator could choose to pursue an alternative fuel cycle of differing duration.

Table 2: Design Parameters for Fuel Transport and Storage

[REDACTED]

24.2.2 Top Level Plant Design Requirements

HI-2240251, SMR-300 Top Level Plant Design Requirements [41] describes the design philosophy and high-level requirements for the generic SMR-300; a number of these requirements are relevant to Fuel Transport and Storage.

24.2.3 Fuel Handling Operations

24.2.3.1 Transfer of New Fuel to Reactor

Based on a nominal [REDACTED] equilibrium fuel cycle, approximately one third of the FAs in the core will be discharged every refuelling outage. The refuelling process for the generic SMR-300 is similar to a conventional Boiling Water Reactor (BWR), given the orientation and close proximity of the SFP to the reactor. Placing the SFP adjacent to the Reactor Pressure Vessel (RPV) eliminates the complexity and maintenance associated with conventional PWR fuel up-ending and transferring equipment to facilitate transfer of fuel between the containment and a fuel handling building.

New Fuel Assemblies (NFAs) received for refuelling are removed one at a time from the shipping container and moved into the new fuel assembly inspection area of the RAB. After inspection, the accepted NFAs are stored in the NFSR in the NFV.

Prior to initiating the refuelling operation, the RCS is cooled down to refuelling shutdown conditions. The water level in the RPV is raised to the top of the RPV. The RPV head, along with the control rod drive mechanisms (CRDM), is removed using the Polar Crane and placed on the reactor head stand next to the reactor cavity. All the fuel in the reactor core is moved to the SFSR in the SFP using the Fuel Handling Bridge Crane (FHBC).

An MPC is placed inside a Holtec International Transfer Cask (HI-TRAC) and brought to the RAB door on a Vertical Cask Transporter (VCT). The HI-TRAC is lowered onto a Low Profile Transporter (LPT) which transports the HI-TRAC into the FHA of the RAB.

The NFV is in the FHA of the RAB. NFAs are loaded into the HI-TRAC one at a time, using the overhead crane in the RAB. The HI-TRAC is then transported on the LPT through the CS equipment hatch[4]. Inside the CS, the HI-TRAC is flooded with borated water and the Polar Crane lifts the HI-TRAC off the LPT and lowers it into the SFP. During refuelling operations, the SFP refuelling water level will be controlled procedurally during HI-TRAC placement into the SFP to prevent overflow.

The NFAs from the HI-TRAC are transferred to the SFSR and then moved into the reactor core using the FHBC (an alternative option involves transferring the NFAs from the submerged MPC direct to the reactor core using the FHBC). Fuel that will be used for another cycle is also returned to the reactor core from the SFSR using the FHBC.

24.2.3.2 Transfer of Spent Fuel to Dry Storage

The generic SMR-300 spent fuel dry storage system is part of the Holtec's optimised, integrated fuel management system. The generic SMR-300 utilises on-site interim spent fuel storage within the HI-STORM UMAX system, utilising an MPC-37 loaded with [REDACTED] Spent Fuel Assemblies (SFAs).

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[4] The CS is equipped with two hatches, the Equipment Hatch (EH) and the Personnel Hatch (PH). The EH is located at the operating floor and permits grade-level access to allow movement of equipment into and out of the containment during refuelling outages. The PH is used for personnel entry during refuelling outages.

Consistent with requirements relating to decay heat, burnup and cooling time, SFAs can be retrieved from the SFP for dry storage after as little as [REDACTED] of cooling in the SFP. It is assumed that dry storage campaigns will commence after [REDACTED] when the discharged SFAs of the first cycle will be removed from the SFP and processed for dry storage in the HI-STORM UMAX system, located at the on-site ISFSI. Dry fuel storage campaigns will be undertaken following every subsequent refuelling outage until all the SFAs are placed into dry storage.

Lifting, handling, processing and transportation equipment is designed to efficiently move SFAs from the SFP to the UMAX system sited at the on-site ISFSI. When a dry storage campaign is performed during the refuelling outage, the same HI-TRAC that was used to bring in NFAs will be used to move SFAs out of containment for processing for dry storage. After the NFAs have been removed, and the HI-TRAC is still within the SFP, the SFAs are loaded by FHBC into a submerged MPC shielded within the HI-TRAC. A thick stainless-steel lid is placed on the MPC using the Polar Crane. The HI-TRAC is removed from the SFP using the Polar Crane and placed on the LPT. The HI-TRAC provides shielding from the SFAs and structural protection for the MPC. The thick MPC lid and the water in the MPC also provide shielding.

The HI-TRAC is moved out of the CS the same way it entered and is placed in a dedicated area in the FHA of the RAB where the SFAs and the MPC are prepared for dry storage[5]. The MPC is drained, dried, backfilled with inert gas (helium) and welded shut to provide a pressure vessel boundary as a high integrity confinement barrier for nuclear material, for example for on-site transfer and during interim storage within the UMAX system. When complete, the HI-TRAC is moved outside of the RAB the same way it entered.

A VCT then lifts the HI-TRAC off the LPT and carries it to the on-site ISFSI. At the ISFSI, the HI-TRAC is positioned over the HI-STORM UMAX cavity. The bottom hatch of the HI-TRAC is removed and the MPC is lowered into place via a shielded mating device. The HI-TRAC is then removed from the area and the UMAX system closure lid is placed over the cavity to complete the dry storage process.

Each HI-STORM UMAX VVM provides storage of an MPC in the vertical configuration inside a cylindrical cavity located entirely below the top-of-grade of the ISFSI. The VVM, akin to an above ground over-pack, comprises of a Cavity Enclosure Container (CEC) and closure lid, as well as interfacing structures. On-site interim dry storage within the HI-STORM UMAX system is expected to be the storage means for spent fuel, any damaged fuel and NFW for at least the design life of the SMR-300. Based on previous GDA submissions by RPs for standard PWR fuel with similar characteristics, an interim storage period for spent fuel of 100 years is assumed, prior to retrieval for repackaging for disposal in the UK's planned GDF.

The full fuel route is summarised in a process diagram set out in Appendix A.

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[5] Processing of the spent fuel takes place in the FHA of the RAB. The RCA HVAC shall maintain a negative design pressure to prevent release of radioactivity [58].

24.2.4 Description of SSCs

The Fuel Transport and Storage SSCs comprise the equipment defined in Table 3 and Table 4. The supporting systems that have key interfaces with the fuel route are also described below.

Table 3: Fuel Route SSCs

Process / Location	SSC or Sub-system / Component	Operational Function
NFA receipt, inspection, storage and transport Designated areas in the FHA of the RAB	FHA RAB Bridge Crane	Transfer of NFAs between FHA RCA designated areas for receipt/ inspection, storage and loading (into an MPC/ HI-TRAC).
	NFV	Storage of NFAs, protects the NFSR.
	NFSR	Storage of NFAs.
	MPC	Transfer of NFAs into the CS.
	HI-TRAC transfer cask	Protective transfer cask for the MPC containing NFAs.
	LPT	Transport of the HI-TRAC with MPC containing NFAs into the CS on guided rails.
NFA transport, storage and handling for core loading CS/ SFP	LPT	Transport of the HI-TRAC with MPC containing NFAs into the CS on guided rails. Restrains the HI-TRAC for postulated seismic events.
	HI-TRAC	Protective cask for transfer of the MPC containing NFAs into the CS and into the SFP.
	MPC	Canister for transfer of NFAs into the CS and the SFP, and for storage submerged in the SFP within the HI-TRAC pending initial core loads.
	Polar Crane	Transfer of HI-TRAC with MPC containing NFAs into the SFP.
	FHBC	Transfer of NFAs from the MPC into the SFSR for storage, to enable loading of SFAs into MPC for a dry storage campaign.
	SFSR	Storage of NFAs removed from the MPC pending core loading, to enable loading of SFAs into MPC for a dry storage campaign.
	FHBC	Transfer of NFAs for core loading.
SFA transport, storage and handling CS/ SFP	FHBC	Transfer of FAs during refuelling to the SFSR in the SFP for storage and cooling (including FAs to be reloaded into the core using the FHBC).
	SFSR	Storage of FAs (including FAs to be reloaded into the reactor core).

Table 4: Dry Fuel Storage SSCs

Process / Location	SSC or Sub-system / Component	Operational Function
SFA transport, storage and handling CS	MPC	Canister for transfer, processing and confining SFAs, along with RCCAs, during a dry fuel storage campaign.
	HI-TRAC	Protective, shielded transfer cask for the MPC containing SFAs.
	FHBC	Transfer of SFAs into the submerged MPC within the HI-TRAC in the SFP, for retrieval for a spent fuel dry storage campaign.
	Polar Crane	Placing of a thick stainless steel lid on the MPC in the SFP. Transfer of the HI-TRAC with MPC containing SFAs out of the SFP onto the LPT.
	LPT	Transport of the HI-TRAC with MPC containing SFAs out of the CS on guided rails. Restrains the HI-TRAC for postulated seismic events.
SFA transport, processing and handling	MPC	Canister for transfer, processing and confining SFAs during a spent fuel dry storage campaign.
	HI-TRAC	Protective, shielded transfer cask for the MPC containing SFAs.

Process / Location	SSC or Sub-system / Component	Operational Function
Designated location in FHA of the RAB for MPC processing	LPT	Transport of the HI-TRAC with MPC containing SFAs out of the CS on guided rails to the spent fuel processing location in the FHA of the RAB RCA. Following drying of the spent fuel and sealing of the MPC, the LPT is used to transfer the HI-TRAC with MPC out of the RAB via the truck bay entrance. Restrains the HI-TRAC for postulated seismic events.
	FHD	Draining of the MPC, drying of SFAs and backfilling of the MPC using an inert gas at pressure (out of scope).
	Ancillary cooling and fluid management equipment	Cooling and fluid management during MPC processing (out of scope).
	Automatic Welding System (AWS)	Remotely welds the thick stainless steel lid of the MPC to the canister to create a sealed pressure vessel (out of scope).
SFA transport between the RAB and ISFSI Cross-site transfer	MPC	Canister for transfer and confining SFAs during a dry storage campaign.
	HI-TRAC	Protective, shielded transfer cask for the MPC containing SFAs.
	VCT	Loading of the HI-TRAC with MPC from the LPT onto the VCT for transport to the ISFSI for MPC downloading into the UMAX system VVM.
MPC downloading into a VVM of UMAX system for interim storage ISFSI	MPC	Canister for transfer and confining SFAs during a dry storage campaign.
	HI-TRAC	Protective, shielded transfer cask for the MPC containing SFAs.
	Mating Device	Enables safe downloading of an MPC from the HI-TRAC into a VVM in shielded conditions.
	VCT	MPC downloading, following removal of the HI-TRAC bottom hatch, into the VVM of the UMAX system using the mating device, followed by installation of the VVM tamper-proof lid for final storage configuration.
Interim storage of SFAs ISFSI	MPC	Canister for containment/ confinement of SFAs during the interim storage phase.
	DFC	Thin-walled canister that prevents dispersion of damaged fuel fragments and gross particulate within the MPC, whilst allowing natural circulation of helium over the SFAs for cooling.
	HI-STORM UMAX system	Interim storage of sealed MPCs containing SFAs in a passively safe state in an underground VVM of the UMAX system.

The processes, SSCs and operational functions described above are similarly applied to the management of redundant in-core components co-managed with SFAs, for example RCCAs inserted within FAs loaded within MPCs for a dry storage campaign, and NFW components loaded in prefabricated containers in NFWCs for processing for dry storage. Dry storage campaigns using the NFWC occur when sufficient NFW has been accumulated within designated cells within the SFSR to fill a canister.

SSCs involved in fuel transport and storage are described in the following sub-chapters.

24.2.4.1 Spent Fuel Pool

The SFP is located inside the CS. The SFP is a Seismic Category I structure and is sufficiently sized (with margin) to accommodate discharged fuel for the requisite minimum cooling period prior to transfer for processing for dry storage. In addition, the SFP includes a laydown area for a Holtec on-site shielded HI-TRAC transfer cask which is used to deliver new fuel to, and remove spent fuel from, the SFP.

The engineered features which contribute toward criticality control for the FAs stored in the SFSR within the SFP are discussed in Sub-chapter 24.4.3.9. SFP criticality is controlled by geometry and neutron absorbing materials, Metamic™, in the design of the SFSR. The SFSRs are Region 1 (flux trap design) racks which enables any FA to be placed in any cell (no credit is taken for fuel burnup in the criticality safety case) and include integral neutron absorbing material, Metamic™ [42]. The SFP water is borated and, although credit is not taken for the reactivity control of the soluble boron, this provides an additional margin of reactivity control (defence in depth). The use of Metamic™ in both the MPC fuel basket and SFSR applications has been evaluated by the NRC and is widely used in light water reactors (LWRs) as described in Neutron Absorber Material [43].

The SFP is designed with sufficient water coverage to ensure that personnel surrounding the SFP are exposed to radiation levels that are ALARP when a SFA is being moved to or from the SFSR. Collection of any leakage from the SFP, via refuelling floor drains or construction joints in the SFP liner, is routed via the Radioactive Drain System (RDS)[6] to the Liquid Radwaste System (LRW), as described in the SDD for the Liquid Radwaste System [44]. Leak detection equipment would be expected for the SFP liner to enable identify location of leakages, noting that details for the RDS are not yet available. Radiation monitoring is also provided on the refuelling floor.

The Equipment Floor and Drainage System Radiation Monitor measures the concentration of radioactive materials in the floor drainage discharge and is used as an indicator for system breaches (see PER Chapter 5 [26]).

Figure 1 below illustrates the location of the SFP within the SMR-300, adjacent to the RPV. The normal level for the SFP water is illustrated, along with the level when filled to the refuelling cavity level for refuelling; the Annular Reservoir (AR) can be seen between the CS and CES.

The SFP is described in SMR-160 Fuel Handling and Storage [31], in Drawing No. 10903, Spent Fuel Pool [45] and Drawing No. 11506, Spent Fuel Pool Liner [46].

[6] RDS design information is not within the DRP [38] as it was not available during the GDA timeframe, the case is presented in the Design Reference Point and GDA Scope Change Proposal No 1 [127] paper. The system's function is to collect the liquid effluent arisings from the generic SMR-300 and direct them to the correct location for treatment in the LRW. This includes the capability for segregation at source of the liquid effluents into the waste holdup tanks based on their conductivity levels. See Part B Chapter 13 [14] for further information on the RDS.

[REDACTED]

Figure 1: SFP, Refuelling Cavity and CS[7]

24.2.4.2 Spent Fuel Pool Cooling System

The SFC consists of the SFP, one SFC pump, one SFC heat exchanger, one demineraliser, one demineraliser filter, the Refuelling Water Storage Tank (RWST) and the RWST purification pump. The SFP is located inside the CS.

The main function of the SFC is to provide cooling and cleanup of the SFP during all modes of plant operation[8]. The system also maintains the water quality within specified chemistry, radioactivity and clarity limits. In addition to SFP cooling and cleanup, the SFC is used to transfer water to and from the RWST and Passive Core Makeup Water Tank (PCMWT) to support refuelling operations and to purify the water in the RWST and PCMWT.

The SFC maintains containment boundary integrity by performing a containment isolation function. The SFC piping that penetrates the containment is provided with Containment Isolation Valves (CIVs). The SFC has three containment penetrations: one penetration for the pump suction piping, a second penetration for the pump discharge piping and a third penetration for the PCMWT recirculation line. In case of a break outside the containment, the redundant isolation valves ensure containment isolation to maintain SFP water inventory.

Non-safety functions of the SFC are to:

- remove the maximum decay heat load from the SFP to maintain the temperature within normal operating limits;
- provide normal, non-safety makeup water to the SFP from the Demineralised Water System (DWS) to maintain the SFP water level, and from the Chemical and Volume Control System (CVC) for addition of borated water to the SFP;

[7] The impact of a change to the design of the CS, set out in a Prospective Design Change paper, is summarised in Part B Chapter 20 [19].

[8] The Residual Heat Removal System (RHR) provides backup cooling to the SFC. The RHR is cross connected with the SFC train so that RHR can function as back up to SFC during both normal and refuelling operations.

- maintain the SFP and RWST water quality within specified chemistry, radioactivity and clarity levels (a skimmer suction near the top of the normal SFP level ensures turnover of the water near the surface for cooling and cleanup); and
- transfer refuelling water between the PCMWT and SFP, and RWST and SFP.

The SFC purification flow rate for the normal SFP water volume provides a minimum of two SFP water changes in 24 hours using one 100% demineraliser. The required purification flow rate through the demineraliser is determined by the SFP water volume turnover requirement in Electric Power Research Institute (EPRI) Utility Requirements Document (URD) [47].

The cooling capacity of the SFC design accounts for the maximum expected heat load for the SFP, with additional margin. The maximum heat load for the SFP is based on decay heat generated by the accumulated maximum number of FAs stored in the SFP, including one full core placed in the pool after shutdown ([REDACTED] FAs in total). The time during the plant operating cycle at which the design full core off-load occurs is chosen to conservatively maximise SFC heat load (during a refuelling outage, about [REDACTED] hours after shutdown). For further details, see Spent Fuel Pool Cooling System Heat Exchanger Thermal Report [48].

During refuelling, a temporary demineraliser connected to the SFP will support the SFC demineraliser to handle the increased purification flow as described in Spent Fuel Pool Cooling System Flow and Line Sizing [49]. During refuelling, a temporary filter is used, in addition to the permanent demineraliser and filter, to clean the total SFP water volume [37].

The SFC is described in the SDD for Spent Fuel Pool Cooling System [37]. This describes the system functions, SSCs, operations, requirements, system boundaries and interfaces, codes and standards, and design parameters, and sets out a failure modes and effects analysis.

See Part B Chapter 5 [11] for further details on the SFC.

24.2.4.3 Spent Fuel Storage Racks

Spent fuel storage is provided by two SFSRs which are located at the bottom of the SFP. The SFSRs are stainless steel racks, with an integral neutron absorbing material. The SFSRs are sized to accommodate the number of FAs equivalent to 2.5 times a fully loaded SMR-300 core. The SFSR is designed to provide criticality control, allow decay heat removal by natural circulation and withstand the impact of FAs dropped onto the structure of the rack as described in Specification for SMR-160 Spent Fuel Storage Racks [50]. Further details can be found in SMR-160 Fuel Handling and Storage [31].

24.2.4.4 Polar Crane

The Polar Crane is a circular bridge crane. A single trolley with a main hoist, auxiliary hoist and maintenance jib crane travels the length of the bridge. The runway rail, access platforms, maintenance platforms, electrification, and controls are considered part of the Polar Crane.

The major lifting operations performed by the Polar Crane are for: HI-TRAC (loaded); RPV Head; CRDM Frame; Reactor Lower Vessel Internals; Reactor Upper Vessel Internals; Reactor Reflecting Rings; Reactor Coolant Pump Motor; Reactor Coolant Pump Internals; Reactor Core Barrel; other equipment and tools required to service the equipment in the CS; special lifts to be conducted during the construction of the generic SMR-300; and maintenance jib for maintenance and repairs of the Polar Crane.

Further details on the Polar Crane can be found in the SMR-300 Purchase Specification - Polar Crane [35].

24.2.4.5 Light Load Handling Machine and Fuel Handling Bridge Crane

The LLHM is located inside the CS and is designed to carry out operations for the movement of FAs, SFAs, RCCAs and Burnable Poison Rod Assemblies (BPRA), as follows: FAs from MPC to SFSRs; FAs from reactor core to SFSRs; SFA transfers from the reactor core to the SFSRs; SFA transfers from the reactor core to the MPC; SFA transfers from the SFSRs to the MPC and vice versa; transfer operations for RCCA movements; transfer operations for BPRA movements; and to support reactor disassembly/ reassembly during refuelling activities. Additionally, the LLHM will be used for movements of miscellaneous loads and tools to support the generic SMR-300 operations and maintenance processes within the SFP.

The LLHM system handles FAs and various loads in a safe and efficient manner. It operates like an overhead bridge crane, travelling along the length of the SFP via a rail or track system mounted on each side of the SFP. The main hoist and mast mounted on the trolley can move back and forth across the width of the SFP. This arrangement allows the gripper mounted to the mast to pick up and manoeuvre loads within the SFP.

A second auxiliary hoist and trolley operates on a separate beam or track, enabling it to traverse the length of the FHBC. This auxiliary hoist is designated for conducting miscellaneous maintenance work within the SFP.

The LLHM controls and facilitates both manual and semi-automatic movement of loads. In the semi-automatic mode, the operator can accurately input load positions, prompting the crane to automatically manoeuvre and travel to the designated location. A precision positioning system allows the operator to make minute adjustments to the load's position. Real-time monitoring of key crane parameters, including load weight, load elevation (including a mechanical backup to verify load elevation), bridge position and trolley position is provided.

Further details on the FHBC can be found in the SMR-300 Purchase Specification - Fuel Handling Bridge Crane [34].

24.2.4.6 Reactor Auxiliary Building Crane

The RAB Crane is a double girder overhead electric traveling bridge crane equipped with two hoists and trolleys. The RAB Crane will employ hooks for load handling. The main hoist will feature a double shank hook with powered rotation, while the auxiliary hoist will utilise a single shank hook with manual rotation. The RAB main hoist is designed for lifting operations involving CS equipment. The RAB auxiliary hoist is specifically engineered for handling NFAs.

The following list provides a general overview of the main loads that the RAB Crane is expected to transport (additional items or materials may be included in the scope of its lifting and transportation capabilities as per project requirements): SFC components; NFAs from NFSR; up-ending of NFAs and placement into the NFSR in the NFV; MPC Drying Equipment; MPC Welding Equipment; MPC into a HI-TRAC; HI-TRAC Shielding; HI-TRAC work platform; FHD; Reactor Coolant Pump (RCP)/ RCP Motor and Pump Internals; HI-TRAC on LPT; HI-TRAC movement to the dry storage preparation area; and any other equipment and tools required to service the equipment in the RAB.

Further details on the RAB Crane can be found in the SMR-300 Purchase Specification - Reactor Auxiliary Building Crane [36].

24.2.4.7 New Fuel Vault

New fuel is stored in the NFV in the RAB. The NFV is a seismically robust open box type structure which contains the NFSR. The NFV is normally dry and has openings at the bottom of each cell of the NFSR to permit drainage of unanticipated water that may enter the vault. Any water that enters the NFV drains to the LRW via the RDS. Radiation monitoring is provided for this area. The NFV is described in SMR-160 Fuel Handling and Storage [31].

24.2.4.8 New Fuel Storage Rack

The NFSR is designed with an array of cells connected to each other and to a baseplate. The NFSR is supported vertically by pedestals that rest on the bottom of the NFV. The NFSR is free-standing in the NFV. The NFSR is supported horizontally by lateral supports that extend to the walls of the NFV to preclude tipping or sliding.

The NFSR can store enough NFAs for at least two refuelling cycles, with margin. NFAs are physically precluded from being inserted into any location in the NFSR, other than those designated to store a NFA.

The NFSR is described in SMR-160 Fuel Handling and Storage [31] and Specification for SMR-160 New Fuel Storage Racks [51].

24.2.4.9 Multi-Purpose Canister

The MPC is a cylindrical stainless-steel canister which utilises a honey-comb fuel basket made of co-planar slotted plates of Holtec's proprietary neutron absorber Metamic-HT™ to provide positions for SMR-300 FAs, as described in Structural Acceptance Criteria for the Metamic-HT™ Fuel Basket [52] and Metamic-HT™ Qualification Sourcebook [42].

The MPC-37 containing [REDACTED] SFAs is planned for use in conjunction with the generic SMR-300 and is tailored to balance the SFP size inside containment with the plant refuelling operational needs. Each MPC provides sufficient capacity (with margin) to transport the nominal core batch size of new fuel during refuelling and discharge spent fuel for processing for dry storage after as little as [REDACTED] of cooling in the SFP.

Holtec's optimised, integrated fuel management solution, followed throughout the generating life of the generic SMR-300, minimises the accumulation of spent nuclear fuel (including any damaged fuel) inside containment. This paradigm changing approach ensures that only a low source term is maintained within the SFP, compared with a typical PWR. This strategy minimises the accumulation of SFAs within the reactor building, by enabling them to be transferred as soon as reasonably practicable to the ISFSI, where they are stored in passively safe conditions for the duration of the interim storage period.

The MPC is an integrally welded pressure vessel designed to meet the stress limits of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section III, Subsection NB. The MPC enclosure vessel defines the Confinement Boundary for the stored SFAs. The HI-STORM VVM storage over-pack provides structural protection, passive cooling and radiological shielding for the MPC.

During a refuelling outage, the new fuel needed for refuelling is taken from the NFV, loaded into an MPC within the HI-TRAC transfer cask, and then moved to the SFP. The MPC is also used to remove spent fuel from the SFP for placement into dry storage. When new fuel is loaded into the MPC, the MPC is dry. The MPC is flooded with borated water before it is lowered into the SFP.

Parameters for the MPC-37 are described in Final Safety Analysis Report on the HI-STORM FW MPC Storage System [33].

24.2.4.10 Holtec International Transfer Cask

The HI-TRAC is a protective transfer cask used to move the MPC and NFWC. The HI-TRAC is a steel, lead and water-shielded transfer cask which houses the MPC/ NFWC during on-site transfers prior to placement in the HI-STORM UMAX system. The HI-TRAC includes both structural and non-structural biological shielding. It provides shielding to workers during loading operations and protects the MPC/ NFWC from DBAs. The HI-TRAC is lifted by the two lift points located at the top of the cask.

The HI-TRAC is described in FSAR on the HI-STORM FW MPC Storage System [33].

24.2.4.11 Low Profile Transporter

The LPT is used to move a loaded or empty cask with the cask directly situated on a low-lying platform founded on a structurally robust frame such that an uncontrolled lowering of the cask is not credible. The LPT must be sufficiently low to ensure that the loaded cask can clear any door apertures when transferred out of buildings.

The LPT is illustrated in Drawing No. 10877 [53].

24.2.4.12 Vertical Cask Transporter

The VCT has the ability to raise or lower a cask or a canister with the built-in safety of a redundant drop load protection system. The VCT is used to lift the HI-TRAC off the LPT, transport it to the ISFSI pad area and lower it into the VVM.

24.2.4.13 Holtec International Storage Module Underground Maximum Safety Vertical Ventilated Module

The HI-STORM UMAX system is a spent fuel dry storage system consisting of any number of VVMs each containing one canister.

The UMAX system represents the state-of-the-art in dry storage technology; it is the most modern configuration of Holtec's dry storage system. It is licensed by the NRC for generic use. To date over 103 MPCs have been safely loaded in HI-STORM UMAX systems. The UMAX system is fully compatible with the range of Holtec's HI-TRAC transfer casks, MPCs, the DFC and the NFWC, and ancillary handling and transfer equipment such as the LPT and VCT. The VVM Components include the CEC and the Closure Lid.

The VVM is described in FSAR on the HI-STORM UMAX Canister Storage System [32]. A drawing showing the arrangement of the VVM is presented in Figure 2. A brief description of each part is provided in the following sub-chapters.

[REDACTED]

Figure 2: Key Constituent Parts of the HI-STORM UMAX VVM

24.2.4.14 Cavity Enclosure Container

The CEC consists of a thick walled shell integrally welded to a bottom plate. The top of the container shell is stiffened by a ring shaped flange which is also integrally welded. The constituent parts of the CEC are made of low carbon steel plate, with stainless steel as an optional material for some components. In its installed configuration, the CEC is interfaced with the surrounding subgrade for most of its height except for the top region where it is girdled by the ISFSI pad.

With the Closure Lid removed, the CEC is a closed bottom, open top, thick walled cylindrical vessel that has no penetrations or openings. Thus, groundwater has no path for intrusion into the interior space of the CEC. Likewise, any water that may conceivably be introduced into the CEC through the air passages in the top lid will not drain into the groundwater.

The CEC and interfacing structures isolate the MPC from groundwater. Site-specific design features can also be incorporated, if necessary, for example an Enclosure Wall is an optional structure and serves as a means to prevent water intrusion in the VVM subgrade space.

24.2.4.15 The Closure Lid

The Closure Lid is a steel structure filled with plain concrete that can withstand the impact of design basis missile impacts. Both the inlet and outlet vents are located at grade level. A set of inlet passages located on top of the CEC provide maximum separation from the large outlet passage which is located in the centre of the lid. Using a flue extension, the air exhaust from the outlet passage is set to be several feet higher than the inlet and prevents significant preheating of the incoming air.

24.2.4.16 Independent Spent Fuel Storage Installation

The ISFSI is a below grade concrete structure that facilitates the storage of MPCs in VVMs. The ISFSI structures include the ISFSI Pad, the Support Foundation Pad and the Subgrade and Under-grade.

The ISFSI is described in FSAR on the HI-STORM UMAX Canister Storage System [32]. A brief description of each part is provided in the following:

24.2.4.17 Independent Spent Fuel Storage Installation Pad

The ISFSI Pad serves to augment shielding, to provide a sufficiently stiff riding surface for the cask transporter, to act as a barrier against gravity-induced seepage of rain or floodwater around the VVM body as well as to shield against a design basis missile impact. The ISFSI Pad is a monolithic reinforced concrete structure that provides the load bearing surface for the VCT. The portion of the ISFSI pad adjacent to the VVM is slightly sloped and thicker than the rest of the ISFSI pad to ensure that rainwater will be directed away from the VVM.

The ISFSI pad and Closure Lid are designed to minimise the risk of surface water ingress via the ventilation inlet ducts [4]. Operational guidelines are also set out in the UMAX FSAR [32] for the removal of water/ debris from the VVM cavity following a surface flood event, which has no safety consequences for spent fuel storage.

Figure 3 below illustrates a HI-STORM UMAX system ISFSI Pad designed for 48 VVMs for dry storage of MPCs and NFWCs, as an illustrative example.

Figure 3: A HI-STORM UMAX System ISFSI Pad Layout

24.2.4.18 Support Foundation Pad

The Support Foundation Pad (SFPD) is the underground pad which supports the HI-STORM UMAX system and ISFSI. The SFPD on which the VVM rests must be designed to minimise long-term settlement. The SFPD and the under-grade must have sufficient strength to support the weight of all the loaded VVMs during long-term storage and earthquake conditions. As the weight of the loaded VVM is comparable to the weight of the subgrade which it replaces, the additional pressure acting on the SFPD is quite small.

24.2.4.19 Subgrade and Under-grade

The lateral space between each CEC, the SFPD and the ISFSI Pad is referred to as the subgrade and is filled with a Controlled Low-Strength Material (CLSM). Alternatively, "lean concrete" may also be used. CLSM is a self-compacted, cementitious material used primarily as a backfill in place of compacted fill. The space below the SFPD is referred to as the under-grade. It may be possible to utilise a recycled or low carbon material for the CLSM, contributing to sustainability considerations, subject to meeting all nuclear safety requirements. This potential design option could be considered during site-specific design.

24.2.4.20 Non-Fuel Waste Container

NFW components will be packaged non-encapsulated in a NFWC, which is fully compatible with the UMAX system for interim storage. The NFWC consists of an outer canister called an Enclosure Vessel (EV) which contains a free-standing steel basket for storing the NFW. Prefabricated container boxes loaded with NFW are stored inside the NFWC thus minimising voidage. Thin steel plates welded to the inside of the NFWC act as separators and guides during the loading of the container box. The NFWC is fitted with a lid which is bolted and welded to the shell, as opposed to the MPC which is an all-welded design.

Like the MPC, the stainless steel closure welds will ensure that the NFWC will remain leak-tight for the duration of its service life. The vent and drain port connections in the NFWC lid are compatible with the use of the FHD system, which is used to dewater and remove moisture from the NFWC. The canister is filled with helium to provide a non-aqueous and inert environment. Since moisture is removed from the system, the incidence of corrosion during the interim storage phase is not expected. The helium also acts as a medium for heat transfer.

Further details on the NFWC can be found in the Design Report on the HI-SAFE FW Non-Fuel Waste Storage System [54].

24.2.4.21 Damaged Fuel Container

Once cooled sufficiently in the SFP, any damaged FAs and fuel debris will be loaded into Damaged Fuel Containers (DFCs) emplaced within the MPC basket within the submerged HI-TRAC for a dry fuel storage campaign. The DFC has mesh screens on the top and bottom, to allow fluids (water or helium) to circulate to enable decay heat removal from the fuel, whilst preventing the dispersal of fuel debris and particulate within the MPC. The DFC will have a removable lid to allow the damaged fuel assembly to be inserted. In the SFP, a latch or other physical constraint will be used to ensure the lid remains in place. If also used to move FAs, DFCs will be designed for lifting with either the lid installed or with a separate handling lid [55].

Damaged fuel will be loaded into a DFC within an MPC in the submerged HI-TRAC for a dry fuel storage campaign for processing and transfer to the UMAX system at the ISFSI. Up to 12 DFCs containing PWR damaged fuel and/ or fuel debris may be stored in the MPC-37 in defined locations with the remaining basket cells containing undamaged FAs [55]. The number of DFCs with SMR-300 fuel that can be placed in the SMR-300 MPC-37 configuration utilising [REDACTED] MPC basket cells for FAs will be confirmed during the site-specific development phase.

24.3 FUEL TRANSPORT AND STORAGE CLAIMS, ARGUMENTS AND EVIDENCE

This chapter presents the fuel transport and storage aspects for the generic SMR-300 and therefore directly supports Claims 2.2.17 and 2.3.4.

Claim 2.2.17: SSCs which support operational fuel activities are designed to ensure safety functions and measures are delivered and radiation exposure and release of radioactivity are minimised ALARP.

Claim 2.2.17 has been further decomposed within Part B Chapter 24 into the following sub-claims:

Claim 2.2.17.1 contributes to the *Design* phase by defining the codes and standards that the Fuel Route SSCs will be assessed against.

Claim 2.2.17.2 is also important to the *Design* phase, by ensuring that Fuel Route SSCs are analysed using best practice engineering analysis methodologies.

Claim 2.2.17.3 is important to the *Safety Analyses* phase, by ensuring that a robust safety case assessment for the fuel route is undertaken and clearly interfaces with other PSR chapters.

Claim 2.2.17.4 then ensures fuel route SSCs achieve their design intent through quality Manufacturing and Installation processes, noting that the maturity of evidence for this claim will be limited at a PSR stage. Claim 2.2.17.4 also covers through-life operational maintenance aspects noting the overall approach to EIMT is provided in Part B Chapter 9 [12].

Claim 2.3.4: Spent fuel will be safely managed throughout the entire reactor lifecycle.

Claim 2.3.4 has been further decomposed to cover the development of an appropriate spent fuel management strategy and the NLR requirements and expectations for spent fuel management:

Claim 2.3.4.1 ensures that an appropriate spent fuel management strategy is developed early during the Design phase and is updated regularly to reflect the position in the plant lifecycle.

Claim 2.3.4.2 ensures that the SMR-300 is compliant with the NLR expectations for spent fuel management.

Table 5 shows the breakdown of Claims 2.2.17 and 2.3.4, and identifies in which chapter of this PSR these claims are demonstrated to be met to a maturity appropriate for PSR Revision 1.

Appendix B provides a full Claims, Arguments and Evidence mapping for Part B Chapter 24, which includes any lower level claims, arguments and evidence needed to support the Claims in the table above. This includes identification of evidence available at PSR Revision 1 and aspects for future development of evidence to support these claims beyond PSR Revision 1.

Table 5: Claims Covered by Part B Chapter 24

Claim No	Claim	Chapter Sub-chapter
2.2.17.1	Fuel Transport and Storage SSCs are designed using appropriate Codes and Standards, taking cognisance of RGP and OPEX.	24.4.1 Codes, Standards and Methodologies
2.2.17.2	Fuel Transport and Storage SSCs are designed using best practice analysis methodologies, taking cognisance of RGP and OPEX.	24.4.2 Design of SSCs
2.2.17.3	Fuel transport and storage operations deliver the required functionality and performance during normal operations and accident conditions.	24.4.3 Safety Assessment
2.2.17.4	Fuel Transport and Storage SSCs achieve the design intent through quality, manufacturing and installation and EIMT processes.	24.4.4 Quality, Manufacturing, Installation and EIMT
2.3.4.1	An appropriate spent fuel management strategy consistent with the lifecycle phase is maintained.	24.5.1 Spent Fuel Management Strategy
2.3.4.2	The NLR expectations for spent fuel management are addressed in an appropriate manner commensurate with the lifecycle phase.	24.5.2 Nuclear Liabilities Regulation Compliance

24.4 FUEL TRANSPORT AND STORAGE DESIGN AND SAFETY ASSESSMENT

Claim 2.2.17: SSCs which support operational fuel activities are designed to ensure safety functions and measures are delivered and radiation exposure and release of radioactivity are minimised ALARP.

Claim 2.2.17 concerns the design and safety assessment of relevant Fuel Transport and Storage SSCs and has been decomposed into four level four claims:

- Claim 2.2.17.1 – Covered within Sub-chapter 24.4.1.
- Claim 2.2.17.2 – Covered within Sub-chapter 24.4.2.
- Claim 2.2.17.3 – Covered within Sub-chapter 24.4.3.
- Claim 2.2.17.4 – Covered within Sub-chapter 24.4.4.

Each of these claims has been further decomposed into a number of arguments to addresses specific aspects of the design and operation of SSCs relevant to fuel transport and storage.

24.4.1 Codes, Standards and Methodologies

Claim 2.2.17.1: Fuel Transport and Storage SSCs are designed using appropriate Codes and Standards, taking cognisance of RGP and OPEX.

Claim 2.2.17.1 has been broken down into two arguments to capture the differencing source of codes and standards, namely:

- The codes and standards used within the US to develop the SMR-300 design to date.
- The codes and standards and RGP that will be utilised in the development of the generic SMR-300.

24.4.1.1 US Codes and Standards

Argument 2.2.17.1-A1: The Fuel Transport and Storage SSCs have been designed in accordance with good practice codes and standards within the US regulatory environment as required by the NRC.

Evidence:

- SMR-160 Fuel Handling and Storage [31].
- SMR-300 Purchase Specification - Fuel Handling Bridge Crane [34].
- SMR-300 Purchase Specification - Polar Crane [35].
- SMR-300 Purchase Specification - Reactor Auxiliary Building Crane [36].
- Specification for SMR-160 New Fuel Storage Racks [51].
- Specification for SMR-160 Spent Fuel Storage Racks [50].
- FSAR on the HI-STORM UMAX Canister Storage System [32].
- FSAR on the HI-STORM FW MPC Storage System [33].
- Design Report on the HI-SAFE FW Non-Fuel Waste Storage System [54].
- SDD for Spent Fuel Pool Cooling System [37].
- Spent Fuel Pool Cooling System Regulatory Compliance Report [56].
- SDD for Containment Ventilation System [57].

- Design Specification for Reactor Auxiliary Building [58].
- Containment Ventilation System Regulatory Compliance Report [59].

The SMR-300 is designed in accordance with NRC regulatory guidance and applicable US Code of Federal Regulations (CFR). The selection of codes and standards applied to the development and design of the SMR-300 is commensurate with the importance of the relevant safety functions delivered. This is imparted from the classification of SSCs based on NRC requirements.

SMR-300 SSCs are classified based on their importance to safety in accordance with SMR-160 Systems, Structures, and Components Classification Standard [60]. The classification assigned determines requirements for design and construction of the SSCs, including compliance to codes or standards. SSC safety classifications defined at PSR stage for the SMR-300 are based on the US approach. The application of US Category and Classification compared to UK Category and Classification is discussed in Part B Chapter 14 [15].

The standard design of the SMR-300 intends to comply with NRC requirements set out in Title 10 CFR Part 50 for the licensing of facilities [61]. General Design Criterion (GDC) 61, Fuel Storage and Handling and Radioactivity Control, set out in 10 CFR 50, Appendix A, General Design Criteria for Nuclear Power Plants, requires that fuel storage and handling systems be designed to ensure adequate safety under anticipated operating and accident conditions.

Specifically, GDC 61 requires: (1) periodic inspections; (2) suitable radiation shielding; (3) appropriate containment, confinement, and filtering systems; (4) residual heat removal capability consistent with its importance to safety; and (5) prevention of significant reduction in fuel storage inventory under accident conditions.

To augment these requirements, the SFP design basis is also covered by GDC 2, Design Bases for Protection Against Natural Phenomena; GDC 4, Environmental and Dynamic Effects Design Bases; and GDC 63, Monitoring Fuel and Waste Storage.

United States Nuclear Regulatory Commission (US NRC) RG 1.13, Spent Fuel Storage Facility Design Basis [62], provides current guidance regarding the design basis for spent fuel storage facilities. This regulatory guide endorses (with certain additions, clarifications, and exceptions) an older standard, Design Objectives for Light-Water Spent Fuel Storage Facilities at Nuclear Power Plants, American National Standards Institute (ANSI) Standard N210-1976/ American Nuclear Society (ANS)-57.2-1983 [63].

Regulatory Guide 1.26, Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants [64], and Regulatory Guide 1.29, Seismic Design Classification for Nuclear Power Plants [65], which are referenced in Regulatory Guide 1.13.

Title 10 CFR Part 72 on licensing requirements for the independent storage of spent fuel applies to the HI-STORM UMAX system and the MPC, and sets common safety objectives to ensure that: (1) doses are less than the limits prescribed in the regulations; (2) subcriticality under all credible conditions is maintained; (3) there is adequate confinement and containment of the spent fuel under all credible conditions; and (4) spent fuel is readily retrievable.

ISG-11, Rev 3, Cladding Considerations for the Transportation and Storage of Spent Fuel [66] sets out criteria to provide assurance of the integrity of the fuel cladding during transport, processing and interim storage during normal, off-normal and accident conditions, including criteria for maximum Peak Cladding Temperatures (PCTs) for FAs.

ISG-18, Rev 1, The Design and Testing of Lid Welds on Austenitic Stainless Steel Canisters as the Confinement Boundary for Spent Fuel Storage [67] addresses the design and testing of closure welds as an acceptable confinement boundary to demonstrate no credible leakage of radioactive material during normal and off-normal conditions.

EPRI's Advanced Nuclear Technology: Advanced Light Water Reactor URD [47] sets out RGP articulated as owner/ operator utility requirements for SMRs as performance-based risk-informed requirements. Chapters 1, 3, 7 and 8 have requirements with particular relevance to fuel transport and storage.

Key US codes and standards applied to the SMR-300, the UMAX system and the MPC are identified in Appendix C. The documents listed as evidence to Claim 2.2.17.1 have sections identifying the codes and standards applied to the design of specific SSCs.

Codes and standards relating to mechanical engineering are set out in the Mechanical Codes and Standards Report [68]. The gap analysis for Mechanical Engineering identifies additional UK requirements, for example for addressing the risks of dropped loads. A GDA Commitment (**C_Mech_094)** in Part B Chapter 19 [18] establishes that a review of UK RGP relating to the design of mechanical handling and lifting and drop load assessments will be undertaken before issue of the PCSR.

In addition, Part B Chapter 22 [69] [REDACTED] and identifies a GDA Commitment (**C_Inte_096**) [REDACTED].

24.4.1.1.1 Design Codes and Standards Applied to Fuel Transport and Storage SSCs

The principal fuel transport and storage design codes and standards for SSCs within the scope of this chapter are summarised in Table 6.

Table 6: Principal Design Codes and Standards applied to Fuel Transport and Storage SSCs

SSC/Technical topic	SMR-300 Codes and Standards
SFP Liner	ASME BPVC, Section III, Division 1, Subsection NE [70] ASME BPVC, Section III, Subsection NCA [71]
MPC	ASME BPVC, Section III, Subsection NB
HI-STORM Overpack (UMAX system VVM)	ASME BPVC, Section III, Subsection NF ACI-318-05 [72]
HI-TRAC VW Transfer Cask	ASME BPVC, Section III, Subsection NF
ISFSI	ACI-318-05 [72]
Reinforced concrete structures	ACI 349-13 [73]
Steel structures	ANSI/AISC N690-18 [74]
Seismic analysis	ASCE/SEI 4-16 [75]
Cranes	ASME NOG-1, Rules for Construction of Overhead and Gantry Crane (Top Running Bridge, Multiple Girder) [76].

It is important to reiterate that no novel or in-house (bespoke) design codes have been applied in the development and design of the generic SMR-300's fuel route. The principal codes and standards applied to the design of nuclear safety related Fuel Transport and Storage SSCs are nuclear-specific and are considered compatible with UK nuclear industry expectations.

24.4.1.1.2 Compliance with US Codes and Standards

The documentation identified as evidence set out the codes and standards applied to the design and development of the SMR-300, and the means by which specific Fuel Transport and Storage SSCs will comply with them.

These include the primary reference for fuel handling and storage, the specifications for new and spent fuel storage racks, the SDD for the SFC, and the purchase specifications for the FHBC, Polar Crane and RAB Crane. The SFC regulatory compliance report includes a compliance matrix relevant to certain fuel transport and storage operations.

The HI-STORM UMAX system, MPC and ancillary handling, processing and transfer equipment are in-service and have a complete suite of safety case documentation setting out the codes and standards relevant to the licensing of Holtec's spent fuel dry storage system proposed as a baseline for the generic SMR-300.

Relevant sections of the HI-STORM UMAX system FSAR [32] include:

- Chapter 2 (Principal Design Criteria) which provides a complete set of requirements premised on US codes and standards[9] - see Sub-sections 2.0.1 – 2.09 addressing general, structural, thermal, shielding, criticality, confinement, operations, acceptance tests and maintenance and decommissioning aspects.
- Chapter 4 (Thermal) which addresses compliance relevant to thermal performance, and Sub-section 4.7 (Regulatory Compliance) which concludes that all applicable regulatory requirements and guidance germane to the integrity of the stored spent fuel and the UMAX system are satisfied.
- Chapter 8 (Material Evaluation) which presents an assessment of the materials selected for use in the UMAX system and addresses, and Sub-section 8.14 (Regulatory Compliance) which concludes that all applicable regulatory requirements germane to suitability of materials for their intended functions are met.
- Chapter 9 (Operating Procedures) which describes the operations associated with placing a loaded MPC into a UMAX VVM cavity/ unloading an MPC from a VVM, and Sub-section 9.5 (Regulatory Compliance) which concludes that the regulatory guidelines germane to safety and As Low As Reasonably Achievable (ALARA) are fully met.
- Chapter 10 (Acceptance Criteria and Maintenance Program) which addresses compliance relevant to fabrication, inspection, testing and maintenance programmes for the HI-STORM UMAX system, and Sub-section 10.6 (Regulatory Compliance)

[9] Chapter 2 states that the underground VVM design, while not specifically mentioned in the regulatory guidance literature associated with implementing the requirements in 10 CFR 72, meets and exceeds the intent of the guidance. Chapter 2 also references the MPC FSAR [33] in relation to codes and standards for the MPC and ancillary handling, processing and transfer equipment – see the mapping table on Page 2-2 of [32].

which concludes that regulatory requirements pertaining to testing and maintenance, resolution of issues on adequacy and reliability, and for cask identification are met.

Relevant sections of the HI-STORM FW MPC Dry Storage System FSAR [33] include:

- Chapter 7 (Confinement) which describes the confinement design of the MPC and how it satisfies the requirements of the relevant US regulatory requirements and guidance. It determines that leakage from the MPC confinement boundary is not credible.

The HI-STORM UMAX system is licensed for generic use in the US. The NRC has issued Certificates of Compliance (CoC) in relation to the HI-STORM UMAX system, MPC and ancillary handling, processing and transfer equipment [77] [78]. These indicate compliance with all applicable US codes and standards, subject to fulfilling the requirements of 10 CFR 72 and provided the relevant SSCs are operated in accordance with the conditions in the CoC.

In the US, an Ageing Management Programme (AMP) has not been required as part of the initial 20 years duration licensing cycle of an ISFSI, and instead is a regulatory requirement for license extension following the initial licensed phase of 20 years. This requirement differs from that in the UK, where ageing management should be considered from an early stage of development. Despite this, an AMP is available for the UMAX system proposed for implementation at the HI-STORE Consolidated Interim Storage (CIS) Facility in New Mexico.

Appendix G sets out key excerpts relevant to ageing management from the Licensing Report for the HI-STORE CIS Facility [79], that serves as an example of US good practice for an AMP for a UMAX system. The proposed CIS Facility utilises UMAX modules similar to those for the SMR-300. An AMP for the UMAX system for the SMR-300 will be developed to meet UK requirements, including those outlined in Appendix H, at the site-specific stage and has been captured as GDA commitment (**C_MSQA_109**) in Part A Chapter 4 [7].

24.4.1.2 UK Codes and Standards/ Relevant Good Practice

Argument 2.2.17.1-A2: A review of the design of the Fuel Transport and Storage SSCs demonstrates alignment with UK codes and standards and with RGP/ OPEX from other GDA submissions.

Evidence:

- Holtec SMR-300 GDA Spent Fuel Management Strategy [4].

UK and international codes and standards relevant to NLR aspects of fuel transport and storage include:

- Office for Nuclear Regulation (ONR) Licence Conditions.
- ONR Safety Assessment Principles (SAPs).
- EA Environmental Principles.
- ONR, EA and NDA Guidance.
- NWS Packaging Specifications.
- International Atomic Energy Agency (IAEA) safety standards and Western European Nuclear Regulators Association (WENRA) Reference Levels.

Government Policy for Radioactive Substances Management and Decommissioning [80] sets out a UK-wide policy framework for managing radioactive substances including spent fuel and Higher Activity Waste (HAW) such as NFW.

Under the Energy Act 2008, operators of new nuclear power stations are required to have an approved Funded Decommissioning Programme (FDP) in place before construction of a new nuclear power station begins, and to comply with this FDP thereafter. FDP Guidance [81] sets out the principles that the Secretary of State will expect to see satisfied in the FDP.

NDA Inter-Industry Guidance on Storage [82] draws together RGP/ OPEX from across the UK nuclear industry to provide an integrated approach to the management of storage facilities, setting out principles, approaches and toolkits as an aid to current and prospective store operators. Appendix H sets out key points for ISFSI implementation in the UK.

Details of the codes and standards identified above are set out in Appendix D.

Summaries and details of UK expectations for the management of spent fuel, damaged fuel and NFW are set out in Appendices E and F of the Spent Fuel Management Strategy [4].

Further codes and standards applicable to radioactive waste management are set out in Part B Chapter 13 [14] e.g. in relation to liquid wastes, gaseous wastes and secondary wastes down-stream of fuel route SSCs.

24.4.1.2.1 Compliance with UK Codes and Standards/ Relevant Good Practice

A detailed comparison between US and UK codes and standards for NLR related aspects was not undertaken as many requirements for this Spent Fuel Management topic are UK specific e.g. interim storage and GDA packaging requirements. In addition, many of the regulatory expectations relate to good practice and OPEX from the UK's sole operating dry store at Sizewell B. The focus for NLR related aspects has thus been on demonstrating compliance with UK requirements, assessment against good practice and OPEX, and providing substantiation for identified differences in approach.

Spent Fuel Management Strategy [4] describes how Holtec's baseline strategies for the long-term management of spent fuel, any damaged fuel and NFW arising from the generic SMR-300, align with Government Policy [80] (see Appendix G of [4]) and Base Case assumptions set out in FDP Guidance [81] for new nuclear power stations (see Appendix H of [4]).

Spent Fuel Management Strategy [4] describes a small number of deviations from UK practices and the basis for these (see Section 6.1 of [4]). It includes a comparison between assumptions for the SMR-300 and those made by other UK developers and RPs (see Appendix E of [4]).

The review of fuel, transport and storage codes and standards is underpinned through justification regarding the gap analysis [83], taking into account differences between the US and UK regulatory approaches.

Sub-chapter 24.4.3.3 provides a discussion of the UK Safety Function Categories and SSC Classifications that will be applied to the Fuel Transport and Storage SSCs beyond step 2. At this stage, formal Design Basis Accident Analysis (DBAA), which determines the Safety Function Categories and SSC Classifications, has not been carried out for these SSCs. Once

DBAA has been carried out following Step 2 of the GDA, a comparison between the SSCs' design and the UK Category and Classification expectations will be conducted. Any gaps identified in this activity will be subject to an ALARP optioneering process.

Sub-chapter 24.4.3.5 summarises the basis for confinement of nuclear material during both generic SMR-300 operations and interim storage in the UMAX system, to demonstrate compliance with relevant UK codes and standards.

The design of the SFP, located below grade and entirely within the CS/ CES, provides enhanced protection against external hazards or missiles compared with a traditional PWR. It also ensures that fuel stored in the SFP is confined within the CS, as opposed to a typical fuel handling building at a PWR which is located outside of containment (see Appendix F). The siting of the SFP inside the CS is considered an example of good practice.

Sub-chapter 24.5 addresses UK requirements for a Spent Fuel Management Strategy aligned with UK policy and strategic frameworks (see Sub-chapter 24.5.1) and in relation to wider NLR requirements and expectations (see Sub-chapter 24.5.2). As noted in the Spent Fuel Management Strategy [4], whilst there are some differences with UK expectations (e.g. use of a single wall MPC, the NFWC and the DFC), these relate to aspects which have been incorporated based on Holtec's commitment to continuous improvement of its proprietary dry storage technology, based on international learning and experience. Extensive safety case underpinning for such aspects is set out in the UMAX and MPC FSARs [32] [33]. Further SSEC substantiation against UK requirements will be provided during post-GDA.

24.4.1.3 CAE Summary

Two arguments have been developed to substantiate the use of recognised, regulatory-aligned codes and standards Claim 2.2.17.1 to a level of detail commensurate to the current level of design development.

The first argument justifies the application of codes and standards as the foundation of the design basis of the generic SMR-300. These standards have been selected based on the alignment with US NRC requirements and international standards.

The second argument aims to demonstrate how the ongoing development of generic SMR-300 fuel transport and storage SSCs is aligned with UK regulatory expectations and UK GDA OPEX. Following Step 2 of the GDA and once a DBAA has been carried out, this alignment will be further substantiated through a comparison between the SMR-300 design and the UK Category and Classifications. Any gaps identified will be subject to an ALARP optioneering process.

Together, these arguments demonstrate that the codes and standards applied for the design and development of the generic SMR-300 design are underpinned by internationally accepted standards and UK expectations, to an appropriate level for a 2-Step GDA. At the site-specific stage, further confidence will be provided as the design is developed and further hazard identification and assessments are undertaken.

24.4.2 Design of SSCs

Claim 2.2.17.2: Fuel Transport and Storage SSCs are designed using best practice analysis methodologies, taking cognisance of RGP and OPEX.

Claim 2.2.17.2 has been decomposed into two arguments to cover the analysis methodologies used in the design of SMR-300 Fuel Transport and Storage SSCs. It considers:

- Safety Functions of Fuel Transport and Storage SSCs, according to the SMR Class and Seismic Category for each SSC.
- Analysis and Design Methodology to demonstrate that the design meets the requirements of the design codes and standards.

The categorisation and classification of SSCs and the differences between the US and UK approaches are outside the scope of this PSR chapter, and are covered within Part B Chapter 14 [15]. The design codes and standards that Fuel Transport and Storage SSCs are designed to are outlined within Table 6 and are not repeated within this sub-chapter for brevity.

24.4.2.1 Safety Functions of Fuel Transport and Storage SSCs

Claim 2.2.17.2-A1: The Safety Functions for Fuel Transport and Storage SSCs have been identified in accordance with best practice and will align with RGP and take due cognisance of OPEX.

Evidence:

- SMR-160 Fuel Handling and Storage [31].
- SMR-300 Purchase Specification - Fuel Handling Bridge Crane [34].
- SMR-300 Purchase Specification - Polar Crane [35].
- SMR-300 Purchase Specification - Reactor Auxiliary Building Crane [36].
- Specification for SMR-160 New Fuel Storage Racks [51].
- Specification for SMR-160 Spent Fuel Storage Racks [50].
- FSAR on the HI-STORM UMAX Canister Storage System [32].
- FSAR on the HI-STORM FW MPC Storage System [33].
- Design Report on the HI-SAFE FW Non-Fuel Waste Storage System [54].
- SDD for Containment Ventilation System [57].
- Design Specification for Reactor Auxiliary Building [58].
- SDD for the Spent Fuel Pool Cooling System [37].

The SMR classification system provides a means of identifying the extent to which SSCs provide safety-related functions and provides a way to designate applicable codes and standards to SSCs. The UK safety function categorisations and SMR classifications for the generic SMR-300 for Fuel Transport and Storage will be determined post-GDA. The mapping of US to UK classification systems is discussed broadly in Part A Chapter 2 [5].

The following provides an outline of the safety functions that the Fuel Transport and Storage SSCs for the SMR-300 are designed to meet and the US SMR Class/ Seismic Category applied. These are summarised in Table 8. The safety function categorisations and SSC classifications are described to reflect the current level of detail available. Where SSC classifications/ safety function categorisations are not provided this is because this level of detail has not been developed at this stage of the design for these SSCs, but in such instances the level of detail is judged to be sufficient for a Step 2 assessment. UK safety function categorisations and SSC classifications for the generic SMR-300 will need to be confirmed following more comprehensive DBAA during the next stage of development.

SFP and SFC

The purpose of the SFP is to store FAs under a sufficient level of water to protect operators from radiation and to aid in controlling the heat dissipated. All connections to the SFP are above the minimum required level (10ft, or 3.05m), preventing the pool from draining below the minimum permitted level. The SFC pump discharge piping includes a hole above the SFP minimum water level to prevent the pool from siphoning. The SFP is also fitted with both high and low level alarms for monitoring purposes during all modes of operation.

The SFP is designated as a SMR Class B and seismic category C-I to prevent operator exposure to radiation and ensure sufficient heat removal in the event of a design basis earthquake. The SFC pipework will penetrate containment and therefore performs a containment isolation function to maintain the containment pressure boundary. For this reason, the SFC system (inside the CS) is designated as an SMR Class B SSC.

Polar Crane and FHBC

Both the Polar Crane and FBHC are designated as Single Failure Proof cranes in accordance with NUREG-0554 and shall comply with the design criteria of ASME NOG-1 and ANSI/ANS-57. The main hoists are Type I components in accordance with ASME NOG-1. The auxiliary hoists will be designed as a Type III component.

The safety features associated with the cranes are detailed within the purchase specifications [34] [35] [36]. The hoist reeving systems will be a dual load path in the event that if one rope fails, the remaining rope(s) can carry the load. Various interlocks and switches will be provided to prevent a dropped load or crane fault such as over-load, over-speed and over-raise. The cranes will also be designed to be capable of stopping and holding the load during a seismic event. Both cranes are designated as SMR Class C SSCs.

SFSR

The SFSRs will store irradiated FAs in a configuration that provides sufficient radiation shielding and heat removal and that protects the fuel against damage and inadvertent criticality. The racks will be designed to withstand seismic forces and mechanical impacts due to a dropped fuel assembly and will employ fixed neutron absorbers for criticality control as discussed in Section 24.2.4.3. The Specification for SMR-160 Spent Fuel Storage Racks [50] details the specific safety functions that the SFSRs will be designed to meet. Mechanical impacts due to any other credible potential dropped loads will be considered as DBAA is further progressed during the next stage of project development.

The SFSRs are designed to prevent inadvertent criticality and therefore are specified as an SMR Class A SSC and are designated as a Seismic Category I SSC.

NFV / NFSR

The NFSR will store fresh FAs within the NFV. The racks will be designed to withstand seismic forces and mechanical impacts due to dropped loads and will employ fixed neutron absorbers for criticality control as discussed in Section 24.2.4.8. The NFSR is specified as an SMR Class A SSC and is designated as a Seismic Category C-I SSC.

MPC

The MPCs will ensure long-term confinement of all radioactive materials for all normal and design basis accident conditions. The confinement safety function of the MPC is verified through pressure testing and helium testing as well as rigorous weld examination. The redundant closure of the MPC is provided by the MPC closure ring welds which provide a second barrier to the release of radioactive material from the MPC internal cavity.

Criticality control is maintained by the geometric spacing MPC honey-comb fuel basket made of co-planar slotted plates of Holtec's proprietary neutron absorber Metamic-HT™. Any failed fuel is stored in a designated cell for damaged fuel within the SFSR. Failed fuel is packaged in a DFC contained within an MPC, for transfer from the SFP to the HI-STORM UMAX system for interim storage. The DFC is designed to prevent the dispersal of fuel fragments or particulate within the MPC during interim storage.

ISFSI

The HI-STORM UMAX system is designed to provide enhanced protection against threats from environmental phenomena such as hurricanes, tornado borne missiles, earthquakes, tsunamis, fires and explosions. The performance is assured by a combination of the performance of the ISFSI structure and pad, the VVM, the closure lid arrangement and the CEC (see the UMAX FSAR, Section 2.4, page 195 of [32]).

The UMAX FSAR demonstrates the assessment of the performance of the storage system to these design basis accidents. Where appropriate, for each loading type, a bounding value is selected in this FSAR to ascribe an additional margin of safety for the associated loading events. Such bounding loads are referred to as Design Basis Loads (DBL).

The VVM system is designed such that the radiation shielding in the storage system does not degrade under normal and off-normal conditions of storage, the system does not deform under credible loading conditions in a manner that would jeopardise the subcritical state of the storage system or ready retrievability of the MPC, and the MPC maintains confinement of radiological matter (see the UMAX FSAR, Section 2.4, page 195 of [32]).

The CEC shell is subject to performance-based limits, which require that the deformation of the CEC does not prevent MPC retrievability, does not cause loss of MPC confinement and that the system remains subcritical. This is accomplished by demonstrating that after the seismic event, permanent ovalisation of the Container Shell does not result in a geometry that precludes retrievability of the MPC and that the impact loadings on the MPC due to its rattling inside the CEC do not cause a breach of the MPC confinement boundary (see UMAX FSAR, Section 2.2.3, page 199 of [32]).

HI-TRAC

The HI-TRAC transfer cask provides shielding to maintain occupational exposures ALARP and protects the MPCs within from hazards.

LPT and VCT

The LPT and VCT will be designed to safely transport the HI-TRAC containing the MPC or NFWC and prevent dropped loads/ impacts.

The high-level safety functions assigned to the key SMR-300 Fuel Transport and Storage SSCs, along with their corresponding SMR classifications are summarised in Table 7.

Table 7: High Level Safety Functions for Fuel Transport and Storage SSCs

SSC Name	SSC Code	SMR Class	High-Level Safety Function
SFP and SFC	SFP SFC	B	Radiological protection. Containment integrity.
SFSR	SFSR	A	Protect against internal/ external hazards Criticality control.
FHBC	FHBC	C	Prevent dropped loads.
Polar Crane		C	Prevent dropped loads.
NFV	NFV	To be confirmed	Protect against internal/ external hazards.
NFSR	NFSR	A	Protect against internal/ external hazards. Criticality Control.
MPC	MPC	To be confirmed	Containment/ confinement of radioactivity. Radiological protection. Criticality control.
HI-TRAC Transfer Cask	HI-TRAC	To be confirmed	Radiological protection. Protect against internal/ external hazards.
LPT	LPT	To be confirmed	Prevent dropped loads.
VCT	VCT	To be confirmed	Prevent dropped loads.
VVM of UMAX	VVM	To be confirmed	Decay heat removal. Radiological protection. Protect against internal/ external hazards.
ISFSI	ISFSI	B	Protect against internal/ external hazards. Radiological protection.

Table 8 provides the classification of the civil Fuel Transport and Storage SSCs within the scope of this PSR chapter and their associated seismic category.

Table 8: SMR Class and Seismic Categories for Structures

Structure	SMR Class	Safety Classification	Seismic Category
SFP	B	Safety-related	C-I
RAB	C	Safety-related	C-I
ISFSI	B	Safety-related	C-I
SFSR	A	Safety-related	C-I
NFSR	A	Safety-related	C-I

24.4.2.2 Analysis and Design Methodology

Claim 2.2.17.2-A2: The analysis and design methodologies to be applied to Fuel Transport and Storage SSCs are in line with RGP and take due cognisance of OPEX.

Evidence:

- SMR-160 Fuel Handling and Storage [31].
- SMR-300 Purchase Specification - Fuel Handling Bridge Crane [34].
- SMR-300 Purchase Specification - Polar Crane [35].
- SMR-300 Purchase Specification - Reactor Auxiliary Building Crane [36].
- Specification for SMR-160 New Fuel Storage Racks [51].
- Specification for SMR-160 Spent Fuel Storage Racks [50].
- FSAR on the HI-STORM UMAX Canister Storage System [32].
- FSAR on the HI-STORM FW MPC Storage System [33].
- Design Report on the HI-SAFE FW Non-Fuel Waste Storage System [54].
- SDD for Containment Ventilation System [57].
- Design Specification for Reactor Auxiliary Building [58].
- SDD for the Spent Fuel Pool Cooling System [37].

Fuel Transport and Storage SSCs are designed and analysed for normal operation loads and design basis loads arising from internal hazards, external hazards and internal plant faults.

The design methodologies applied to the Fuel and Transport SSCs are presented in the Design Basis Report for Mechanical SSCs [84] and the Design Basis Report for Nuclear Classified Civil Engineering Structures [85]. The information available on the analysis and design methodologies applied to SSCs are limited at this stage and this sub-chapter provides a summary of the information available.

The design methodologies for the mechanical Fuel Transport and Storage SSCs are set out in Mechanical Engineering Design Basis Report [84]. The suitability of the codes and standards used for the design, analysis and substantiation of mechanical SSCs are outlined Part B Chapter 19 [18]. The mechanical engineering aspects of the UMAX system are considered through the methodologies used to substantiate the UMAX system. Also considered are the methodologies applied to the design of SFC aspects such as tanks, pumps and filters.

It is noted that analysis and design methodologies for aspects such as the cranes, LPT, VCT and storage racks will need to be defined when further information is available.

The design methodologies for the civil Fuel Transport and Storage SSCs are set out in Design Basis Report for Nuclear Classified Civil Engineering Structures [85]. This considers the load combinations and generic design parameters to be considered for safety related civil structures, including seismically qualified SSCs. The suitability of the codes and standards used for the design, analysis and substantiation of civils SSCs are outlined in Part B Chapter 20 [19].

24.4.2.2.1 Materials

The materials used for the design and construction of Fuel Transport and Storage SSCs are specified according to the requirements of the applicable design codes and standards.

The material selection for the generic SMR-300 will be carried out through a systematic approach by considering the behaviour of equipment in the manufacturing, operation, inspection and maintenance stages, as well as previous OPEX in similar environments. The evaluation of material suitability will involve characterising the applicable environment, identifying potential degradation modes and assessing potential hazards that could impact the continued effectiveness of the selected material.

Stainless steel, for which corrosion-resistant characteristics are well established in the nuclear industry, is expected to be widely used for Fuel Transport and Storage SSCs. Any materials other than stainless steel used will be chosen to address specific needs such as strength, aversion to fatigue, and / or low corrosion susceptibility. The materials used will be evaluated for susceptibility to known ageing and degradation mechanisms.

<u>MPC</u>

The MPC shall be constructed from stainless steel. The structural material of the fuel basket shall be Metamic-HT™. The materials used in the construction of the MPC are compatible and qualified for the facility environment. Materials that may contact the surface of FAs and other materials are corrosion resistant and will not contaminate the FAs or SFP environment [31].

<u>HI-TRAC Transfer Casks</u>

The HI-TRAC is a steel, lead and water-shielded transfer cask [39]. The materials used in the construction of the HI-TRAC are compatible and qualified for the environment of the facility. Materials that may contact the surface of FAs and other materials are corrosion resistant and will not contaminate the FAs or SFP environment [31].

<u>NFSR</u>

The NFSR structural and sheathing material shall be stainless steel. The neutron absorber in the rack shall be Metamic™. All materials used in the construction of the NFSR are compatible for the environment of the NFV. Surfaces that may come into contact with the NFAs shall be made of stainless steel. Other materials of the NFSR shall be corrosion resistant to prevent contamination of the NFV [31].

<u>SFSR</u>

The SFSR structural and sheathing material shall be stainless steel. The neutron absorber in the racks shall be Metamic™. Other materials shall be corrosion resistant and shall not contaminate the FAs or SFP environment. All materials used in the construction of the SFSR shall be compatible with the storage environment of the racks [31].

<u>HI-STORM UMAX System</u>

All exposed surfaces of the VVM components shall be made from stainless steel alloys that are readily painted. The primary shielding materials used in the UMAX system shall be plain concrete, reinforced concrete and steel. The use of plain concrete for shielding purposes in the underground VVMs shall be limited to the Closure Lid. The materials used in the UMAX system shall be examined to establish that these materials do not participate in any chemical or galvanic reactions when exposed to the various environments during all normal operating conditions and off-normal and accident events [32].

ISFSI Sub-grade

CLSM or lean concrete will be used as the material infill for the Sub-grade between each CEC, the SFPD and the ISFSI Pad. The American Concrete Institute (ACI) 229R-13 [86] notes several terms, such as flowable fill, unshrinkable fill, controlled density fill, flowable mortar, flowable fly ash, fly ash slurry, plastic soil-cement and soil-cement slurry to describe CLSMs. ACI 116R-00 [87] defines lean concrete as a material with low cementitious content.

CLSM and lean concrete are also referred to as Self-hardening Engineered Subgrade (SES).

SFP

The SFP shall be comprised of reinforced concrete, with a stainless steel liner [31].

NFWC

The internal and external surfaces of the NFWC are made of austenitic stainless steel to prevent corrosion during long-term storage [55].

DFC

The DFC shall be fabricated from structural aluminium or stainless steel [32].

RAB Crane

The materials used in the manufacture of parts of the crane and its components shall comply with the requirements of ASME NOG-1, Section 5000 and ANSI/ANS 57.1. Exposed aluminium materials shall be minimised [36].

FHBC

The materials used in the manufacture of parts of the FHBC and its components shall comply with the requirements of ASME NOG-1, Section 4000. Exposed aluminium materials shall be minimised [34].

Polar Crane

The materials used in the manufacture of parts of the Polar Crane and its components shall comply with the requirements of ASME NOG-1, Section 4000. Exposed aluminium materials shall be minimised [35].

24.4.2.3 CAE Summary

Two arguments have been developed to substantiate Claim 2.2.17.2 to a level of detail commensurate to the current level of design development.

The first argument aims to demonstrate that while UK-specific categorisations will be finalised post-GDA, safety functions of relevant SSCs have been defined based on the current SMR classification system and seismic categorisation which is appropriate for this stage of development.

The second argument reviews the suitability of the codes and standards used for the design and analysis methodologies applied to Fuel Transport and Storage SSCs. These codes and standards include guidance and expectations from the US NRC, ONR and design and analysis codes from US professional engineering organisations (ACI, ANSI/ AISC, ASCE/ SEI and ASME). Beyond Step 2, following the undertaking of further hazard identification and assessments, further evidence shall be developed to ensure continued compliance.

24.4.3 Safety Assessment

Claim 2.2.17.3: Fuel Transport and Storage operations deliver the required functionality and performance during normal operations and accident conditions.

Argument 2.2.17.3-A1: SSCs which support Fuel Transport and Storage ensure faults and hazards arising from failures of the SSCs are minimised. Safety functional requirements are identified and are adequately satisfied by the SSCs in the generic SMR-300 design.

Evidence:

- Preliminary Fault Schedule (PFS) Report [88].
- SMR-300 Fuel Storage and Transport Route HAZOP Report [89].
- Safety Assessment Handbook [90].
- Undertaking safety function identification and subsequent categorisation and classification exercises.
- Consequence Assessments.

This sub-chapter describes the approach to be adopted for the safety assessment of the generic SMR-300 fuel route.

The following topics are discussed in the context of Part B Chapter 24 and reference made to the PSR Chapter(s) where these topics are addressed in detail:

- Hazard Identification.
- Key Hazards and Safety Measures.
- Categorisation and Classification of SSCs.
- Safety Functional Requirements
- Confinement
- Internal Hazards.
- External Hazards.
- Radiological Protection (including Shielding Design).
- Criticality Assessment.
- Normal Operation Dose Assessment.
- Design Substantiation.

It is recognised that at this stage the safety assessment carried out on the Fuel and Transport SSCs has not been comprehensively completed for the generic SMR-300 design. Post Step 2, a complete safety analysis of the Fuel Transport and Storage SSCs will be completed, comprising a comprehensive hazard identification phase, a screening / bounding phase of identified initiating events, an assessment phase using relevant assessment methods, and an ALARP optioneering phase to address any potential gaps in the design against safety requirements.

Further details on the overall safety assessment process for the generic SMR-300 can be found in Part B Chapter 14 [15].

24.4.3.1 Hazard Identification

Hazard identification includes an assessment of the available RGP and OPEX available from other PWR projects. The RGP and OPEX derived listings for the fuel route are based on a review of the fault analysis for other PWR projects, in particular, those which have progressed through the UK's GDA process including the European Pressurised Water Reactor (EPR), the AP-1000, the HPR-1000 and other SMR projects.

A Hazard and Operability Study (HAZOP) 1 has been performed for the Fuel Storage and Transport SSCs and operations [89]. The purpose of a HAZOP is to carry out a systematic review of activity nodes with a multidisciplinary group using specific keywords to identify hazards associated with the design and associated activities. The following operations of the fuel route SSCs for the SMR-300 design are the subject of the HAZOP Study:

- Receipt of NFAs into the FHA of the RAB.
- Storage of NFAs within the NFSR within the NFV.
- Receipt of NFAs in the CS and transfer of the MPC within the HI-TRAC into the SFP.
- Transfer of FAs to and from the reactor core.
- Storage of FAs within the SFSR in the SFP.
- Transfer of SFAs in the MPC within the HI-TRAC out of the SFP.

The output of the HAZOP has been sentenced into the Consolidated Fault Listing (CFL) which is contained within the PFS [88].

Further HAZID on the fuel handling and storage operations will be carried out as the design develops for site specific application. Additional HAZOP studies will be supplemented by other recognised HAZID techniques such as Failure Modes and Effects Analysis (FMEA), Master Logic Diagrams (MLD) and Systems Theoretic Process Analysis (STPA).

24.4.3.2 Key Hazards and Safety Measures

The HAZOP identified 108 individual hazards associated with the Fuel Transport and Storage SSCs. The following is a high-level summary of the key hazards identified within the HAZOP, the full list can be found in the SMR-300 Fuel Storage and Transport Route HAZOP Report [89].

- Loss of containment of nuclear material e.g. a breach of SFAs.
- Increased external doses to operators e.g. inadequate shielding between operators and SFAs.
- A criticality accident due to loss of safe configuration of fissile material.

Various dropped loads/ crane faults associated with the Polar Crane, FHBC or the RAB crane are identified with initiating events including operator errors, crane control system faults or mechanical failures. Faults associated with the Polar Crane and FHBC will have the most significant consequences as there is the potential to damage SFAs resulting in a radiological release. There is also the potential for dropped loads to result in loss of water inventory within the SFP and result in increased doses to operators.

The safety measures which prevent or mitigate these faults are not yet confirmed; however, these will involve claims on nuclear qualified cranes (including seismic qualification), withstanding of SSCs to impacts, annual EMIT inspections and procedural controls. The choice of load handling routes will avoid lifts over nuclear significant SSCs, where reasonably practicable. Confirmation of safety measures will follow more comprehensive DBAA during the next stage of development.

It is noted in the HAZOP that a high-level hazard assessment will need to be undertaken to consider the consequences associated with radiological releases within containment during outages, as it has not been confirmed whether the two hatches between the CS and RAB will be interlocked during significant lifting operations (e.g. lift of the HI-TRAC with an MPC containing [REDACTED] SFAs using the Polar Crane at the start of a dry fuel storage campaign) to maintain the containment boundary. This will be determined as further safety work is carried out and as more design detail becomes available post-GDA.

[REDACTED]

The potential for a criticality is identified during movement of FAs. These are typically associated with operator errors or mechanical or structural failures associated with fuel handling. The design of the SFSR (see Sub-chapter 24.4.3.8), including neutron absorbers within them, will mitigate the risks but it is noted that criticality assessments will need to be carried out to UK requirements. The designation of the SFSR as Class A SSCs will mitigate the potential for failure. Criticality safety is covered in Part B Chapter 10 [13].

The layout of the generic SMR-300 with respect to fuel handling and storage is designed to control risk. Spent fuel storage will be either within the SFP inside the CS, or within sealed MPCs within the UMAX system. This arrangement ensures control of the source term, provides a high level of protection to the fuel and minimises (or avoids) the risk of radioactive releases to the environment.

A number of features or aspects of the fuel route and dry fuel storage system design relate to layout considerations and contribute to ALARP and / or BAT. Examples are set out in Table 19, Appendix F. These include the simplified fuel route design compared with a typical PWR, which contributes to minimising the number of required fuel handling and transfer operations. The regular dry fuel storage campaigns following a sufficient period of cooling enables the transfer of spent fuel to passively safe dry storage as soon as reasonably practicable (limiting the source term within the compact generic SMR-300 SFP).

24.4.3.3 Categorisation and Classification of SSCs

As set out in Part A Chapter 2 [5], the approach adopted for UK deployment is centred around demonstrating equivalency between the generic SMR-300 design, and UK categorisation and classification expectations. This is achieved through the application of formal safety assessment techniques, which are consistent with UK context expectations. These safety assessment techniques are developed to identify a comprehensive set of UK aligned safety functions and associated safety measures, and to demonstrate that radiological risks are tolerable and ALARP. Safety assessments were conducted during GDA Step 2 through the development of a PFS and a limited set of UK DBAA. Further details of the approach and current status of the PFS and UK DBAA are provided in Part B Chapter 14 [15].

In GDA Step 2, UK DBAA was applied to a selection of 'in-reactor' faults. UK DBAA has not yet been applied to the Fuel Transport and Storage related faults and therefore UK equivalent categories and classifications have not yet been formally derived for the SSCs in scope of this Chapter.

[REDACTED]

A GDA Commitment is raised in Part B Chapter 14 (**C_Faul_103**) [15] to progress and complete UK DBAA for the SMR-300 which will include all fuel route related faults.

The radiological hazards identified in Sub-chapter 24.4.3.2 are addressed more specifically in in Part B Chapter 10 [13] which considers the provision of shielding.

Any requirements on operator actions to provide safety functions will need to be substantiated by the Human Factors discipline post-GDA. The approach to substantiation is provided within Part B Chapter 17 [17].

Confidence that completion of the safety assessment will not lead to fundamental changes to the fuel handling and fuel route design, and that a future demonstration can be provided that risks are reduced to ALARP, can be based on the following:

- The adoption of defence-in-depth in the fundamental design principles of the generic SMR-300, which is applied across the development of the design.
- The prevention of fuel damage for spent fuel contained within the SFP via passive means, such as evaporative cooling and gravity fed make-up.
- The decision to site the spent fuel pool within the containment volume, eliminating the need for a separate cooled fuel building.
- Dose constraints and numerical targets adopted by Holtec at the design stage, in relation to normal operation and accident conditions, to ensure radiological protection of both on-site workers and members of the public.
- The application of good practice in the development of the design, particularly regarding compliance with ALARA for normal operations doses.
- A commitment to undertake a robust demonstration of the fault tolerance of the plant and of the effectiveness of its safety measures, by application of a UK context approach to hazard identification and fault assessment.
- The mature design substantiation provided within the UMAX and MPC FSARs [37] [38] which demonstrate that the MPC ensures confinement of nuclear material for all normal and design basis accidents.
- The Holtec International dry store has been successfully licensed in the UK for Sizewell B and is proposed for the nuclear new build project at Hinkley Point C.

24.4.3.4 Safety Functional Requirements

At this stage Safety Functional Requirements (SFRs) have not been specifically identified for the Fuel Transport and Storage SSCs. SFRs will be developed as part of normal design development, in accordance with the process set out in the Safety Assessment Handbook [90].

24.4.3.5 Confinement of Radioactive Material

There are multiple confinement barriers within the SMR-300 design to prevent release of radioactive material to the environment under both normal and design basis accident conditions.

Some SSCs within the scope of this Chapter provide a direct confinement safety function, such as the MPC, while SSCs such as the CS considered in Part B Chapter 20 [19] for its civils aspects respectively, contribute to the confinement safety function. This sub-chapter provides a holistic overview of the confinement barriers within the design.

<u>Confinement during SMR-300 Operations</u>

Confinement barriers for radioactive material during normal operations, abnormal operating occurrences (AOO) and accident conditions are the fuel cladding and the pressure boundaries provided by the RCS and the CS, as identified in SMR-300 Acceptance Criteria for Deterministic Safety Analysis [91]. See Part B Chapter 2 [9] for details on the confinement barrier provided by the fuel cladding, Part B Chapter 1 [8] on the confinement barrier provided by the RCS pressure vessel boundary and Part B Chapter 20 [19] and Part B Chapter 19 [18] for civils/ mechanical aspects of the CS confinement barrier.

The CS is designed to withstand high pressure and temperature release caused by postulated accidents, as well as shield against radiation and prevent leakage of radioactive material[10].

Design requirements for the CS are set out in Design Specification for the Containment Structure [92]. These include the requirement of 10 CFR 50 Appendix A, General Design Criteria (GDC), GDC 50, which states that the reactor Containment Structure (including penetrations), shall be designed to accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from any Loss of Coolant Accident (LOCA).

The CS is designed to maintain containment integrity under both normal operating and DBA conditions in accordance with 10 CFR 50 Appendix A, GDC, GDC 16. Leak-tightness of the CS must be tested at regular intervals during the life of the plant in accordance with the requirements of 10 CFR Part 50, Appendix J. The design specification for the CS [92] states that appropriate instrumentation should be incorporated inside containment for conducting

[10] The safety functions of the CS are to: (1) prevent or limit the release of radioactive material in all operational states and DBA conditions throughout the life of the plant (in accordance with GDC 16); (2) house the containment internal structures; and (3) provide heat removal and pressure reduction functions, as per the Passive Containment Heat Removal System (PCH).

such periodic leak tests. This should include instrumentation for monitoring pressures, temperatures, humidity and system flow rates for estimating leak rate.

CS Hatches for personnel and equipment are provided. Both CS Hatches are designed to maintain containment integrity under design basis conditions, including pressure, temperature, and radiation, to prevent or limit the release of radioactive material in all operational states and DBA conditions throughout the life of the plant [92]. When one or both of the CS Hatches is open for an extended period of time, for example during refuelling operations, the Containment Ventilation System (CBV) ensures that air is drawn into containment through openings in the CS, to ensure containment air is not vented through them.

The CS Hatches shall be able to close and seal in a time interval not greater than the time interval between loss of normal cooling and steam release to containment for the most limiting plant conditions that may occur during cold shutdown or plant refuelling operations. The CS Hatches shall be leak tight for design basis events and containment hydrotests in order to satisfy leak rate test requirements. Should a fuel handling accident occur, the CS Hatches must be closed to prevent excessive radiation release outside containment (in accordance with U.S. NRC Regulatory Guide 1.183). As described in Design Specification for Containment Structure Hatches [93], administrative controls should require that a dedicated individual be present, with the necessary equipment available, to close the CS Hatches. As discussed in Sub-chapter 24.4.3.2, the provision of interlocks for the hatches will be considered as further safety assessment is carried out.

The Containment Isolation System (CISO) is an Engineered Safety Feature (ESF). The safety function of the CISO is to isolate containment to prevent or limit the release of fission products in the event of a DBA. The CISO is not a discrete system; it is a grouping of valves and piping penetrating containment from other process systems that perform the containment isolation function e.g. the SFC and CBV. Power operated CIVs, which meet the requirements in 10 CFR 50, Appendix A GDC 56 for primary containment isolation, that perform the containment isolation function are automatically closed via a signal from the Plant Safety System (PSS). Details for the CISO are provided in SDD for Containment Isolation System [94], and for example in the SDDs for the SFC [37] and CBV [57].

The CBV performs two safety functions[11]: (1) a containment isolation function to maintain the containment pressure boundary; and (2) supplies makeup air to relieve vacuum pressures that may arise following a reactor trip with a loss of offsite power. Information on the CBV is provided in [57] and [59]. The CBV has CIVs which close for containment isolation to maintain the containment boundary integrity upon receipt of an isolation signal from the PSS during accident conditions. They are configured to fail closed on the loss of power or air.

Radiation monitoring is provided within the CBV's purge system exhaust duct for iodine and nobles gases and includes grab sample capability to identify any potentially high radiation

[11] The CBV also performs a number of non-safety functions: it maintains temperature in containment during normal operations below maximum levels, provides air changes to reduce airborne radioactivity levels, maintains the relative pressure in containment relative to its surroundings within approved limits, and controls and monitors gaseous radioactive effluent releases prior to discharge.

levels within containment air that could indicate a source of leakage within containment[12]. In addition, radiation monitors within containment will trip the purge system should radiation levels be sufficiently high to require containment isolation[13]. The purge system will also automatically shut down when high radiation is measured on the plant vent stack discharge to prevent an uncontrolled release. The CBV Containment Cooling System condenses water vapour from containment atmosphere and directs it to the RDS for use in leak detection.

Primary Sampling System (PSL) radiation monitors are also used to identify potential fuel cladding breach. The Primary Sampling System Liquid Sample Radiation Monitor continuously measures and indicates the concentration of radioactive materials in the samples from the RCS. The Primary Sampling System Gaseous Sample Radiation Monitor continuously measures the concentration of radioactive materials in the gaseous samples taken from containment atmosphere, see PER Chapter 5 [26] for details.

Confinement during Dry Storage

The MPC confines radioactive materials for all design basis normal, off-normal and postulated accident conditions. Summary of the FSAR on the HI-STORM Underground Maximum Capacity (UMAX) System [32] is provided below.

The MPC enclosure vessel does not contain gaskets or seals, all confinement boundary closure locations are welded. The MPC confinement boundary is formed by the sealed, cylindrical enclosure of the MPC shell welded to a solid baseplate, a lid welded around the top circumference of the shell wall, the port cover plates welded to the lid, and the closure ring welded to the lid and MPC shell providing redundant sealing of the MPC. The MPC closure ring welds provides a redundant closure to the MPC internal cavity.

Austenitic stainless steel is used in construction due to its extensive record of industrial applications that require high integrity, high ductility and high fracture strength welds. The MPC enclosure vessel welds provide a secure barrier against leakage. None of the postulated environmental phenomenon or accident conditions identified in the UMAX FSAR will cause failure of the confinement boundary. Therefore, no monitoring system for the confinement boundary is required.

The MPC confinement boundary ensures that the inert gas (helium) atmosphere inside the MPC is maintained during normal, off-normal and accident conditions of storage and transfer. It maintains the confinement helium atmosphere below design temperatures and pressures. The design of the MPC meets the guidance in ISG-18 so that leakage of radiological matter from the confinement boundary is considered to be non-credible. The confinement function of the MPC is verified through pressure testing, helium leak testing and a rigorous weld

[12] The Containment Air Filtration Exhaust Radiation Monitor continuously measures the concentration of radioactive materials in the containment purge exhaust air of the CBV system (see PER Chapter 5 [26]).

[13] The Containment Atmosphere Radiation Monitor continuously measures the radiation from the radioactive gases in the containment atmosphere and is used for safety initiations (see PER Chapter 5 [26]).

examination regime executed in accordance with the acceptance test program[14] (see HI-STORM FW FSAR [33]).

No credit is taken for the confinement provided by the fuel cladding, the HI-TRAC during MPC transfer to the ISFSI or the VVM during dry storage of the MPC; the VVM does not perform any confinement function. Confinement evaluations also account for damaged fuel and fuel debris.

The structural design of the HI-STORM UMAX system has considered: design basis loads; allowable stresses; brittle fracture; fatigue; buckling; and consideration of manufacturing and material deviations. For accident condition loadings, any permissible degradation in shielding must be shown to result in dose rates sufficiently low to permit recovery of the MPC from the damaged VVM, including unloading if necessary, and any loss of function must be readily discernible.

The HI-STORM UMAX system has been designed such that, with the exception of the earthquake induced impact load on its confinement boundary, the MPC stored in the VVM will always be subject to a less severe loading than what would be obtained in the above-ground HI-STORM spent fuel dry storage system.

The thermal evaluation of the HI-STORM UMAX system follows the guidelines of NUREG-1536 [95] and ISG-11 [66]. These guidelines provide specific limits on the permissible maximum cladding temperature of the stored spent fuel and other confinement boundary components, and on the maximum permissible pressure in the confinement space under certain operating scenarios, specifically:

- The fuel cladding temperature must meet the temperature limit appropriate to its burnup.
- The maximum internal pressure of the MPC should remain within its design pressures for normal, off-normal and accident conditions.
- The temperatures of the cask materials shall remain below their allowable limits under all scenarios.

The case for confinement integrity is made on the strength of the MPC's licensing basis in the Certificate of Compliance for Spent Fuel Storage Casks No. 1032 [77].

The confinement case described in the FSARs for the governing thermal configuration is considered to be bounding for the SMR-300 (SMR-300 fuel has a lower burn-up than that assumed for the governing thermal configuration, and the SMR-300 MPC-37 configuration

[14] Most confinement boundary welds are inspected by radiography or ultrasonic examination. Field welds are examined by the liquid penetrant method on the root (if more than one weld pass is required). and final weld passes. In addition to multi-pass liquid penetrant examination, the MPC lid-to-shell weld is pressure tested. The vent and drain port cover plates are also subject to proven non-destructive evaluations for leak detection, such as liquid penetrant examination. These inspection and testing techniques are performed to verify the integrity of the confinement boundary (see UMAX FSAR, Section 2.3.3.4 of [32]).

utilising only [REDACTED] MPC basket locations for SFAs provides a further margin of safety).

24.4.3.6 Internal Hazards

The methodologies for assessing internal hazards have been specified however specific assessment of the internal hazards affecting SSCs has not been undertaken. Generically, internal hazards that can affect the Fuel Transport and Storage SSCs, such as fire, explosion, missiles, etc. will be assessed on a room-by-room basis to identify the sources, magnitudes and consequences of each hazard. Safety measures will then be identified depending upon the IEF and unmitigated consequence.

24.4.3.7 External Hazards

External hazards associated with the generic SMR-300 are considered within Part B Chapter 21 [20]. A UK Generic Site Envelope has been established to allow for DBEs to be derived and allow for the preliminary screening of some external hazards. However, full assessment of external hazards will be carried out on a site-specific basis.

24.4.3.8 Radiological Protection

A key aspect of fuel transport and storage design is the need to demonstrate compliance with relevant radiological protection legislative requirements ensure that normal operational dose uptake is ALARP. This will require the design of SSCs including fuel transfer flasks to incorporate appropriate levels of gamma and neutron shielding to reduce dose rates in operational areas. The dose rates combined with the occupancy times of the operators will enable a Normal Operational Dose Assessment to be undertaken. The doses received by the operators will need to achieve the normal operation dose targets defined in the ONR SAPs [96].

The approach to radiological protection for the generic SMR-300 is presented in Part B Chapter 10 [13].

24.4.3.9 Criticality Safety

The fuel transport and storage design includes operations where arrays of NFAs or SFAs are brought together in fuel storage racks or MPCs. There is a need to demonstrate that these arrays can be shown to be criticality safe during normal operations, AOO and accident conditions. The NFSR, SFSR, and MPC use geometrically safe configurations to prevent criticality. Credit is taken for the fixed neutron absorber material of the NFSR, the SFSR and the MPC. The NFSR and SFSR are Region 1 (flux trap design) racks; the empty cells in the MPC-37 perform a similar function to these flux traps. For new fuel wet loading in the MPC credit is also taken for the presence of gadolinia in the fuel. The fuel handling equipment limits the number of FAs that can be handled at any one time.

The design basis for the NFSR and SFSR is a K_{eff} which does not exceed 0.95, at a 95 percent probability, 95 percent confidence level, when loaded with fuel of the maximum FA reactivity and assumed to be flooded with unborated water, as defined in SMR-160 Fuel Handling and Storage [31]. The use of Region 1 racks for the SFSR enables any FA to be placed in any cell in terms of criticality safety; no credit need be taken for fuel burnup. Boron present in the SFP water is not claimed for criticality control for FAs stored in the SFSR, and can be regarded as a defence-in-depth measure consistent with RGP for PWRs.

More information on the engineered design features of fuel storage racks is provided in the Spent Fuel Management Strategy [4].

Information on criticality control for an MPC-37 loaded with 37 SFAs is set out in the UMAX and MPC FSARs [32] [33]. The planned update to the UMAX FSAR will set out the criticality case for SMR-300 fuel in the assumed MPC-37 configuration utilising [REDACTED] basket cells for FAs.

The high-level approach to criticality assessment for the generic SMR-300 is presented in Part B Chapter 10 [13]. The criticality safety case for the generic SMR-300 to satisfy UK regulatory requirements will be prepared during the next stage of development.

24.4.3.10 Design Substantiation

All aspects of the fuel transport and storage design will be substantiated to a level consistent with the requirements of the safety function category and classification process defined in Part B Chapter 14 [15]. The US SMR class of each SSC is presented in Table 7.

At this stage, the substantiation of SSCs against relevant safety functional requirements is at a high level and commensurate with a concept design. As the design matures, design substantiation will be provided in the PSR Chapters defined in Table 9.

Table 9: PSR Chapters Providing Substantiation of SSCs

SSC	PSR Chapter
SFP	Part B Chapter 20 [19]
SFC	Part B Chapter 5 [11]
RHR	Part B Chapter 5 [11]
FHBC	Part B Chapter 5 [11]
Polar Crane	Part B Chapter 5 [11]
LLHM	Part B Chapter 5 [11]
Fuel Handling Bridge	Part B Chapter 5 [11]
NFV	Part B Chapter 20 [19]
MPC	Part B Chapter 24
HI-TRAC Transfer Cask	Part B Chapter 24
LPT	Part B Chapter 24
VCT	Part B Chapter 24
VVM of UMAX	Part B Chapter 24
UMAX ISFSI	Part B Chapter 24

24.4.3.11 CAE Summary

This sub-chapter aims to substantiate Claim 2.2.17.3, demonstrating that key SSCs are capable of supporting safe operation through appropriate hazard identification, confinement, classification and radiological controls.

A HAZOP study, that includes an assessment of RGP and OPEX from PWRs, has been performed which has identified key hazards and faults associated with the fuel route. Individual fault conditions have been captured, traced into a Preliminary Fault Schedule and will be subject to full DBAA post-GDA.

A GDA Commitment has been raised in Part B Chapter 14 (**C_Faul_103**) [15] to progress and complete UK DBAA for the SMR-300 which will include all fuel route related faults. Confidence that the safety assessment will not result in fundamental design changes to the SMR-300 fuel route is supported by the application of defence-in-depth principles, passive safety features, ALARA-based dose constraints embedded in the design and by existing design substantiation for key systems such as the MPC and UMAX.

This safety assessment process will provide a 'golden thread' from hazard identification through to design substantiation. As the design matures, further HAZOP studies shall be undertaken that will ensure all safety measures applicable to the fuel transport and storage operations are identified and substantiated.

The design is underpinned by CAE and assessments in other PSR Chapters that address internal hazards, external hazards, radiological protection, criticality, human factors and fire safety.

Although formal DBAA and UK-aligned classification are still to be completed, the planned work, including GDA Commitment **C_Faul_103** [15], and existing design features provide confidence that Claim 2.2.17.3 is met to a level consistent with Step 2 of GDA.

24.4.4 Quality, Manufacturing, Installation and EIMT

Claim 2.2.17.4: Fuel Transport and Storage SSCs achieve the design intent through quality manufacturing and installation process.

Argument 2.2.17.4-A1: Appropriate quality assurance, construction/fabrication and EIMT will be applied to the Fuel Transport and Storage SSCs to ensure that they are able to meet their safety functional requirements.

Evidence:

- FSAR for the HI-STORM Underground Maximum Capacity (UMAX) System [32].
- FSAR for the HI-STORM FW MPC Dry Storage System [33].
- SMR-160 Fuel Handling and Storage [31].
- Climate Change Design Basis (to be developed during the site-specific stage).

24.4.4.1 Quality Assurance

The quality assurance requirements for Fuel Transport and Storage SSCs can be found in the Design Specifications for the respective structures, plant and equipment. These include requirements for:

- Materials and coatings.
- Construction and fabrication.
- Pre-operational structural proof tests.
- Containment leak tests.

The requirements are generally taken from the applicable ASME BPVC.

24.4.4.2 Construction/ Fabrication

This section will be updated during the next stage of development to set out the approach to construction and fabrication of Fuel Transport and Storage SSCs.

Further information on the SMR-300 construction approach is provided in Part B Chapter 25 [22].

24.4.4.3 Examination, Inspection, Maintenance and Testing (EIMT)

The operational design life of the generic SMR-300 design is 80 years [39] and the SMR-300 Top Level Plant Design Requirements [41] document states that ‘All major plant structures shall meet the specified design life of 80 years. To achieve the design life, defined maintenance plans reflecting best available technique can be used’. However, Fuel Transport and Storage SSCs will need to perform their respective safety functions beyond the 80-year operational design life to account for the construction and decommissioning phases.

The UMAX ISFSI has a design life of 60 years and a service life of 100 years. The service life is the duration for which the component is reasonably expected to perform its intended function, if operated and maintained in accordance with the provisions of the UMAX FSAR [32]. Service life may be significantly extended beyond the design life because of the conservatism inherent in the codes, standards, and procedures used to design, fabricate, operate, and maintain the component. The UMAX ISFSI will also need to meet UK ageing management requirements, including those set out in the Inter Industry Guidance (IIG) on storage [82]. SSCs associated with the generic SMR-300 UMAX system and ISFSI will need to perform their respective safety functions for an assumed period of 100 years, during the interim on-site storage of spent fuel.

Part A Chapter 2 [5] describes the use of UK Climate Projections (UKCP) to determine the generic site envelope for GDA and account for reasonably foreseeable climate change throughout the design life of the SMR-300. Taking into account the defueling and decommissioning phases of the plant, spent fuel will be stored on site beyond the 80 year design life of the SMR-300. Spent Fuel Transport and Storage (e.g. SFP and SFC) and dry fuel storage infrastructure (e.g. ISFSI) will need to ensure the passive safety of spent fuel under future environmental conditions that could, for example be impacted by higher extreme ambient air temperature. Part B Chapter 21 [20] evaluates applicable climatic-related external hazards. Detailed assessment of risks and identification of mitigations will be required during the site-specific stage. Future evidence will include a Climate Change Design Basis to assess climate-related hazards and risks applicable to the SMR-300 site for climate scenarios and models, to identify exposures and vulnerabilities that would inform the development of resilience measures for SSC in the design.

An effective EIMT programme for managing ageing and degradation of Spent Fuel Transport and Storage SSC shall be in place throughout the lifetime of the facility. The Further information on the EIMT programme for demonstration of through-life reliability is provided in Part B Chapter 9 [12].

Pre-operational structural proof tests and containment leak tests will be undertaken during SMR-300 construction.

EMIT associated with inspections of MPCs/ FAs is set out in Sub-chapter 24.5.2.6.

24.4.4.4 CAE Summary

Quality assurance requirements for Fuel Transport and Storage SSCs are defined in the Design Specifications for the respective structures, plant and equipment. The quality assurance requirements are considered suitable as they are based on internationally recognised standards. The construction and fabrication techniques utilised will ensure that any defects during build are minimised to reduce the need for corrective action. This will be ensured through build quality assurance documentation.

An EIMT programme will be implemented that will demonstrate how SSCs are expected to perform safely beyond their 80-year operational design life, particularly for long-term systems such as the UMAX ISFS.

Together, these measure support Claim 2.2.17.4 by showing that the Fuel Transport and Storage SSCs will be manufactured, installed, and maintained to a standard that ensures their long-term functionality and compliance with safety requirement.

24.5 SPENT FUEL MANAGEMENT LIFECYCLE

Claim 2.3.4: Spent fuel will be safely managed throughout the entire reactor lifecycle.

Claim 2.3.4 concerns the lifecycle management of spent fuel, any damaged fuel and NFW for the generic SMR-300 and has been decomposed into two level four claims.

Each of these claims addresses specific aspects of the lifecycle management of spent fuel and NFW for the SMR-300.

24.5.1 Spent Fuel Management Strategy

Claim 2.3.4.1: An appropriate spent fuel management strategy consistent with the lifecycle phase is maintained.

The production and maintenance of a Spent Fuel Management Strategy for the generic SMR-300 demonstrates that the spent fuel, any damaged fuel and NFW arising from normal operations can be managed safely, securely and in an environmentally protective manner throughout its lifecycle, and in a manner that is compatible with UK policy/ strategic frameworks, and with modern standards and practices.

Claim 2.3.4.1 addresses the production and maintenance of the Spent Fuel Management Strategy [4]. There is confidence that the arguments and evidence presented to substantiate the claim do so to a level commensurate with the design maturity of the generic SMR-300.

Argument 2.3.4.1-A1: A compliant spent fuel management strategy for the generic SMR-300 aligned with UK policy and strategic frameworks has been established.

Evidence:

- Holtec SMR-300 GDA Spent Fuel Management Strategy [4].
- Integrated Waste Strategy [97].
- Site-specific Spent Fuel Management Strategy (to be produced during the site-specific stage).

During Step 2 of the GDA, the RP prepared a Spent Fuel Management Strategy for the Generic SMR-300 [4] to support Part B Chapter 24 Revision 1.

The report establishes a fully UK compliant baseline strategy for the management of spent fuel, any damaged fuel and NFW across the entire reactor lifecycle, compatible with the SMR-300 design and design philosophy, and aligned with:

- UK Policy [80], which sets out the UK Government's and devolved administrations' policies in relation to the management of spent fuel and radioactive waste.
- Funded Decommissioning Programme Guidance [81], including Base Case assumptions for management of spent fuel and HAW for new nuclear power stations.
- Nuclear Decommissioning Authority (NDA) Strategy [98] applicable to spent fuel and HAW management.
- NWS Corporate Strategy 2023 [99].
- NWS Waste Packaging Specifications [100] for spent fuel and HAW.

The strategy helps to demonstrate that spent fuel management plans, including arrangements for packaging, on-site interim storage and disposal, are capable in due course of meeting all ONR and EA regulatory requirements, and of being endorsed by NWS through the issue of a Final stage Letter of Compliance (FLoC) following Disposability Assessment during future development phases[15].

Strategy development was undertaken with the aim to ensure consistency with Holtec's overarching aim to develop a standardised SMR-300, fully optimised and suitable for fleet scale deployment across global markets including the UK. This ensures an efficient approach to design development, the relevance of learning and OPEX for the benefit of the entire global fleet and will enable future design changes to be implemented on a consistent basis[16].

The strategy also sets out: US and UK good practices and OPEX relevant to spent fuel, damaged fuel and NFW management; strategic interfaces, constraints and dependencies; and key differences with UK regulatory expectations for the SMR-300 spent fuel management strategy in the US, along with the reasons for such deviations.

The strategy addresses the following key aspects:

- The capacity for and duration of storage in the SFP, prior to transfer for processing for interim storage.
- The high-level strategy for interim storage.
- The containers to be used for interim storage.

The strategy report documents a strategic assessment undertaken, commensurate with generic SMR-300 design maturity. A long list of potential options for the various strategic aspects considered is identified, a set of suitable criteria for application to the down-selection of options developed and a range of credible options derived through application of these criteria.

Recognising the level of plant design maturity at the PSR stage, the strategy establishes a compliant baseline that is aligned with UK policy and strategic frameworks, along with alternative credible options retained as risk mitigations for key aspects for which a fully underpinned SSEC is to be made in future phases. This approach recognises that supporting information to support selection of preferred options may not be available until the PCSR stage. It also reflects design uncertainties and the potential need to accommodate future changes to the generic SMR-300 design e.g. from plant design development or GDA review.

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[15] The Committee on Radioactive Waste Management (CoRWM) has emphasised the need for early engagement with NWS as part of the GDA process. This is necessary to provide assurance that all fuel and waste streams arising from SMRs are disposable, recognising the importance of considering optimisation early in the design process to minimise the volume and activity of such streams.

[16] Relevant fuel transport and storage related examples of Holtec's standardised design for SMR-300 global deployment include: the HI-STORM UMAX system and supporting components and equipment for spent fuel dry storage campaigns (including the HI-TRAC, LPT, VCT and MPC); the DFC/ MPC for the dry storage of damaged fuel; and the NFWC for the dry storage of NFW.

To provide the most beneficial offering to a prospective developer or licensee, the strategic 'design envelope' that has been developed spans:

- Innovative approaches from SMR-300 plans, in many cases underpinned by US/ international good practice and OPEX, that provide prospects for significant safety, security, environmental and sustainability benefits, and cost/ efficiency savings.
- More established approaches from a UK perspective, for example, similar to those that have been successfully deployed at Sizewell B or proposed during other RPs' GDAs, as alternative credible options (effectively risk mitigation options for the baseline).

It is recognised that the selection and confirmation of preferred options is a matter for a prospective developer or licensee at the site-specific plant licensing/ permitting phase. This provides flexibility and ensures that credible options for spent fuel management are not prematurely foreclosed, especially whilst ALARP and BAT justifications in the generic SMR-300 design are not yet fully substantiated within the SSEC.

This approach to strategy development provides a high level of confidence that a robust, fully compliant site-specific Spent Fuel Management Strategy can be developed during the next stage of development for the generic SMR-300.

Holtec's generic SMR-300 optimised, integrated spent fuel management strategy establishes a baseline strategy entailing:

- Cooling of SFAs for a minimum of [REDACTED] in the SFSRs of the SFP.
- Use of the same optimised MPC-37 used for new fuel transfer into the SFP, for on-site spent fuel transfer during a dry fuel storage campaign.
- Transfer of the SFAs to the MPC for processing for dry storage (de-watering, drying and sealing/ welding) in the FHA of the RAB.
- Cross-site transfer of the HI-TRAC/ MPC using the VCT, for down-loading of the MPC for storage within UMAX dry fuel storage system at an on-site ISFSI.

NFW arising from operations is packaged in a NFWC once sufficient waste has been accumulated within the SFSR and is then transferred and processed for dry storage in the UMAX system at the on-site ISFSI as for spent fuel.

Any damaged fuel or fuel debris arising from the SMR-300 is packaged in a DFC located in specially designated cells in the MPC. A similar strategy as for spent fuel is followed.

The assumptions set out at the beginning of this Chapter are made in the Spent Fuel Management Strategy to establish a fully compliant baseline across the entire lifecycle, aligned with UK policy and strategic frameworks (see Sub-chapter 24.1.2).

Despite the compact SFP design of the generic SMR-300, analysis in the Spent Fuel Management Strategy [4] demonstrates that adequate capacity is available for the storage of SFAs within the SFSR until the refuelling outage following the [REDACTED]. The analysis is made on a conservative basis, for example taking no credit for the potential utilisation of SFSR cells designated for NFW and damaged fuel[17].

Potential means have also been identified to extend storage capacity until [REDACTED] years after start of generation, through the following options:

- Loading of NFAs direct from the submerged MPC into the reactor core, rather than utilising the designated SFSR cells for NFAs (this would enable alternative utilisation of [REDACTED] cells, extending storage capacity by a further [REDACTED] cycle, and would also reduce the number of FA handling operations required).
- Incorporation of additional rows in the SFSR to increase storage capacity (which would be subject to design development confirmation post-GDA).

In practice, it is planned to commence dry fuel storage campaigns during the refuelling outage following the [REDACTED], with further dry fuel storage campaigns assumed to occur during every subsequent refuelling outage. This ensures that significant SFSR capacity margin is maintained throughout the SMR-300 operating life, to enable operations to continue for a minimum of [REDACTED] even if availability for dry fuel storage operations is interrupted.

Should it be necessary, there are a number of potential options for increasing margin to SFP capacity, including loading of the MPC-37 with 37 SFAs during a dry fuel storage campaign, instead of [REDACTED] SFAs (subject to thermal analyses meeting acceptable fuel limits), and loading more than one MPC during a dry fuel storage campaign (noting that this would require the duration of a refuelling outage to be extended).

There is consequently a high level of confidence that sufficient storage capacity is available in the SFP design for the storage of spent fuel, damaged fuel and NFW[18].

With circa [REDACTED] SFAs estimated to be generated over the operating life of a single unit ([REDACTED] SFAs for a dual unit), resulting in circa [REDACTED] MPCs ([REDACTED] MPCs for a dual unit), alongside an estimated 1 to 2 NFWCs (2 to 4 NFWCs for a dual unit), it can be seen that a UMAX system with [REDACTED] VVMs ([REDACTED] VVMs for a dual unit) has adequate interim storage capacity with margin for lifetime arisings of spent fuel and NFW from both a single and dual SMR-300 unit.

The strategy also includes an assumption for the required duration of the interim storage phase for spent fuel and NFW, prior to consignment to the GDF. It also sets out how the risks

[17] It should be noted that over the 80 year operating life of a unit, the number of damaged FAs generated is expected to be in the single digits, based upon international fuel performance OPEX reported by the IAEA [125]. Only 1 to 2 dry storage campaigns are estimated to be required for NFW (no dry storage campaigns for NFW are assumed for at least the first decade of operations).

[18] It should also be noted that the proposed partial loading of the MPC-37 with [REDACTED] SFAs within the UMAX system configuration and the capacity to cool the fuel for [REDACTED] years are each expected to lead to a fuel PCT which is less than a reasonably postulated limit (e.g. 400°C as set out in ISG-11 [66], or [REDACTED] as for the Sizewell B HI-STORM system), and together could conceivably result in a calculated fuel PCT which is less than 300°C. Such a fuel temperature is judged to be sufficiently low to mitigate uncertainty associated with fuel criteria, in particular hydride re-orientation [126]. See Part B Chapter 2 [9] for more information on fuel criteria.

of package and waste form degradation over the interim storage period and the need for any repackaging operations and 'double handling' are minimised, consistent with ALARP and BAT.

Strategies for EIMT for operation of the UMAX system are described, including for monitoring the condition of the spent fuel and NFW during the interim storage period, both directly or indirectly (noting that the UMAX system is specifically designed to minimise the requirements for monitoring and human intervention). Credible approaches to retrieve spent fuel and NFW from dry storage are also described, as mitigations should it ever be necessary to do so at any time prior to its planned retrieval at the end of the interim storage phase.

Key strategic benefits of the RP's baseline strategy for the generic SMR-300, many of which contribute toward ALARP and BAT, include:

- The sub-surface SFP design located inside the CS provides enhanced protection against external hazards e.g. missiles.
- The simplified fuel route design – without fuel elevator, fuel handling building and fuel canal – improves efficiency, avoids fuel handling and transfer steps with associated risks, reduces the raw material requirements and the overall footprint/ size of the plant. This avoids and minimises contaminated inventory management across wet fuel route SSCs and simplifies materials selection for construction and equipment. Regular dry storage campaigns for fuel that has cooled sufficiently, scheduled at the end of each refuelling outage, minimises the accumulation of nuclear material in the SFP.
- This paradigm changing approach reduces the fuel related source term for any postulated accident scenarios and enables SFA transfer to passively safe dry storage conditions as soon as reasonably practicable.
- Regular dry storage campaigns also contribute to enhancing worker competency for MPC handling, transfer, and processing. This enables a culture of continuous improvement based on site based OPEX, contributing to worker dose reduction.
- The shorter duration of wet storage of spent fuel, any damaged fuel and NFW in the SFP results in lower waste generation - spanning liquid, aerial and solid wastes (i.e. SFC filters and resins) - and resulting discharges, contributing to both an ALARP and BAT and ALARA approach to minimise worker, public and environmental doses. The partially sub-surface containment designed with deep refuelling/ SFP cavities provides additional separation/ shielding via the intervening SFP water to minimise worker doses during refuelling outages.
- The optimised, integrated fuel management approach improves efficiency/ minimises refuelling outage duration, contributing to minimising worker doses.
- The use of the same optimised MPC (with capacity for re-fuel batch size) for new fuel transfer into containment and for spent fuel transfer for dry storage, reduces the number of lifting, transfer and handling operations, contributing to minimising associated risks.
- The UMAX system provides an integrated single solution for the passively safe dry storage of spent fuel, damaged fuel and NFW, resulting in the highest standards of safety, security, and environmental protection for high hazard streams.
- The sub-surface UMAX system provides the highest standards of protection achievable, resulting in public and worker exposures significantly lower than for an above ground dry storage system, and provides enhanced protection against external hazards e.g. missiles.

- The use of the DFC/ MPC for dry storage of damaged fuel in the UMAX system enables early hazard reduction, by transfer to passively safe dry storage conditions as soon as reasonably practicable.
- The use of the NFWC for the dry storage of NFW in the UMAX system enables early hazard reduction, by transfer to passively safe dry storage conditions as soon as reasonably practicable upon accumulation of sufficient waste to fill an NFWC.
- Holtec's dry storage system process is vertically integrated, with ancillary handling, transfer, and transport equipment compatible with the range of MPCs and the NFWC and compatible with the generic SMR-300 design.
- The process is also fully reversible, enabling the safe and secure retrieval of fuel and NFW from the UMAX system for inspection or repackaging at any stage in the lifecycle.
- Holtec's dry storage technology avoids the foreclosure of alternative long-term management options e.g. involving off-site consolidated storage, off-site repackaging, risk-informed consignment to waste routes, and spent fuel reprocessing.

The strategy also identifies strategic risks and mitigations[19], and strategic opportunities and enablers, which will be actively managed.

The strategy will be subject to regular future updates to ensure its content reflects the developing plant design and, in due course, its operational status, to a level of detail and with underpinning data commensurate with the phase in the generic SMR-300 lifecycle.

A site-specific Spent Fuel Management Strategy will be prepared as part of the next stage of development, building on the strategy developed during GDA Step 2. This will support the development of a fully costed plan for spent fuel management to be prepared as a key element of Decommissioning Waste Management Plan (DWMP)/ FDP development during the site-specific licensing/ permitting phase for the generic SMR-300.

24.5.1.1 CAE Summary

A Spent Fuel Management Strategy [4] has been developed for the generic SMR-300, demonstrating that spent fuel, damaged fuel and NFW can be managed safely, securely and in an environmentally protective and sustainable manner throughout the reactor lifecycle. The strategy establishes a compliant baseline consistent with UK Government Policy, the FDP Guidance, NDA and NWS strategies, and modern safety and environmental standards.

An options assessment has been undertaken to identify credible interim storage, packing and handling approaches. The strategy includes a detailed capacity analysis, showing that the compact SFP design can accommodate spent fuel from multiple operating cycles, with contingency options to extend capacity if needed. As the design progresses, the strategy will

[19] These include the risk of failed fuel being identified using gas sampling during spent fuel processing using the FHD. Any such fuel could either be returned to the SFP, the failed FA inspected and offloaded to a designated cell for damaged fuel in the SFSR, and the dry fuel storage campaign re-commenced during a subsequent outage utilising a DFC; or alternatively, subject to satisfying safety case conditions, the MPC with failed fuel could be sealed and transferred to the UMAX system for interim storage (based on UK experience at Sizewell B, it is considered likely that the FHD drying process for the SMR-300 will be sufficiently effective at removing moisture and oxygen from the MPC such that corrosion of exposed fuel pellets will be negligible).

be updated and will be supported by arrangements for EIMT, retrieval and future disposability, supporting Claim 2.3.4.1 to a level appropriate for the maturity of the design.

24.5.2 Nuclear Liabilities Regulation Compliance

Claim 2.3.4.2: The NLR expectations for spent fuel management are addressed in an appropriate manner commensurate with the lifecycle phase.

The Spent Fuel Management topic area addresses NLR aspects. Arguments for Claim 2.3.4.2 have been devised to ensure that sufficient evidence is presented such that there is confidence that all NLR requirements and expectations have been adequately addressed, commensurate with the level of design maturity of the generic SMR-300.

This sub-chapter sets out how the generic SMR-300 complies with NLR requirements for the management of spent fuel, any damaged fuel and NFW. These NLR requirements apply to the Spent Fuel Management, Radioactive Waste Management and Decommissioning topic areas.

To ensure that all the requirements are suitably addressed, the arguments for this claim have been developed to explicitly align with the broad list of NLR requirements set out within the ONR's GDA Technical Guidance [101] and not addressed via other CAE in this chapter.

The arguments are presented below and the corresponding evidence for each of these arguments can be found in their respective sub-chapters. As the NLR requirements are wide ranging in scope, the number of arguments created to support the claim is larger than for many of the other claims discussed in the PSR.

The information set out in the CAE for NLR compliance contribute to both ALARP and BAT. This sub-chapter also provides information supporting environment claims set out in the PER.

Argument 2.3.4.2-A1: An inventory of spent fuel and NFW arisings are maintained.

Argument 2.3.4.2-A2: The generation of spent fuel, NFW and secondary waste are minimised ALARP.

Argument 2.3.4.2-A3: Spent fuel and NFW are characterised, sorted and segregated to support their future management.

Argument 2.3.4.2-A4: Spent fuel and NFW are processed for interim storage in a passively safe state on-site as soon as reasonably practicable.

Argument 2.3.4.2-A5: The accumulation of spent fuel and NFW are minimised ALARP.

Argument 2.3.4.2-A6: Spent fuel and NFW are retrievable and are capable of being inspected.

Argument 2.3.4.2-A7: Spent fuel and NFW arisings are disposable.

Argument 2.3.4.2-A8: Liability management plans are maintained for spent fuel and NFW arisings.

24.5.2.1 Spent Fuel Inventory

Argument 2.3.4.2-A1: An inventory of spent fuel and NFW arisings are maintained.

Evidence:

- NWS Expert View Submission [55].
- Disposability Assessment Review (Gap Analysis) [83].
- NWS Expert View [28].
- SMR-300 UK GDA Spent Fuel Management Strategy [4].
- SMR-300 Spent Fuel Inventory (to be produced during the site-specific stage).
- SMR-300 NFW Inventory (to be produced during the site-specific stage).

The generic SMR-300 fuel assembly is designed with PWR OPEX. Over the lifetime of the reactor, approximately [REDACTED] SFAs [39] will be generated. The FA model is the Framatome GAIA 17x17, which is a modern, standard PWR fuel design. The radionuclide inventory is expected to be consistent with other PWR fuels. As part of its Expert View Submission to NWS [55], the RP set out relevant details for the fuel and requested that NWS draw analogies with other standard PWR fuels as part of their review process. A physical, chemical and radionuclide inventory for spent fuel will be prepared post-GDA. This will be maintained and updated based on any changes that could impact the inventory, for example changes to the fuel design, fuel enrichment or fuel cycle.

NFW comprises redundant activated components associated with the FAs, other in-core components and non-fuel hardware e.g. redundant instrumentation. Design information on NFW components for the generic SMR-300 is currently under development, which will reflect best practice for modern PWRs. Available information on NFW is set out in Holtec's Expert View submission [55]. A physical, chemical and radionuclide inventory for NFW will be prepared for the next phase of development. This will be maintained and updated based on any changes that could impact the inventory.

24.5.2.2 Minimisation of Generation

Argument 2.3.4.2-A2: The generation of spent fuel, NFW and secondary waste are minimised ALARP.

Evidence:

- SMR-300 UK GDA Spent Fuel Management Strategy [4].
- SMR-160 Fuel Handling and Storage [31].
- FSAR on the HI-STORM UMAX Canister Storage System [32].
- FSAR on the HI-STORM FW MPC Storage System [33].

The generic SMR-300 has been designed in accordance with EPRI's Advanced Light Water Reactor URD [47] for SMRs. Use of a modern, standard PWR fuel, along with a core design, fuel cycle, operating regime (including water chemistry) and FA handling, transfer and storage SSC design, which benefit from decades of good practice and OPEX across the global PWR fleet, ensures fuel failure rates and risks of damage to FAs are reduced to very low levels.

Similarly, generic SMR-300 in-core components and instrumentation are designed, and appropriate materials selected for such components, taking into account the available good practice and OPEX, to ensure that the volume and activity of NFW generated is minimised.

Minimising the generation of nuclear material and radioactive waste is considered in the following PSR and PER chapters:

- Part B Chapter 2 [9] for further in relation to fuel and core design.
- Part B Chapter 5 [11] on the design of auxiliary systems such as the FHBC involved in fuel handling and storage.
- Part B Chapter 23 [21] on the methodology, principles and approach to control of water quality.
- Part B Chapter 26 [23] on how Fuel Transport and Storage SSCs contribute to minimising decommissioning wastes.
- PER Chapter 6 [27] for a summary of the minimisation of volume and activity of nuclear material and radioactive waste for the generic SMR-300 (specifically, see Claims 4.2 and 4.3 in PER Chapter 6 [27]).

In relation to Part B Chapter 24, the design of Fuel Transport and Storage SSCs in accordance with OPEX, contributes to ensuring very low levels of fuel failure and minimising the potential for damage to FAs, thus minimising the number of FAs required across the operating life of the SMR-300, and thereby minimising the generation of SFAs.

It should be noted that FAs are kept sufficiently cooled at all times to maintain fuel cladding integrity and avoid the risk of hydride re-orientation. An appropriate chemistry within SFP water is maintained to minimise corrosion.

Handling and transfer equipment is designed to minimise the risks of dropped loads or damage to FAs, for example: use of cranes that are US classified as single failure proof, maintaining SFP water clarity levels, CCTV camera system under the FHBC and strict operational procedures for fuel handling followed (e.g. triple checking / verification for all FA transfers) to minimise the potential for damage to FAs.

The generic SMR-300 design is without fuel a elevator, fuel canal or fuel handling building. In addition, Holtec's integrated fuel management strategy utilising the MPC-37 for new and spent fuel transfers to and from the SFP minimises the number of fuel handling operations required, reducing the risks of fuel handling accidents/ damage to FAs. The simplified fuel route and SFP design for the generic SMR-300 reduces the total surface area that can be potentially contaminated across wet fuel route surfaces[20]. This contributes to minimising the generation of radioactive wastes during the decommissioning phase.

[20] As a result of refuelling cavity flooding, FA transfers and draining, the refuelling pool walls of the SFP typically become contaminated. A temporary demineralised water system is used to wash off the contamination on wall surfaces of the SFP, which will be captured in SFC filter and demineraliser resin secondary wastes. This will reduce activity levels on the refuelling pool walls.

In order to minimise resin consumption and the generation of secondary SFC filter and resin wastes without compromising SFP water chemistry, activity and clarity goals, the SFC is designed for intermittent use [37].

It should also be noted that the Holtec MPC design enables RCCAs to be co-packaged with SFAs in the MPC, minimising the volume of NFW generated[21].

The shorter duration of wet storage of spent fuel, any damaged fuel and NFW in the SFP than is typical for a PWR minimises secondary waste generation (i.e. SFC filters and resins) and liquid and gaseous discharges. This is particularly significant if any failed fuel is present.

As described in Sub-chapter 24.5.2.3 below, the design and loading processes for the MPC, DFC and NFWC (including its prefabricated container boxes for NFW) prevents and minimises the potential for internal contamination of the MPC and NFWC. These canisters can be reused or subject to metals recycling following the retrieval of their contents, supporting application of the waste hierarchy and minimising the generation of secondary wastes.

The features set out above contribute to both the demonstration of the ALARP principle and BAT concept.

24.5.2.3 Characterisation, Sorting and Segregation

Argument 2.3.4.2-A3: Spent fuel and NFW are characterised, sorted and segregated to support their future management.

Evidence:

- Holtec SMR-300 GDA Spent Fuel Management Strategy [4].
- SMR-160 Fuel Handling and Storage [31].
- FSAR on the HI-STORM UMAX Canister Storage System [32].
- FSAR on the HI-STORM FW MPC Storage System [33].

During reactor operations, sampling of the RCS for fission products and fuel sipping for fission gases is undertaken to identify any failed FAs. During a refuelling outage, FAs are inspected using CCTV from beneath the FHB for any damage, deformation or foreign bodies. During storage in the SFP, sampling of SFP water and fuel sipping can be undertaken to identify any leakage of fission products from the fuel. Similarly, any failed fuel detected through fuel sipping using mobile equipment during fuel processing in the FHA of the RAB can be returned to the SFP for management as damaged fuel.

Any damaged fuel and fuel debris is stored in designated cells within the SFSR for damaged fuel/ waste, prior to loading in a DFC for a dry fuel storage campaign once sufficiently cooled. The design of the DFC prevents the dispersal of any fuel debris and particulate within the MPC (see Sub-chapter 24.2.4.21). Structural, thermal, shielding, criticality and confinement evaluations have been performed to account for damaged fuel and fuel debris storage in

[21] RCCAs are considered a fuel waste by Holtec rather than a NFW, as they are co-managed with SFAs in the MPC. By contrast, MPC designs of other suppliers do not have the MPC internal head-space to enable such co-packaging, resulting in larger volumes of NFW that must be managed separately.

MPCs [32]. The SFSR and DFC ensure appropriate segregation of damaged fuel and fuel debris from other FAs.

NFW is stored in designated cells within the SFSR for damaged fuel/ waste. Once sufficient NFW has been accumulated to fill an MPC, it is loaded into an NFWC for a dry storage campaign (see Sub-chapter 24.2.4.20 on use of the NFWC). NFW components will be characterised and segregated based on their physical characteristics and irradiation history, to support their efficient future management. This is achieved by loading similar NFW components into prefabricated container boxes which are stored inside the NFWC. This simplifies future processing and consignment of NFW to the most appropriate waste routes, as it will be easier to undertake segmentation/ waste minimisation/ repackaging activities following the assumed 100 year interim storage phase, during which time dose rates and activity levels will reduce significantly (contributing to ensuring worker doses are ALARP). Wastes requiring disposal can then be consigned to the most appropriate waste route, in line with ALARP and BAT considerations e.g. to the GDF, a Near Surface Disposal (NSD) facility or potentially the Low Level Waste Repository (LLWR). The SFSR and NFWC enable appropriate sorting and segregation of NFW components.

The design of the MPC and NFWC, and operational processes for their loading, also ensures that these canisters do not become significantly contaminated, enabling them to be reused or subject to metals recycling as LLW once their respective contents have been retrieved.

The Holtec SMR-300 GDA Non-Fuel Waste Packaging BAT Workshop Output Report [102] underpins the selection of the NFWC as the preferred option, which includes more details on relevant considerations for sorting and segregation of NFW and why the NFWC is considered to perform better than other options.

24.5.2.4 Processing for Interim Storage

Argument 2.3.4.2-A4: Spent fuel and NFW are processed for interim storage in a passively safe state on-site as soon as reasonably practicable.

Evidence:

- Holtec SMR-300 GDA Spent Fuel Management Strategy [4]
- SMR-160 Fuel Handling and Storage [31].
- FSAR on the HI-STORM UMAX Canister Storage System [32].
- FSAR on the HI-STORM FW MPC Storage System [33].

Spent fuel processing for interim storage involves draining, drying and backfilling the MPC with inert gas (helium) using the FHD before the MPC is sealed using a closure weld. The helium backfill provides an inert, non-reactive atmosphere within the MPC cavity that precludes oxidation and hydride attack on the SFA cladding, and prevents corrosion of the MPC Enclosure Vessel. Use of a helium atmosphere within the MPC contributes to the long-term integrity of the fuel cladding, reducing the potential for release of fission gas or other radioactive products to the MPC cavity.

It is assumed that dry fuel storage campaigns will commence after [REDACTED] and will occur after every refuelling cycle, until all spent fuel has been transferred to the UMAX system at the ISFSI for onsite passively safe interim storage. This approach, a fundamental feature of Holtec's optimised fuel management strategy, involves regular dry storage campaigns for spent fuel that has cooled sufficiently, scheduled at the end of every refuelling outage (in contrast to many other PWRs, where spent fuel is stored for an extended period of time in the SFP, with less regular campaigns to transfer it to a ISFSI).

This innovative, paradigm changing approach minimises the accumulation of nuclear material in the SFP, reducing the fuel related source term for postulated accident scenarios and enables processing of fuel for passively safe dry storage as soon as reasonably practicable. Regular dry fuel storage campaigns after every refuelling outage also contribute to enhancing worker competency for MPC handling, transfer and processing. This enables a culture of continuous improvement based on site based OPEX, contributing to worker dose reduction.

NFW will be subject to a dry storage campaign upon accumulation of sufficient waste within the SFSR to fill an NFWC. NFW components are packaged during a dry fuel storage campaign within a NFWC. Similar to spent fuel, NFW processing for interim storage involves draining, drying and backfilling the NFWC with inert gas (helium) to prevent corrosion, before it is sealed using a closure weld, ensuring that it remains leak-tight for the duration of its service life.

The use of the large capacity NFWC (with similar external dimensions and handling arrangements to the MPC) minimises the number of handling, transfer and processing operations required for NFW, reducing handling, transfer and processing risks, and worker doses. Only up to circa 2 NFWC per unit are estimated to be generated over an 80 year operating life for the SMR-300, with no dry storage campaigns required for NFW within at least the first decade of operations.

Dry storage campaigns for NFW enable it to be processed for passively safe interim storage in the UMAX system at the ISFSI as soon as reasonably practicable, once sufficient waste has been accumulated in the SFSR (by comparison at many PWRs, NFW is not retrieved from the SFP until late in the plant's operating life, or following final plant shutdown).

24.5.2.5 Minimisation of Accumulation

Argument 2.3.4.2-A5: The accumulation of spent fuel and NFW are minimised ALARP.

Evidence:

- Holtec SMR-300 GDA Spent Fuel Management Strategy [4].

Spent fuel and NFW arisings will be consigned to suitable permitted routes as soon as reasonably practicable, upon their availability to receive them. This will ensure that the accumulation of spent fuel and NFW on the nuclear licensed site is minimised SFAIRP.

The baseline spent fuel management strategy assumes that both spent fuel and NFW will be consigned to the GDF for disposal, following a suitable period of interim storage to allow for radioactive decay and cooling. In line with assumptions by other operators, an interim storage duration of 100 years is currently assumed. However, it is recognised that the actual timing and schedule for consignments of spent fuel and NFW to the GDF will need to be agreed with NWS, for example taking into account the results of decay heat calculations for the spent fuel, and the prioritisation of planned consignments from the various nuclear operators across the UK.

The baseline strategies for spent fuel and NFW avoid the foreclosure of alternative management options by maintaining strategic opportunities for use of alternative routes should they become available. Examples of strategic opportunities include reprocessing for spent fuel, Near Surface Disposal (and potentially LLWR disposal) for any suitable NFW, and the potential use of consolidated storage facilities.

If any such alternative routes were to become available prior to planned consignments to the GDF, their use would enable the accumulation of spent fuel and/ or NFW on the nuclear licensed site to be reduced earlier than is currently assumed.

24.5.2.6 Retrievability and Inspection

Argument 2.3.4.2-A6: Spent fuel and NFW are retrievable and are capable of being inspected.

Evidence:

- Holtec SMR-300 GDA Spent Fuel Management Strategy [4].
- SMR-160 Fuel Handling and Storage [31].
- FSAR on the HI-STORM UMAX Canister Storage System [32].
- FSAR on the HI-STORM FW MPC Storage System [33].

Holtec's dry storage system is vertically integrated, with ancillary handling, transfer and transport equipment fully compatible with the SMR-300 design, the UMAX system, the FHD, the MPC and the NFWC. The dry storage process is also fully reversible, enabling the safe and secure retrieval of fuel and NFW from the UMAX system for inspection or repackaging at any stage in the plant lifecycle.

There is the capability to undertake external MPC or NFWC inspections at the ISFSI. A loaded MPC or NFWC can be inspected within the VVM of the UMAX system using remote devices, such as a borescope. An MPC/ NFWC can also be raised out of a VVM using a HI-TRAC and VCT, for visual examination, volumetric examination using ultra-sound, coupon testing and/ or eddy current testing. These techniques can be used for inspections, for example to identify any early evidence of container degradation or stress corrosion cracking (SCC).

An MPC or NFWC can be retrieved from a VVM within the UMAX system using the HI-TRAC and VCT at any time, although it is only assumed to be retrieved at the end of the interim storage phase for cross-site transfer to a shielded repackaging plant. An assumption is set out in the Spent Fuel Management Strategy [4] that spent fuel and NFW will be repackaged into NWS approved fuel canisters/ waste containers for GDF disposal.

If retrieval were necessary for any reason during the SMR-300 operating life, the MPC or NFWC can be transferred back to the FHA of the RAB during a refuelling outage[22]. The

[22] In the highly unlikely event of a need to retrieve an MPC or NFWC between refuelling outages, it could be stored temporarily within a HI-STAR 190 transport container onsite, prior to transfer back to the SFSR/ SFP. After the SFP has been removed during the decommissioning phase, an MPC or NFWC requiring retrieval prior to the end of the interim storage phase could either be transported using a suitable transport container (HI-STAR 190 transport container, or similar) to an offsite facility (wet

container weld can be removed using the Automated Weld Removal system. The container can then be transferred back into containment, filled with SFP water and lowered into the SFP. This enables SFAs or NFW components to be inspected using CCTV in the SFP and returned for storage within designated cells of the SFSR. This process could be followed in the extremely unlikely event of container degradation or failure being detected (noting that MPC/ NFWC containment boundary failure is not considered to be credible [32] [33] [54]).

24.5.2.7 Disposability

Argument 2.3.4.2-A7: Spent fuel and NFW arisings are disposable.

Evidence:

- Disposability Assessment Review [83].
- NWS Expert View Submission [55].
- NWS Expert View [28].
- Spent Fuel Management Strategy [4].
- Letters of Compliance (to be produced during the site-specific stage).

At GDA Step 2 the regulators expect the RP to have sought advice from NWS on whether any of the radioactive wastes and spent fuel that are anticipated to arise from the operation and decommissioning of the plant would present a risk to future disposability, as described in RWPR63-WI11, Preparation of Expert Views to Support Step 2 of the Generic Design Assessment Process [103].

An Expert View provides a mechanism for the RP to seek formal written advice from NWS on the risks to disposability arising from a proposed waste packaging process, or for a suite of packaging options, in the early stages of developing a spent fuel and waste packaging strategy. An Expert View does not provide a full Disposability Assessment and cannot be used as the basis for future endorsement.

It is not appropriate to apply a standard Disposability Assessment during a 2-Step GDA, due to the early stage in design of the proposals. The Expert View on disposability is intended to highlight any inherent, unmitigated risks to disposability arising from a high-level review of the spent and waste streams and future plans for their management.

Following early engagement with NWS a submission was made for an Expert View, as detailed in NWS Expert View Submission [55]. It presents information on the generic SMR-300 reactor design, fuel design, the operational waste groupings (including NFW) and their key characteristics.

Holtec's Expert View submission [55] sought NWS's Expert View on a range of potential options for the packaging and storage of the spent fuel and NFW arising from the generic SMR-300. It describes the management strategies and principles applicable to spent fuel and NFW, following the questionnaire structure defined in RWPR63-WI11 [103].

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ponds, or shielded facility) for inspection/ repackaging, or if this were not feasible an on-site shielded facility would need to be developed for this purpose.

To support the preparation of the Expert Views submission a review was undertaken of existing SMR-300 information on fuel and radioactive waste to identify any gaps in knowledge and understanding, reported in Holtec's Disposability Assessment Review [83]. This includes a comparison with available information on the planned management arrangements for fuel and radioactive waste at other reactor sites and proposed/ assessed PWRs in the UK (Sizewell B, Hinkley Point C, the AP1000 and the HPR-1000).

The Disposability Assessment Review [83] surmises that the information that has been collated on the fuel design and irradiation is comprehensive at this stage in the GDA process. The FA is a standard uranium dioxide Framatome GAIA 17x17 design, with a standard enrichment and will be irradiated in a PWR. It is noted that comparisons with the data available for fuel inventories from other PWRs can be undertaken. Similarly, NFW components will be designed in line with best practice for PWRs; no atypical components have been identified at this stage.

The RP received the NWS Expert View [28] during Step 2. NWS notes in the Expert View that in general the nature of the wastes and spent fuel from the generic SMR-300 are not significantly different to those which would arise from existing and planned PWRs with which they are already familiar, giving confidence that a disposability case could be made. NWS consider that if the spent fuel and NFW are repackaged after the interim storage phase into NWS standard containers as is assumed in the Spent Fuel Management Strategy [4], then risks to disposability will be few and low in nature. However, there are some areas where further information is required to fully establish this position.

[REDACTED]

NWS identified a total of eleven key disposability risk areas of varying severity. For spent fuel and NFW these are all identified as Low Risk status. As well as the disposability risks, NWS has identified areas of uncertainty where further information will be required to support future Disposability Assessments during future development stages [28]. The identified information is broadly consistent with the further information requirements identified in Holtec's Disposability Assessment Review [83] and will be prepared and collated to support future stages of development.

The RP's response to the Expert View has been shared with NWS [104], it details the RP's proposals in managing risks identified by NWS. The document will be used as the basis of future engagement with NWS. It is noted that certain risks remain outside the control of the RP and require a strategic view from the UK Government e.g. timescales for disposal of SMR spent fuel to the GDF.

Further engagement with NWS will be necessary as part of their Letter of Compliance (LoC) disposability assessment process. Submissions for SMR-300 waste packages will first be developed for a Conceptual LoC or interim LoC, the decision will be made post-GDA.

24.5.2.8 Liability Management Plans

Argument 2.3.4.2-A8: Liability management plans are maintained for spent fuel and NFW arisings.

Evidence:

- Holtec SMR-300 GDA Spent Fuel Management Strategy [4].
- FDP (to be developed during the site-specific stage).
- DWMP (to be developed during the site-specific stage).

The RP's baseline strategies for the long-term management of spent fuel, any damaged fuel and NFW for the generic SMR-300, set out in its Spent Fuel Management Strategy [4], align with Government's Base Case assumptions set out in FDP Guidance [81] for new nuclear power stations.

During the PCSR stage, a FDP will be prepared for approval by the Secretary of State, to enable nuclear-related construction to begin and for compliance with the Energy Act thereafter. This will involve preparation of a DWMP, setting out plans for spent fuel management, radioactive waste management and decommissioning, and a Funding Arrangements Plan (FAP). Alongside the approval of an operator's FDP, the UK Government will expect to enter into a contract with the operator regarding the terms on which the UK Government will take title to and liability for the operator's spent fuel and Intermediate Level Waste (ILW). Associated contracts and agreements also typically need to be prepared to support the FDP. During the generating phase of the SMR-300, annual and quinquennial review updates will be made to the FDP, in accordance with the Energy Act [81].

24.5.2.9 CAE Summary

Sub-chapter 24.5.2 demonstrates that the generic SMR-300's approach to spent fuel and NFW management aligns with the NLR expectations through eight supporting arguments.

These arguments collectively show that inventories of spent fuel and NFW will be established and maintained (2.3.4.2-A1), that generation and accumulation will be minimised ALARP (2.3.4.2-A2, 2.3.4.2-A5) and that it will be segregated and processed for passively safe storage at the earliest practicable opportunity using Holtec's innovative, paradigm changing approach (2.3.4.2-A3, 2.3.4.2-A4). The arguments demonstrate that the dry storage systems enable the safe and secure retrieval and inspection of fuel (2.3.4.2-A6) and provides confidence of the eventual disposability through engagement with NWS and by seeking their Expert View on disposability (2.3.4.2-A7). Suitable liability management planning is addressed through the further development of the Fuel Management Strategy and a FDP at the site-specific stage of development (2.3.4.2-A8).

Key evidence for these arguments are the Spent Fuel Management Strategy [4], the Fuel Handling and Storage document for the SMR-160 [31] and the NWS Expert View [28] produced by NWS based on their review of the RP's Expert View Submission [55] document

Together these arguments demonstrate that the NLR-related lifecycle requirements for the long-term management of spent fuel, any damaged fuel and NFW have been identified appropriately for GDA Step 2. This contributes to the demonstration that the design and operation of Fuel Transport and Storage SSCs for the generic SMR-300 are ALARP and BAT.

24.6 CHAPTER SUMMARY AND CONTRIBUTION TO ALARP AND BAT

This sub-chapter provides an overall summary and conclusion of the Fuel Transport and Storage Chapter and how this Chapter contributes to the overall demonstration of ALARP for the generic SMR-300. Part A Chapter 5 [105] sets out the overall approach for demonstration of ALARP and how contributions from individual Chapters are consolidated.

This sub-chapter therefore consists of the following elements:

- Technical Summary.
- ALARP Summary.
    - Demonstration of Relevant RGP.
    - Evaluation of Risk and Demonstration Against Risk Targets (where applicable).
    - Options Considered to Reduce Risk.
- GDA Commitments.
- Conclusion.

A review against these elements is presented below under the corresponding headings.

24.6.1 Technical Summary

Part B Chapter 24 aims to demonstrate the following level 3 claims to a maturity appropriate for a PSR:

Claim 2.2.17: SSCs which support operational fuel activities are designed to ensure safety functions and measures are delivered and radiation exposure and release of radioactivity are minimised ALARP.

The SMR-300 fuel transport and storage design has been undertaken using best practice nuclear industry codes and standards by use of the ASME BPVC, AISC, ACI, ASCE/SEI design codes.

The analysis methods used are also considered best practice, with the use of Finite Element Analysis (FEA) codes. These codes have been subject to appropriate verification and validation.

SSCs are designed to ensure confinement of nuclear matter and NFW across the SMR-300 plant lifecycle.

For normal load conditions and DBEs, all structures are designed to meet the appropriate structural acceptance criteria defined in the relevant design codes. For BDB Events, it will be demonstrated that suitable margins exist and that there are no cliff-edge effects.

The quality assurance requirements for Fuel Transport and Storage SSCs are defined in the Design Specifications for the respective buildings and structures.

A robust safety assessment and design substantiation process will be undertaken to demonstrate that the engineering integrity and reliability of Fuel Transport and Storage SSCs is proportional to the defined safety classification.

Construction and fabrication techniques will ensure that any defects during build are minimised to reduce the need for corrective action. This will be ensured through build quality assurance documentation.

EIMT will demonstrate the fitness for purpose of Fuel Transport and Storage SSCs by pre-operational structural proof tests and containment leak tests and by through life inspections. This will include the non-destructive examination of structures.

Claim 2.3.4: Spent fuel will be safely managed throughout the entire reactor lifecycle.

Sub-chapter 24.5.1 summarises the Spent Fuel Management Strategy [4] developed during Step 2 of the GDA to support PSR revision 1.

A compliant baseline strategy for the long-term management of spent fuel, any damaged fuel and NFW across the entire reactor lifecycle has been established for the generic SMR-300 that is aligned with UK policy and strategic frameworks, along with identified alternative credible options as risk mitigations for aspects subject to further SSEC substantiation, establishing a ‘design envelope’ for Spent Fuel Management Strategy (see Table 10 below).

Table 10: Baseline Strategy and Retained Credible Options

Strategic Aspect	Baseline Strategy	Retained Alternative Credible Options
Timing of transfer of spent fuel to interim storage	Spent fuel transfer from SFP as soon as reasonably practicable, subject to operational considerations (short-term SFP storage, SMR-300 option).	No alternative strategic options (SFP capacity for SFA accumulation is limited by design). The dimensions of the SFP are a constraint on the maximum number of cells that could be incorporated within the SFSR.
Interim Storage Strategy	Dry storage using an MPC/ dry fuel storage system, utilising a below ground vertical storage configuration (UMAX System, SMR-300 Option).	1) Dry storage using an MPC/ dry fuel storage system, utilising an above ground vertical storage configuration, potentially within an overbuilding (HI-STORM System).
Spent Fuel interim storage container	Optimised capacity, single wall MPC (MPC-37, SMR-300 option).	1) Optimised capacity, dual wall MPC[23] (dual wall MPC).
Damaged Fuel Interim Storage Strategy	Dry storage using an optimised capacity single wall MPC with DFC (single wall MPC-37 with DFC, SMR-300 option).	1) Dry storage using an optimised capacity dual wall MPC with DFC (dual wall MPC with DFC). 2) Long-term storage in the SFP (long-term storage in SFP).
Timing of transfer of NFW to interim storage	NFW transfer once sufficient waste has been accumulated to fill an interim storage container (shorter-term accumulation of NFW in SFP, SMR-300 option).	1) NFW transfer after an extended period of storage (longer-term accumulation of NFW in SFP).
NFW Interim Storage Strategy	Dry storage using a NFWC in a dry fuel storage system, utilising a below ground vertical storage configuration (UMAX System, SMR-300 Option).	1) Dry storage using a NFWC in a dry fuel storage system, utilising an above ground vertical storage configuration, potentially within an overbuilding (HI-SAFE System).

Sub-chapter 24.5.1 demonstrates that there is sufficient potential storage capacity with margin available in the generic SMR-300 SFP design, and in the proposed UMAX system with 60

[23] An example is Holtec’s MPC-24 DW canister used at Sizewell B, if necessary a dual wall canister of higher capacity could be developed. Handling arrangements are identical for the range of Holtec MPCs, enabling use in conjunction with the generic SMR-300 and the UMAX system.

VVMs, for the storage of spent fuel, damaged fuel and NFW arisings from the entire operating life of a generic SMR-300 unit.

The Spent Fuel Management Strategy [4] provides compelling evidence supporting the merits of the baseline strategy, which would deliver significant safety, security, environmental and sustainability benefits, and cost/ efficiency savings, compared with the dry fuel storage strategy implemented at Sizewell B. In addition, the FSARs [32] [33] provide extensive substantiation for use of the UMAX system, the NFWC and the DFC.

There is consequently a high level of confidence that the baseline spent fuel management strategy will be confirmed at the site-specific development phase, and that acceptable safety, security and environmental cases will be made for aspects subject to further SSEC substantiation to meet UK requirements.

The Spent Fuel Management Strategy [4] will be subject to regular future updates to ensure its content reflects the developing plant design and, in due course, its operational status, commensurate with the phase in the generic SMR-300 lifecycle.

Sub-chapter 24.5.2 demonstrates that key NLR requirements and expectations for the management of spent fuel, any damaged fuel and NFW across the entire reactor lifecycle have been identified.

CAE are set out to demonstrate the NLR requirements and expectations for the Spent Fuel Management topic area are addressed to a level commensurate with the generic SMR-300 design maturity.

These CAE also help to demonstrate that the design and operation of Fuel Transport and Storage SSCs for the generic SMR-300 are ALARP and BAT.

24.6.2 ALARP Summary

24.6.2.1 Demonstration of RGP

The design of the generic SMR-300 Fuel Transport and Storage SSCs complies with good practice and US NRC requirements applicable in the US. The design adopts nuclear-specific codes and standards endorsed by the US NRC and internationally recognised bodies such as the IAEA. The principal codes and standards identified within Sub-chapter 24.4.1 and listed in Appendix C are considered good practice within the UK nuclear industry. This is based on existing practices adopted on UK nuclear licenced sites, application in earlier and successful GDAs, as well as recognition as RGP by ONR TAGs.

A limited scope of safety assessment of fuel transport and storage operations has been undertaken to a level commensurate to the maturity of the design. The safety assessment process will ensure that there is a ‘golden thread’ from hazard identification through to design substantiation. The hazard identification and safety assessment will be further developed beyond Step 2.

Aspects and features of the generic SMR-300 design that are considered to contribute towards ALARP and BAT are set out in Appendix F.

24.6.2.2 Evaluation of Risk and Demonstration Against Risk Targets

The numerical targets against which the demonstration ALARP is considered can be found in Part A Chapter 2 [5]. Fuel Transport and Storage SSCs, through the defined safety functions, will contribute to the demonstration of ALARP by comparison against the risk targets in two ways:

- By fulfilling safety functions for normal operations (e.g., shielding and containment), and thereby contributing to achieving Targets 1 to 3.
- By achieving their safety classification as a duty system or a protection system, where claimed, they will contribute to the achievement of accident risk, Targets 4 to 9.

Risks above the BSL are not acceptable. Risks below the BSL require a demonstration of ALARP proportionate to the level of risk. The BSO represents the modern safety standards and expectations against which the generic SMR-300 will be assessed.

24.6.2.2.1 Evaluation of Risk

Evaluation of risk is not directly applicable to the Fuel Transport and Storage SSCs. The safety classification of the Fuel Transport and Storage SSCs will be associated with a probability of failure on demand (PFD), which is then used to calculate the overall comparison against the risk targets as described above.

At this time, the evaluation of the normal operations and accident risks against Targets 1-9 has not been provided. This information will be presented in Part B Chapter 10 [13] for normal operations, and Part B Chapter 14 [15], Part B Chapter 15 for BDBA and SAA, and Emergency Preparedness [16].

The evaluation of the normal operations and accident risks against Targets 1 to 9 is summarised in Part A Chapter 5 [105].

24.6.2.3 Options Considered to Reduce Risk

The process for the assessment of risk reduction options is presented in the RP's Design Management [106] process.

A GDA scope change proposal paper [3] revised the scope of the fundamental assessment of the UMAX system to primarily a NLR review, see Sub-chapter 24.1.1 for further details.

As described in Sub-chapter 24.6.1, the compliant baseline Spent Fuel Management Strategy [4] will require confirmation during the site-specific development phase. A number of alternative credible options set out in Table 10 are consequently retained within the design envelope for the RP's spent fuel strategy. Part A Chapter 5 [105] considers the holistic risk-reduction process for the generic SMR-300.

24.6.3 BAT Considerations

NLR is a specific technical specialism within the ONR, with a particular focus on the long timescales over which nuclear materials management, radioactive waste management and decommissioning must be considered.

As such, NLR PSR Chapters within this GDA have claims which address primarily safety design aspects and regulatory requirements within the vires of the ONR. Due to the crosscutting aspects of NLR, this Chapter also supports environment claims set out within the PER, particularly related to the prevention/ minimisation of generation and arisings (volume and activity) of spent fuel, NFW and secondary associated with the operation of Fuel Transport and Storage SSCs.

All environment claims, applicable to both the PER and PSR, reside within the BAT topic area and will be presented in PER Chapter 6 [27]. This Chapter has been prepared in collaboration with the BAT topic area and provides relevant technical details and information which supports environment CAE. The collective process is further supported by Part B Chapter 11 [107] which provides a 'golden thread' between the Safety Case and Environment Case and relevant CAE found in the PER relevant to PSR chapters. Environmental protection claims made in the PER relevant to Part B Chapter 24 include Claim 4.2 and Claim 4.3.

24.6.4 GDA Commitments

There are no GDA Commitments identified for Part B Chapter 24.

The following will be undertaken as normal business during future phases of development:

- Provision of inventories describing the physical, chemical and radiological characteristics of spent fuel and NFW.
- Development of a site-specific Spent Fuel Management Strategy, including confirmation of strategic options for implementation e.g. for MPC design.
- Development of a Storage Strategy/ Storage Implementation Plan for spent fuel, any damaged fuel and NFW.
- Development of a Spent Fuel Management Procedure for the generic SMR-300.
- Further development of the SSEC for fuel transport and storage[24], including supporting ALARP and BAT case development.
- Thermal analyses to demonstrate compliance with appropriate fuel criteria.
- Preparation of an FDP, which will include a DWMP and FAP, along with supporting Contracts and Agreements.

As previously noted, an update to the UMAX FSAR [32] is planned for SMR-300 fuel in the planned UMAX system configuration utilising [REDACTED] MPC basket cells in the MPC-37, and other relevant SMR-160 documentation will need to be updated for the SMR-300 design.

The assumptions set out in Sub-chapter 24.1.2 have been formally captured in the Commitments, Assumptions, Requirements process [6]. Further details of this process are provided in Part A Chapter 4 [7].

[24] This will include analysis of spent fuel processing using the FHD, and analysis by interfacing topics of the design and operation of the UMAX system, MPC and ancillary handling and transfer equipment, which were excluded/ deferred from the scope of this GDA.

24.6.5 Conclusion

The conclusion of this Chapter of the PSR is that:

- The Chapter Claims identified have been met with evidence at a commensurate level of maturity for a 2-Step GDA. Further CAE will be presented in due course as the design develops.
- Safety and non-safety functions have been identified for the Fuel Transport and Storage SSCs based on US methodologies.
- The Fuel Transport and Storage SSCs have been designed to meet UK RGP and apply US codes and standards in design.
- Preliminary HAZID has been carried out on the Fuel Transport and Storage SSCs which is commensurate to the design maturity of the SSCs. Further safety assessment will be carried out beyond Step 2 of the GDA, including detailed HAZID and DBAA.
- The substantiation against the identified codes and standards is likely to result in a design that contributes to the demonstration that risks to people during normal operations and accident conditions are tolerable and ALARP and BAT.
- Engagement with NWS has resulted in their Expert View [28] providing confidence that baseline packaging assumptions for spent fuel and NFW present only a low risk to disposability, and that a case for the disposal of spent fuel and NFW to the GDF can be made in due course.
- The Spent Fuel Management Strategy [4] demonstrates alignment with UK Policy, FDP Base Case assumptions and NWS requirements, ensuring these are understood and included within the design and operation of storage facilities for spent fuel and NFW.
- Part A Chapter 5 [105] concludes that it can be demonstrated that the generic SMR-300 reduces risks to ALARP and that the Fundamental Purpose of the SSEC has been fulfilled.

24.7 REFERENCES

[1] Holtec Britain, “HI-2240332, Holtec SMR GDA PSR Part A Chapter 1 Introduction,” Revision 1, July 2025.

[2] Holtec Britain, “HI-2240334, Holtec SMR GDA PSR Part A Chapter 3 Claims, Arguments & Evidence,” Revision 1, July 2025.

[3] Holtec Britain, “HI-2241518, UK GDA [DC08] Design Challenge - Reduction in GDA Scope for the HI-STORM UMAX System,” Revision 0, December 2024.

[4] Holtec Britain, “HI-2241163, Holtec SMR-300 GDA Spent Fuel Management Strategy,” Revision 0, 2024.

[5] Holtec Britain, “HI-2240333, Holtec SMR GDA PSR Part A Chapter 2 General Design Aspects and Site Characteristics,” Revision 1, July 2025.

[6] Holtec Britain, “HPP-3295-0013, Holtec SMR-300 Generic Design Assessment Capturing and Managing Commitments, Assumptions and requirements,” Revision 1, January 2025.

[7] Holtec Britain, “HI-2240335, Holtec SMR GDA PSR Part A Chapter 4 Lifecycle Management of Safety and Quality Assurance,” Revision 1, July 2025.

[8] Holtec Britain, “HI-2240337, Holtec SMR GDA PSR Part B Chapter 1 Reactor Coolant System and Engineered Safety Features,” Revision 1, June 2025.

[9] Holtec Britain, “HI-2240776, Holtec SMR GDA PSR Part B Chapter 2 Reactor Fuel and Core,” Revision 1, June 2025.

[10] Holtec Britain, “HI-2240338, Holtec SMR GDA PSR Part B Chapter 4 Control and Instrumentation Systems,” Revision 1, June 2025.

[11] Holtec Britain, “HI-2240777, Holtec SMR GDA PSR Part B Chapter 5 Reactor Supporting Facilities,” Revision 1, June 2025.

[12] Holtec Britain, “HI-2240340, Holtec SMR GDA PSR Part B Chapter 9 Conduct of Operations,” Revision 1, June 2025.

[13] Holtec Britain, “HI-2240341, Holtec SMR GDA PSR Part B Chapter 10 Radiological Protection,” Revision 1, June 2025.

[14] Holtec Britain, “HI-2240344, Holtec SMR GDA PSR Part B Chapter 13 Radioactive Waste Management,” Revision 1, June 2025.

[15] Holtec Britain, “HI-2240345, Holtec SMR GDA PSR Part B Chapter 14 Design Basis Accident Analysis,” Revision 1, June 2025.

[16] Holtec Britain, “HI-2240346, Holtec SMR GDA PSR Part B Chapter 15 BDBA, Severe Accidents Analysis and Emergency Preparedness,” Revision 1, June 2025.

[17] Holtec Britain, “HI-2240348, Holtec SMR GDA PSR Part B Chapter 17 Human Factors,” Revision 1, June 2025.

[18] Holtec Britain, “HI-2240356, Holtec SMR GDA PSR Part B Chapter 19 Mechanical Engineering,” Revision 1, June 2025.

[19] Holtec Britain, “HI-2240357, Holtec SMR GDA PSR Part B Chapter 20 Civil Engineering,” Revision 1, June 2025.

[20] Holtec Britain, “HI-2240350, Holtec SMR GDA PSR Part B Chapter 21 External Hazards,” Revision 1, June 2025.

[21] Holtec Britain, “HI-2240352, Holtec SMR GDA PSR Part B Chapter 23 Reactor Chemistry,” Revision 1, June 2025.

[22] Holtec Britain, “HI-2240354, Holtec SMR GDA PSR Part B Chapter 25 Construction and Commissioning,” Revision 1, June 2025.

[23] Holtec Britain, “HI-2240355, Holtec SMR GDA PSR Part B Chapter 26 Decommissioning Approach,” Revision 1, June 2025.

[24] Holtec Britain, “HI-2240360, Holtec SMR GDA PER Chapter 1 Radioactive Waste Management Arrangements,” Revision 1, July 2025.

[25] Holtec Britain, “HI-2240363, Holtec SMR GDA PER Chapter 4 Conventional Impact Assessment,” Revision 1, July 2025.

[26] Holtec Britain, “HI-2240801, Holtec SMR GDA PER Chapter 5 Monitoring and Sampling,” Revision 0, June 2025.

[27] Holtec Britain, “HI-2241253, Holtec SMR GDA PER Chapter 6 Demonstration of BAT,” Revision 0, June 2025.

[28] Nuclear Waste Services, “LTR/WMIDA-582826885-12687, GDA Step 2 Expert View on the Disposability of Wastes and Spent Fuel arising from the Holtec SMR-300,” February 2025.

[29] Holtec Britain, “HI-2240261, SMR-300 Generic Security Report for the Generic Design Assessment,” Revision 0, May 2024.

[30] Holtec Britain, "HI-2240260, Preliminary Safeguards Report," Revision 0, February 2024.

[31] Holtec International, "HI-2188434, SMR-160 Fuel Handling and Storage," Revision 1, March 2020.

[32] Holtec International, "HI-2115090, Final Safety Analysis Report on the HI-STORM Underground Maximum Capacity (UMAX) System," Revision 8, June 2024.

[33] Holtec International, "HI-2114830, Final Safety Analysis Report for HI-STORM FW MPC Dry Storage System," Revision 10, 2023.

[34] Holtec International, "PS-8002-0001, SMR-300 Purchase Specification - Fuel Handling Bridge Crane," Revision 0, June 2024.

[35] Holtec International, "PS-8002-0002, SMR-300 Purchase Specification - Polar Crane," Revision 1, June 2024.

[36] Holtec International, "PS-8002-0003, SMR-300 Purchase Specification - Reactor Auxiliary Building Crane," Revision 0, June 2024.

[37] Holtec International, "HI-2240167, System Design Description for the Spent Fuel Pool Cooling System," Revision 0, April 2024.

[38] Holtec Britain, "HI-2240648, GDA Design Reference Point," Revision 2, May 2025.

[39] Holtec International, "HI-2240077, SMR-300 Plant Overview," Revision 1, April 2025.

[40] Holtec International, " DWG-15131, SMR-300 Fuel Assembly Design Interface Drawing (Exp Ctrl)," Revision 0.

[41] Holtec International, "HI-2240251, SMR-300 Top Level Plant Design Requirements," Revision 2, January 2025.

[42] Holtec International, "HI-2084122, Metamic-HT Qualification Sourcebook".

[43] Holtec International, "Neutron Absorber Material," 2024.

[44] Holtec International, "HI-2240497, System Design Description for Liquid Radwaste System," Revision 0, 2024.

[45] Holtec International, "DWG-10903, Spent Fuel Pool," Revision 3, 2022.

[46] Holtec International, "DWG-11506, Spent Fuel Pool Liner," Revision 0, 2022.

[47] Electric Power Research Institute, “Advanced Nuclear Technology: Advanced Light Water Reactor Utility Requirements Document,” Revision 13, 2014.

[48] Holtec International, “HI-2240216, Spent Fuel Pool Cooling System Heat Exchanger Thermal Report,” Revision 0, March 2024.

[49] Holtec International, “HI-2177933, Spent Fuel Pool Cooling System Flow and Line Sizing,” Revision 1, 2024.

[50] Holtec International, “HI-2188425, Specification for SMR-160 Spent Fuel Storage Racks,” Revision 3, 2020.

[51] Holtec International, “HI-2188516, Specification for SMR-160 New Fuel Storage Racks,” Revision 1, 2019.

[52] Holtec International, “ DS-331, DS331(Dry Storage) Structural Acceptance Criteria for the Metamic-HT Fuel Basket,” Revision 0, July 2017.

[53] Holtec International, “DWG-10877, Integrated Containment General Arrangement Low Profile Transporter,” Revision 0, July 2017.

[54] Holtec International, “HI-2188108, Design Report on the HI-SAFE FW Non-Fuel Waste Storage System,” Revision 3, 2018.

[55] Holtec Britain, “HI-2240759, NWS Expert View Submission,” Revision 1, September 2024.

[56] Holtec International, “HI-2210376, Spent Fuel Pool Cooling System Regulatory Compliance Report,” Revision 0, 2022.

[57] Holtec International, “HI-2145972, System Design Description for Containment Ventilation System,” Revision 1, December 2019.

[58] Holtec International, “HI-2188323, Design Specification for Reactor Auxiliary Building,” Revision 2, 2022.

[59] Holtec International, “HI-2240585, Containment Ventilation System Regulatory Compliance Report,” Revision 0, November 2024.

[60] Holtec International, “HPP-160-3004, SMR-160 Systems, Structures, and Components Classification Standard,” Revision 6, 2021.

[61] United States Nuclear Regulatory Commission, “Code of Federal Regulations, Title 10 CFR Part 50, Domestic Licensing of Production and Utilization Facilities”.

[62] United States Nuclear Regulatory Commission, “Regulatory Guide 1.13, Spent Fuel Storage Facility Design Basis,” Revision 2, 2017.

[63] American National Standards Institute/American Nuclear Society, "ANSI/ANS 57.2, Design Objectives for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Plants," 1983.

[64] United States Nuclear Regulatory Commission, "Regulatory Guide 1.26, Quality Group Classifications and Standards For Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants," Revision 6, December 2021.

[65] United States Nuclear Regulatory Commission, "Regulatory Guide 1.29, Seismic Design Classification for Nuclear Power Plants," Revision 6, July 2021.

[66] United States Nuclear Regulatory Commission, " ISG-11, Cladding Considerations for the Transportation and Storage of Spent Fuel," Revision 3, November 2003.

[67] United States Nuclear Regulatory Commission, "ISG-18, The Design and Testing of Lid Welds on Austenitic Stainless Steel Canisters as the Confinement Boundary for Spent Fuel Storage," Revision 1, October 2008.

[68] Holtec Britain, "HI-2241408, Mechanical Codes and Standards Report," Revision 0, December 2024.

[69] Holtec Britain, "HI-2240351, Holtec SMR GDA PSR Part B Chapter 22 Internal Hazards," Revision 1, June 2025.

[70] American Institite of Mechanical Engineers, "BPVC.III.1.NE, BPVC Section III, Rules for Construction of Nuclear Facility Components, Division 1, Subsection NE, Class MC Components," 2021.

[71] American Society of Mechanical Engineers, "BPVC.III.NCA, BPVC Section III, Rules for Construction of Nuclear Facility Components, Subsection NCA, General Requirements for Division 1 and Division 2," 2021.

[72] American Concrete Institute, "318-05/318R-05, Building Code Requirements for Structural Concrete and Commentary," 2005.

[73] American Concrete Institute, "349M-13, Code Requirements for Nuclear Safety-Related Concrete Structures and Commentary," 2014.

[74] American National Standards Institute/American Institute of Steel Construction, "ANSI/AISC N690-18, Specification for Safety-Related Steel Structures for Nuclear Facilities," 2018.

[75] American Society of Civil Engineers/Structural Engineering Institute, "ASCE/SEI 4-16, Seismic Analysis of Safety-Related Nuclear Structures," 2017.

[76] American Society of Mechanical Engineers, "NOG-1, Rules for Construction of Overhead and Gantry Crane (Top Running Bridge, Multiple Girder)," 2020.

[77] United States Nuclear Regulatory Commission, “Certificate of Compliance No. 1032, Certificate of Compliance for Spent Fuel Storage Casks,” June 2011.

[78] United States Nuclear Regulatory Commission, “Certificate of Compliance No. 1040, Certificate of Compliance for Spent Fuel Storage Casks,” April 2015.

[79] Holtec International, “HI-2167374, Licensing Report on the HI-STORE CIS Facility,” Revision 1, July 2023.

[80] Department for Energy Security and Net Zero, Scottish Government, Welsh Government, Department for Agriculture, Environment and Rural Affairs, “UK Policy Framework for Managing Radioactive Substances and Nuclear Decommissioning,” 2024.

[81] Department of Energy and Climate Change, “The Energy Act 2008, Funded Decommissioning Programme Guidance for New Nuclear Power Stations,” 2011.

[82] Nuclear Decommissioning Authority, “Industry Guidance, Interim Storage of Higher Activity Waste Packages – Integrated Approach,” Issue 4, December 2021.

[83] Holtec Britain, “HI-2240781, Disposability Assessment Review (Gap Analysis),” Revision 0, 2024.

[84] Holtec Britain, “HI-2250097, Mechanical Engineering Design Basis Report,” Revision 0, March 2025.

[85] Holtec Britain, “HI-2241296, Design Basis Report for Nuclear Classified Civil Engineering Structures,” Revision 0, 2024.

[86] American Concrete Institute, “229R-13: Report on Controlled Low-Strength Materials,” June 2013.

[87] American Concrete Institute, “116R-00, Cement and Concrete Terminology,” 2000.

[88] Holtec Britain, “HI-2241323, Preliminary Fault Schedule Report,” Revision 1, 2025.

[89] Holtec Britain, “HI-2241432, SMR-300 Fuel Storage and Transport Route HAZOP Report,” Revision 0, 2025.

[90] Holtec Britain, “HI-2241279, Safety Assessment Handbook,” Revision 1.

[91] Holtec International, “HI-2240235, SMR-300 Acceptance Criteria for Deterministic Safety Analysis,” Revision 2.

[92] Holtec International, “HI-2240159, Design Specification for the Containment Structure,” Revision 1, September 2024.

[93] Holtec International, "HI-2167396, Design Specification for Containment Structure Hatches," Revision 1, November 2022.

[94] Holtec International, "HI-2146278, System Design Description for Containment Isolation System," Revision 0, August 2019.

[95] United States Nuclear Regulatory Commission, "NUREG-1536, Standard Review Plan for Spent Fuel Dry Storage Systems at a General License Facility," Revision 1, July 2010.

[96] Office for Nuclear Regulation, "Safety Assessment Principles for Nuclear Facilities, 2014 Edition," Revision 1, January 2020.

[97] Holtec Britain, "HI-2241151, Integrated Waste Strategy," Revision 1, January 2025.

[98] Nuclear Decommissioning Authority, "NDA Strategy, Effective from March 2021, Cleaning up the UK's earliest nuclear sites, caring for people and the environment," March 2021.

[99] Nuclear Waste Services, "Corporate Strategy," 2023.

[100] Nuclear Waste Services, "WPS/240/02, Specification for High Heat Generating Waste precursor Product," October 2024.

[101] Office for Nuclear Regulation, "ONR-GDA-GD-007, New Nuclear Power Plants: Generic Design Assessment Technical Guidance," Revision 0, May 2019.

[102] Holtec Britain, "HI-2241494, Holtec SMR-300 GDA - Non-Fuel Waste Packaging Workshop Output Report (worked example)," Revision 0, March 2025.

[103] Nuclear Waste Services, "RWPR63-WI11, Preparation of Expert Views to Support Step 2 of the Generic Design Assessment Process," Revision 0, October 2023.

[104] Holtec Britain, "HI-2240760, Response to NWS Expert View," Revision 0, May 2025.

[105] Holtec Britain, "HI-2240336, Holtec SMR GDA PSR Part A Chapter 5 Summary of ALARP and SSEC," Revision 1, July 2025.

[106] Holtec Britain, "HPP-3295-0017, Holtec SMR-300 Design Management," Revision 1, December 2024.

[107] Holtec Britain, "HI-2240342, Holtec SMR GDA PSR Part B Chapter 11 Environmental Protection," Revision 1, June 2025.

[108] American National Standards Institute, "ANSI N14.5, American National Standard for Radioactive Materials – Leakage Tests on Packages for Shipment," 2022.

[109] American National Standards Institute/American Nuclear Society, “ANSI/ANS 57.2, Design Objectives for Light-Water Spent Fuel Storage Facilities at Nuclear Power Plants,” 1983.

[110] American National Standards Institute/American Nuclear Society, “ANSI/ANS 8.1, Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors,” 2014.

[111] American National Standards Institute/American Nuclear Society, “ANSI/ANS 8.17, Criticality Safety Criteria for the Handling, Storage, and Transportation of LWR Fuel Outside Reactors,” 2004.

[112] American Society of Civil Engineers/Structural Engineering Institute, “ASCE/SEI 7-16, Minimum Design Loads in Buildings and Other Structures,” 2017.

[113] American Society of Mechanical Engineers, “BPVC.III.1.NCD, BPVC Section III, Rules for Construction of Nuclear Facility Components, Division 1, Subsection NB, Class 1 Components,” 2021.

[114] American Society of Mechanical Engineers, “BPVC.III.NF, BPVC Section III, Rules for Construction of Nuclear Facility Components, Subsection NF, Supports,” 2021.

[115] United States Nuclear Regulatory Commission, “NUREG-0554, Single-Failure-Proof Cranes for Nuclear Power Plants,” May 1979.

[116] United States Nuclear Regulatory Commission, “NUREG-0612, Control of Heavy Loads at Nuclear Power Plants,” January 1980.

[117] United States Nuclear Regulatory Commission, “NUREG-0800, Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants,” March 2017.

[118] United States Nuclear Regulatory Commission, “Regulatory Guide 1.140, Design, Inspection, and Testing Criteria for Air Filtration and Adsorption Units of Normal Atmosphere Cleanup Systems in Light-Water-Cooled Nuclear Power Plants,” Revision 3, August 2016.

[119] United States Nuclear Regulatory Commission, “Regulatory Guide 1.183, Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors,” Revision 1, October 2023.

[120] United States Nuclear Regulatory Commission, “ISG-15, Materials Evaluation,” January 2001.

[121] Holtec Britain, “HI-2241193, Holtec SMR-300 GDA Safety Principles Compliance Review,” Revision 1.

[122] Environment Agency, “Radioactive substances regulation (RSR): objective and principles,” December 2021. [Online]. Available: https://www.gov.uk/government/publications/radioactive-substances-regulation-rsr-objective-and-principles/radioactive-substances-regulation-rsr-objective-and-principles#:~:text=Principle%204%3A%20protecting%20wildlife,-Radioactive%20substances%20activitie.

[123] Environment Agency, “Radioactive substances management: generic developed principles,” May 2024. [Online]. Available: https://www.gov.uk/government/publications/rsr-generic-developed-principles-regulatory-assessment/radioactive-substances-management-generic-developed-principles.

[124] Environment Agency, “Engineering: Generic Developed Principles,” May 2024. [Online]. Available: https://www.gov.uk/government/publications/rsr-generic-developed-principles-regulatory-assessment/engineering-generic-developed-principles#:~:text=meet%20our%20expectations.-,ENDP1%20%E2%80%93%20inherent%20environmental%20protection,operational%20purpose%.

[125] International Atomic Energy Agency, “Review of Fuel Failures in Water Cooled Reactors (2006–2015),” 2019.

[126] Holtec Britain, “HI-2241327, Holtec SMR-300 Fuel Design Criteria and Limits,” Revision 0.

[127] Holtec Britain, “HI-2241229, Design Reference Point and GDA Scope Change Proposal No 1,” Revision 0, October 2024.

24.8 LIST OF APPENDICES

Appendix A Fuel Route Process Diagram

[REDACTED]

Figure 4: Process Diagram for the Fuel Route

Appendix B Summary of the Current CAE Route Map for SPR Part B Chapter 24

Table 11: CAE Route Map for PSR Part B Chapter 24

[REDACTED]

Appendix C US Codes and Standards

The principal codes and standards used for the design of Fuel Transport and Storage SSCs for the generic SMR-300 design are summarised in Table 12.

The principal codes and standards applied to the design of nuclear safety related Fuel Transport and Storage SSCs are nuclear-specific and are considered to represent best practice within the UK nuclear industry.

Table 12: Principal Design Codes and Standards used for Fuel Transport and Storage

Label	Title	Revision/ Date
ACI 349-13	Code Requirements for Nuclear Safety-Related Concrete Structures and Commentary [73]	2014
ANSI/American Institute of Steel Construction (AISC) N690-18, Appendix N9	Specification for Safety-Related Steel Structures for Nuclear Facilities [74]	2018
ANSI N14.5	American National Standard for Radioactive Materials – Leakage Tests on Packages for Shipment [108]	1997
ANSI/ANS 57.2	Design Requirements for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Plants [109]	1983
ANSI/ANS 8.1	Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors [110]	2014
ANSI/ANS 8.17	Criticality Safety Criteria for the Handling, Storage, and Transportation of LWR Fuel Outside Reactors [111]	2004
American Society of Civil Engineers (ASCE)/Structural Engineering Institute (SEI) 4-16	Seismic Analysis of Safety-Related Nuclear Structures [75]	2017
ASCE/SEI 7-16	Minimum Design Loads in Buildings and Other Structures, American Society of Civil Engineers [112]	2017
ASME BPVC.III.1.NB	Boiler and Pressure Vessel Code (BPVC) Section III - Rules for Construction of Nuclear Facility Components – Division 1 – Subsection NB – Class 1 Components [113]	2021
ASME BPVC.III.1.NE	Boiler and Pressure Vessel Code (BPVC) Section III - Rules for Construction of Nuclear Facility Components – Division 1 – Subsection NE – Class MC Components [70]	2021
ASME BPVC.III.1.NF	Boiler and Pressure Vessel Code (BPVC) Section III - Rules for Construction of Nuclear Facility Components – Division 1 – Subsection NF – Supports [114]	2021
ASME BPVC.III.NCA	Boiler and Pressure Vessel Code (BPVC) Section III – Rules for Construction of Nuclear Facility Components – Subsection NCA – General Requirements for Division 1 and Division 2 [71]	2021
ASME NOG-1	Rules for Construction of Overhead and Gantry Crane (Top Running Bridge, Multiple Girder) [76].	2020
NUREG-1536	Standard Review Plan for Spent Fuel Dry Storage Systems at a General License Facility [95]	Revision 1, July 2010
NUREG-0554	Single-Failure-Proof Cranes for Nuclear Power Plants [115]	May 1979
NUREG-0612	Control of Heavy Loads at Nuclear Power Plants [116]	January 1980
NUREG-0800	Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants [117]	March 2017
EPRI URD	Advanced Nuclear Technology: Advanced Light Water Reactor URD [47]	Revision 13, 2014
RG 1.140	Design, Inspection, and Testing Criteria for Air Filtration and Adsorption Units of Normal Atmosphere Cleanup Systems in Light-Water-Cooled Nuclear Power Plants [118]	Revision 3, August 2016
RG 1.183	Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors [119]	Revision 1, October 2023

Label	Title	Revision/ Date
ISG-11	Cladding Considerations for the Transportation and Storage of Spent Fuel [66]	Revision 3, November 2003
ISG-15	Materials Evaluation [120]	January 2001
ISG-18	The Design/Qualification of Final Closure Welds on Austenitic Stainless Steel Canisters as Confinement Boundary for Spent Fuel Storage and Containment Boundary for Spent Fuel Transportation [67]	Revision 1, October 2008
10CFR72.24(c), (p)	Contents of Application: Technical Information	N/A
10CFR72.82(d)	Inspections and Tests	N/A
10CFR72.104(a)	Criteria for Radioactive Materials in Effluents and Direct Radiation from an ISFSI or monitored retrievable storage installation (MRS).	N/A
10CFR72.106(b)	Controlled area of an ISFSI or MRS.	N/A
10CFR72.122(a), (f), (h)(1)	Quality Standards	N/A
10CFR72.124	Criteria for Nuclear Criticality Safety	N/A
10CFR72.128(a)(1), (a)(2)	Criteria for Spent Fuel, High-level Radioactive Waste, Reactor-related Greater than Class C Waste, and Other Radioactive Waste Storage and Handling.	N/A
10CFR72.162	Test Control	N/A
10CFR72.174	Quality Assurance Records	N/A
10CFR72.212(b)(8)	Conditions of General License Issued Under § 72.210.	N/A
10CFR72.236(c), (e), (g), (h), (j), (k), (l), (m)	Specific requirements for spent fuel storage cask approval and fabrication.	N/A
10CFR50 Appendix A	General Design Criteria for Nuclear Power Plants	N/A
10CFR50 Appendix J	Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors	N/A
10CFR50.68(b)(2), (b)(3), (b)(4)	Criticality Accident Requirements	N/A
CoC No. 1032	Certificate of Compliance for Spent Fuel Storage Casks [77]	June 2011
CoC No. 1040	Certificate of Compliance for Spent Fuel Storage Casks [78]	April 2015

Appendix D UK Codes and Standards

ONR Site Licence Conditions

Licence Conditions (LCs) applicable to NLR related aspects of fuel transport and storage are:

- **LC4 Restrictions on nuclear matter on the site.** This condition seeks to ensure, in the interests of safety, that the licensee has adequate arrangements to control the introduction and storage of nuclear matter on the licenced site.
- **LC32 Accumulation of radioactive waste.** The purpose of this condition is to ensure that the licensee has adequate arrangements to ensure that the production and accumulation of radioactive waste on the site is minimised.
- **LC34 Leakage and escape of radioactive material and radioactive waste.** The purpose of this condition is to place a duty on the licensee to ensure so far as reasonably practicable that radioactive material and radioactive waste is adequately controlled or contained so as to prevent leaks or escapes, and that in the event of any fault or accident which results in a leak or escape, the radioactive material or radioactive waste can be detected, recorded and reported to the ONR.

ONR Safety Assessment Principles

ONR SAPs for Nuclear Facilities [96] applicable to NLR related aspects of fuel transport and storage are set out in Table 13. SAPs applicable to non-NLR related aspects of fuel transport and storage are set out in the relevant PSR Chapters.

Table 13: NLR Related SAPs Relevant to Fuel Transport and Storage

SAP Number	Definition
RW.1	Strategies for radioactive waste. A strategy should be produced and implemented for the management of radioactive waste on a site.
RW.2	Generation of radioactive waste. The generation of radioactive waste should be prevented or, where this is not reasonably practicable, minimised in terms of quantity and activity.
RW.3	Accumulation of radioactive waste. The total quantity of radioactive waste accumulated on site at any time should be minimised so far as is reasonably practicable.
RW.4	Characterisation and segregation. Radioactive waste should be characterised and segregated to facilitate its subsequent safe and effective management.
RW.5	Storage of radioactive waste and passive safety. Radioactive waste should be stored in accordance with good engineering practice and in a passively safe condition.
RW.6	Passive safety timescales. Radiological hazards should be reduced systematically and progressively. The waste should be processed into a passive safe state as soon as is reasonably practicable.
RW.7	Making and keeping records. Information that might be needed for the current and future safe management of radioactive waste should be recorded and preserved.
ENM.1	Strategies for managing nuclear matter. A strategy (or strategies) should be made and implemented for the management of nuclear matter.
ENM.2	Provisions for nuclear matter brought onto, or generated on, the site. Nuclear matter should not be generated on the site, or brought onto the site, unless sufficient and suitable arrangements are available for its safe management on the site.
ENM.3	Transfers and accumulation of nuclear matter. Unnecessary or unintended generation, transfer or accumulation of nuclear matter should be avoided.

SAP Number	Definition
ENM.4	Control and accountancy of nuclear matter. Nuclear matter should be appropriately controlled and accounted for at all times.
ENM.5	Characterisation and segregation. Nuclear matter should be characterised and segregated whenever practicable to facilitate its safe management.
ENM.6	Storage in a condition of passive safety. When nuclear matter is to be stored on site for a significant period of time it should be stored in a condition of passive safety whenever practicable and in accordance with good engineering practice.
ENM.7	Retrieval and inspection of stored nuclear matter. Storage of nuclear matter should be in a form and manner that allows it to be retrieved and, where appropriate, inspected.
ECV.1	Prevention of leakage. Radioactive material should be contained and the generation of radioactive waste through the spread of contamination by leakage should be prevented.
ECV.2	Minimisation of releases. Containment and associated systems should be designed to minimise radioactive releases to the environment in normal operation, fault and accident conditions.
ECV.3	Means of confinement. The primary means of confining radioactive materials should be through the provision of passive sealed containment systems and intrinsic safety features, in preference to the use of active dynamic systems and components.
ECV.4	Provision of further containment barriers. Where the radiological challenge dictates, waste storage vessels, process vessels, piping, ducting and drains (including those that may serve as routes for escape or leakage from containment) and other plant items that act as containment for radioactive material, should be provided with further containment barrier(s) that have sufficient capacity to deal safely with the leakage resulting from any design basis fault.
ECV.5	Minimisation of personnel access. The need for access by personnel to the containment should be minimised.
ECV.6	Monitoring devices. Suitable and sufficient monitoring devices with alarms should be provided to detect and assess changes in the materials and substances held within the containment.
ECV.7	Leakage monitoring. Appropriate sampling and monitoring systems should be provided outside the containment to detect, locate, quantify and monitor for leakages or escapes of radioactive material from the containment boundaries.
ECV.8	Minimisation of provisions for import or export of materials or equipment. Where provisions are required for the import or export of materials or equipment into or from containment, the number of such provisions should be minimised.
ECV.9	Containment and ventilation system design. The design should ensure that controls on fissile content, radiation levels, and overall containment and ventilation standards are suitable and sufficient.
ECE.26	Provision for decommissioning. Special consideration should be given at the design stage to the incorporation of features to facilitate radioactive waste management and the future decommissioning and dismantling of the facility.

SAPs relating to nuclear material management (ENM.1 – ENM.7) are applicable to new fuel and spent fuel. SAPs related to radioactive waste management (RW.1 – RW.7) apply to the management of NFW.

Spent fuel is not classified as radioactive waste as it could potentially be subject to reprocessing in the future, subject to enabling Government policy being put in place.

A number of SAPs detailed in Holtec SMR-300 GDA Safety Principles Compliance Review [121] are relevant to Part B Chapter 24.

EA Environmental Principles

Certain environmental principles applicable to NLR related aspects of fuel transport and storage are set out in:

- Radioactive Substances Regulation (RSR): Objectives and Principles [122].
- Radioactive Substances Management Generic Developed Principles (RSMDPs) [123].
- Radioactive Substances Management Engineering Generic Developed Principles (ENDPs) [124].

Table 14 sets out RSMDPs relevant to fuel transport and storage.

Table 14: RSMDPs Relevant to Fuel Transport and Storage

RSMDP Number	Definition
RSMDP1	Radioactive substances strategy. A strategy should be produced for the management of all radioactive substances.
RSMDP2	Justification. Radioactive wastes shall not be created unless the practice giving rise to the waste has been justified (in advance for new practices).
RSMDP3	Use of BAT to minimise waste. BAT should be used to ensure that production of radioactive waste is prevented and where that is not practicable minimised with regard to activity and quantity.
RSMDP4	Methodology for identifying BAT. BAT should be identified by a methodology that is timely, transparent, inclusive, based on good quality data, and properly documented.
RSMDP5	Actions having irreversible consequences. Actions with radioactive substances having irreversible consequences should only be undertaken after thorough, detailed, consideration of the potential consequences of those actions and of the other available options. BAT should be used to prevent irreversible consequences from occurring inadvertently.
RSMDP6	Application of BAT. In all matters relating to radioactive substances, BAT means the most effective and advanced stage in the development of activities and their methods of operation.
RSMDP7	BAT to minimise environmental risk and impact. When making decisions about the management of radioactive substances, BAT should be used to ensure that the resulting environmental risk and impact are minimised.
RSMDP8	Segregation of wastes. BAT should be used to prevent the mixing of radioactive substances with other materials, including other radioactive substances, where such mixing might compromise subsequent effective management or increase environmental impacts or risks.
RSMDP9	Characterisation. Radioactive substances should be characterised using BAT so as to facilitate their subsequent management, including waste disposal.
RSMDP10	Storage. Radioactive substances should be stored using BAT so that their environmental risk and environmental impact are minimised and that subsequent management, including disposal is facilitated.
RSMDP11	Storage in a passively safe state. Where radioactive substances are currently not stored in a passively safe state and there are worthwhile environmental or safety benefits in doing so then the substances should be processed into a passively safe state.
RSMDP12	Limits and levels on discharges. Limits and levels should be established on the quantities of radioactivity that can be discharged into the environment where these are necessary to secure proper protection of human health and the environment.
RSMDP13	Monitoring and assessment. BAT, consistent with relevant guidance and standards, should be used to monitor and assess radioactive substances, disposals of radioactive wastes and the environment into which they are disposed.
RSMDP14	Record keeping. Sufficient records relating to radioactive substances and associated facilities should be made and managed so as: to facilitate the subsequent management of those substances and facilities; to demonstrate whether compliance with requirements and standards has been achieved; and to provide information and continuing assurance about the environmental impact and risks of the operations undertaken, including waste disposal.
RSMDP15	Requirements and conditions for disposal of wastes. Requirements and conditions that properly protect people and the environment should be set out and imposed for disposal of radioactive waste. Disposal of radioactive waste should comply with imposed requirements and conditions.

ONR, EA and NDA Guidance

The ONR, EA and NDA standards and guidance relevant to NLR related aspects of fuel transport and storage applicable in the UK are set out in Table 15. Codes and standards, regulations and guidance relevant to non-NLR related aspects of fuel transport and storage are set out in the relevant PSR Chapters.

Table 15: ONR, EA and NDA Guidance

Codes and Standards	Type
NS-TAST-GD-081 (Issue 4.1) – Safety Aspects Specific to Storage of Spent Nuclear Fuel	ONR TAG
NS-TAST-GD-024 (Issue 7.2) – Management of Radioactive Materials and Radioactive Waste on Nuclear Licenced Sites	ONR TAG
NS-TAST-GD-005 (Issue 12.1) – Regulating duties to reduce risks to ALARP	ONR TAG
The Joint Regulatory Guidance on the Management of Higher Activity Waste (HAW) on Nuclear Licensed Sites	Joint Regulatory Guidance (ONR and Environment Agencies)
Industry Guidance: Interim Storage of Higher Activity Waste Packages – Integrated Approach, NDA	Industry Guidance, NDA
ONR-GDA-GD-007 - Revision 0 - New Nuclear Power Plants – Generic Design Assessment Technical Guidance	ONR GDA Technical Guidance
ONR-GDA-GD-006 Revision 0 - New Nuclear Power Plants – Generic Design Assessment Guidance to Requesting Parties	ONR GDA Guidance

NWS Waste Packaging Specifications

NWS guidance on packaging specifications for spent fuel and radioactive waste (for this chapter, redundant in-core components) are set out in Table 16.

Table 16: NWS Guidance on Packaging Specifications

NWS Guidance Document	Type
Packaging Specifications for High Heat Generating Wastes https://www.gov.uk/guidance/high-heat-generating-waste-hhgw-specifications	NWS Packaging Specification
Packaging Specifications for Low Heat Generating Wastes https://www.gov.uk/guidance/low-heat-generating-waste-lhgw-specifications	NWS Packaging Specification

IAEA and WENRA Codes and Standards

The IAEA and WENRA standards and guidance relevant to NLR related aspects of fuel transport and storage are set out in Table 17.

Table 17: IAEA and WENRA Codes and Standards

International Codes and Standards	Type
IAEA Safety Standards – Predisposal Management of Radioactive Waste, GSR Part 5	IAEA Safety Standard (Safety Requirement)
IAEA Safety Standards – Safety of Nuclear Power Plants: Design, SSR-2/1 (Rev. 1)	IAEA Safety Standard (Safety Requirement)
IAEA Safety Standards – Storage of Spent Nuclear Fuel, SSG-15 (Rev. 1)	IAEA Safety Standard (Safety Guide)
IAEA Safety Standards – Predisposal Management of Radioactive Waste from Nuclear Power Plants and Research Reactors, SSG-40	IAEA Safety Standard (Safety Guide)
IAEA Safety Standards – Ageing Management and Development of a Programme for Long Term Operation of Nuclear Power Plants, SSG-48, 2018	IAEA Safety Standard (Safety Guide)
IAEA Safety Standards – Design of Fuel Handling and Storage Systems for Nuclear Power Plants, SSG-63	IAEA Safety Standard (Safety Guide)
IAEA Publications, Ageing Management for Nuclear Power Plants: International Generic Ageing Lessons Learned (IGALL Safety Reports Series No. 82 (Rev. 1), 2020	IAEA Publication
Waste and Spent Fuel Storage Safety Reference Levels (version 2.2)	WENRA Guidance

Appendix E UK and US Classification Schemes and Long-term Management Options for Spent Fuel and NFW

The classification schemes, storage options and disposal options that apply to the long-term management of spent fuel and redundant in-core components (referred to as NFW in Holtec documentation) arising from commercial light water reactors in the US are set out in Table 18, along with the relevant US codes and standards used for their classification.

Table 18: Classification, Storage and Disposal Options in the US

	Spent Fuel	Redundant In-core Components/ NFW
Material or waste type	Light water reactor spent fuel.	RCCAs, in-core instrumentation assemblies (ICIAs), neutron sources, etc.
Legislation/ regulation used for classification	Nuclear Waste Policy Act of 1982.	10 CFR Part 61. 10 CFR Part 72.
Definition	Fuel that has been withdrawn from a nuclear reactor following irradiation, the constituent elements of which have not been separated by reprocessing.	Typically, such waste has concentrations of certain radionuclides that exceed the Class C LLW limits in 10 CFR Part 61 and is classified as Greater than Class C (GTCC) LLW. Some in-core components, or parts thereof were they to be separated, might be classified as Class A, B or C LLW.
Storage options	Temporary storage of spent fuel at an on-site ISFSI or at a centralised ISFSI for the consolidated storage of spent fuel from a number of sites.	Temporary storage at an on-site ISFSI or at a centralised ISFSI for consolidated storage of spent fuel and GTCC wastes from multiple sites. Any Class A, B or C LLW can be promptly consigned to an appropriate disposal facility, if such a route is available.
Disposal options	Geological repository.	Generally not suitable for near surface disposal, unless classed as Class A, B or C LLW. Disposal to a geological repository typically required.

UK Government Policy [80] sets out the classification scheme for spent fuel and radioactive wastes in the UK. It states that spent fuel should not be categorised as waste whilst a future use for it can be foreseen by the owner. Spent fuel that is not to be reprocessed should be subject to interim storage, pending disposal to the UK's planned GDF. The UK's most hazardous radioactive wastes should also be managed by geological disposal, following a period of interim storage.

In the US, the near surface disposal option refers to a land disposal facility in which radioactive waste is disposed of in or within the upper 30 m of the ground surface. Strategic licensing options for the disposal of NFW to facilities at the surface, within 30 m of the surface and at intermediate depth have all been considered in the US.

In the UK, Government Policy [80] introduces an alternative strategic option potentially applicable to certain NFW components, which subject to making an environmental safety case for disposal may be suitable for consignment to a NSD facility. NSD facility options at or near to the surface and at intermediate depth are described[25].

[25] It should be noted that the consignment of highly irradiated redundant in-core components to intermediate depth disposal facilities is either practised or planned in a number of countries, such as Finland, Sweden and Japan. The intermediate depth disposal option to an NSD facility for suitable ILW was incorporated in UK policy partly in recognition of such OPEX.

In the UK, NFW components would be classified as either High Level Waste (HLW) or ILW at the point of generation upon retrieval from the reactor core for storage in the SFP, depending on whether their heat output requires consideration in the design of storage or disposal facilities. Any NFW components initially classified as HLW would be expected to decay relatively rapidly into ILW.

Some NFW components, or parts thereof, might in the future be identified as being potentially suitable for consignment to alternative waste routes following radioactive decay over the interim storage phase e.g. the top-ends of certain NFW components, that may not have been subject to such intense irradiation. It may be practicable to separate such lower activity portions of components by means of segmentation upon retrieval from interim storage, for segregation for consignment to alternative waste routes e.g. to an NSD, or potentially to the LLWR.

In the US, spent fuel and NFW can be subject to storage on-site, or transferred to a centralised, consolidated storage facility if available. Holtec is involved in developing the HI-STORE Consolidated Interim Storage Facility project in New Mexico, as a solution for the storage of spent fuel and NFW arisings from across the US[26].

Whilst no centralised, consolidated storage facilities for spent fuel arising from commercial reactors exist in the UK, the merits of such alternative strategic opportunities have previously been noted. Consolidated storage facilities have been utilised for certain ILW in the UK, such as at Magnox sites.

Whilst the classification schemes for the US and the UK differ, it can be seen that long-term management options relating to storage and disposal are broadly similar. It is also notable that the US has a broader range of potential storage, transport and disposal options.

[26] The HI-STORE is licensed by the NRC and utilises multiple UMAX systems, suitable for storage of any US origin commercial spent fuel currently stored within spent fuel pools or dry storage canisters. The facility has a 120 year service life. Initial proposed storage capacity is for [REDACTED] canisters, with potential to store up to [REDACTED] canisters. Transport of canisters to the facility could be undertaken using Holtec's HI-STAR 190 transport container, which is licensed by the NRC for the transport of spent fuel and GTCC waste.

Appendix F Fuel Route Features that Contribute to ALARP and/ or BAT

This Appendix sets out features and aspects of the fuel route and dry fuel storage system design and operation that contribute to ALARP and/ or BAT (see Table 19 below).

Table 19: Identified Features that Contribute to ALARP and / or BAT

Concept	Feature or Aspect that Contributes to ALARP and / or BAT
	Design Features:
ALARP	The design of the SFP, located below grade and entirely within the CS/ CES, provides enhanced protection against external hazards or missiles compared with a traditional PWR.
ALARP	The below grade SFP, designed with deep SFP and refuelling cavities, provides additional separation and shielding to minimise worker doses.
BAT	The atypically small SFP reduces materials requirements for the structure and components; smaller facilities tend to have lower energy and water requirements.
ALARP/ BAT	With the SFP located next to the RPV, the need for a separate spent fuel handling building and for a fuel transfer canal, which are typically required for a standard PWR, are negated. This avoids a number of operational steps with associated risks, reduces materials requirements and contributes to minimising the overall footprint and size of the plant.
	Optimised, Integrated Fuel Management:
BAT	The optimised use of the MPC/ HI-TRAC, for new fuel transport into the SFP for storage in the SFSR, negates the need for a separate in-containment rack, as is typical at a standard PWR, minimising required in-containment components and materials usage.
ALARP	The integrated fuel management strategy, utilising on-site dry storage once sufficient cooling of spent fuel has occurred, limits the accumulation of spent fuel within the SFP. This reduces the fuel related source term for any postulated accident scenarios.
ALARP/ BAT	Prompt consignment of spent fuel for on-site interim storage, with dry storage campaigns occurring every refuelling outage, contributes to minimising the release of radioactivity into the SFP. This reduces the liquid source term in the SFP and down-stream resins and filters.
ALARP	Regular dry storage campaigns contribute to enhancing worker competency, for example for MPC transfer and processing. This enables a culture of continuous improvement based on site-based OPEX, contributing to worker dose reduction.
ALARP/ BAT	The Holtec dry fuel storage process can be reversed using the same handling, transfer and processing equipment to recover an MPC from dry storage should it be necessary to do so, and return it to the SFP for spent fuel storage in the SFSRs.
	International Good Practice and OPEX for PWRs:
ALARP/ BAT	Holtec has selected the most widely used fuel type in the US reactor fleet for the SMR-300, for which extensive OPEX exists.
ALARP	The plant uses a conventional fuel bridge design for refuelling operations, consistent with international best practice for PWRs and with extensive OPEX available.
ALARP/ BAT	The primary use of single entrances/ exits to the CS via hatches for equipment and personnel respectively provides greater administrative control over access and should reduce the quantity of tools and equipment that needs to be taken into containment, minimising the generation of contaminated tools and items generated.
ALARP/ BAT	The Holtec spent fuel dry storage system is the most widely used system of its type in the world. It benefits from decades of improvements and configuration upgrades. It represents class-leading international best practice with extensive OPEX.
ALARP	The move over recent years to MPCs with greater capacities for SFAs reduces the number of lifting, transfer and handling operations required, contributing to reducing the risks of a dropped load or toppled MPC.
BAT	The use of a larger capacity MPC, for example in comparison with that used at Sizewell B, reduces the total quantity of metal required for MPC manufacture. It also minimises the required footprint of the ISFSI, as less MPCs are needed overall.
ALARP/ BAT	The HI-STORM UMAX system represents the state-of-the-art in spent fuel interim storage technology. The system has been implemented at a coastal site (San Onfre Nuclear Generating Site) and is licenced by the NRC for use at a proposed inland site for the consolidated storage of spent fuel (the HI-SAFE CIS Facility project in New Mexico, US).
ALARP	The below grade HI-STORM UMAX system contributes to ALARP by reducing risks and consequences from external hazards, such as aircraft collisions and missiles.

Concept	Feature or Aspect that Contributes to ALARP and / or BAT
ALARP/ BAT	The additional shielding afforded by the below grade HI-STORM UMAX system configuration reduces doses to workers and the public to very low levels.
BAT	The below grade HI-STORM UMAX system, similar to above ground configurations, does not require an overbuilding for the ISFSI, reducing materials requirements for the building super-structure and its visual impact.
BAT	The CECs/ VVMs within the HI-STORM UMAX system can potentially be reused following export of MPCs off-site, or alternatively be subject to metal melting and recycling, as they are only activated to relatively low concentrations of radioactivity.
BAT	Interim storage of NFW in a non-encapsulated state, either with spent fuel in MPCs for RCCAs, or in NFWCs for NFW, within the HI-STORM UMAX system, avoids the need for materials for grout reagents.
ALARP/ BAT	Non-encapsulated storage of NFW ensures items are readily retrievable and avoids foreclosing future management options e.g. decay storage, metals recycling for any suitable material, segmentation, characterisation, sorting and segregation for consignment to suitable waste routes (e.g., to the GDF, a future NSD facility or to the LLWR).

Appendix G Ageing Management Programme Case Study

An AMP can be used to predict, monitor and control the degradation of a storage system's SSCs, so that the ageing effects will not result in loss of their safety-significant function during their service life in interim storage. An effective AMP prevents, mitigates and detects the ageing effects, provides for the prediction of the extent of the effects of ageing and for timely corrective actions before there is a loss of intended function.

The AMP set out in the HI-STORE CIS Facility Licensing Report [79] for the UMAX system proposed for deployment at this consolidated interim storage project consists of four major components:

- Monitoring for emerging signs of potential degradation.
- Periodic inspection and testing to uncover onset of the SSC's degradation.
- Implementation of preventive measures (barriers) to arrest degradation.
- Recovery and remedial measures if all barriers were to fail.

The principal effects that can cause ageing of an SSC are:

- Cyclic fatigue.
- Creep.
- Erosion.
- General corrosion.
- Boron depletion.
- Crack propagation.
- Repetitive mechanical loading.
- Stress corrosion cracking (SCC).

Because the MPC provides the confinement barrier and criticality control, its AMP is the most important and is accordingly the central focus of the programme. The MPC AMP consists of visual examination, volumetric examination, accelerated coupon testing and Eddy current testing. For the HI-STORE CIS Facility, a minimum of one MPC from each originating site shall be selected for visual inspection. The MPCs with highest susceptibility for SCC should be selected for inspection. Selection criteria include oldest and coldest MPCs, with potential for accumulation and deliquescence of deposited salts that may promote localized corrosion and/ or SCC. All the accessible weld areas of the MPC will be covered for SCC inspection/ monitoring and the MPCs selected for inspection will be visually inspected for conditions listed below:

- Localised corrosion pits, stress corrosion cracking, etching or deposits.
- Discrete coloured corrosion products, especially those adjacent to welds and weld heat affected zones.
- Linear appearance of corrosion products parallel to or traversing welds or weld heat affected zones.
- Red-orange coloured corrosion products combined with deposit accumulations in any location.
- Red-orange coloured corrosion tubercles of any size.

Additional assessment is necessary for suspected areas of localised corrosion or SCC. Volumetric and surface examinations are conducted to characterize the extent and severity of localised corrosion and SCC.

As defence in depth, small coupons pre-stressed to varying levels installed in the cold (air inlet) region of the VVM cavity annulus are planned to be used at the HI-STORE CIS site to serve as an early warning system for predicting the onset of SCC or any other anomalous behaviour. The coupons shall be installed after evaluation, in VVMs that contain the oldest and coldest MPCs, where inspections are expected. In addition to oldest and coldest canisters, U-bend coupons will also be installed in the hottest canister. The hottest MPC is expected to have the highest airflow of the VVMs. Coupons withdrawn for testing shall not be reused. Monitoring and inspecting U-bend coupons is an accelerated approach of predicting degradation of MPC material. If the U-bend coupon indicates any kind of defect and/ or anomaly, the external surface of the representative canister may be tested using an Eddy current NDE technique developed by EDF Energy and Holtec to ensure the quality and integrity of the canister. The Eddy Current testing on a canister is performed by staging the HI-TRAC over the VVM cavity with a custom engineered Eddy current probe system housed in a shielded enclosure interposed between the two. The surface of the MPC is circumferentially assayed as it is progressively raised from the VVM cavity.

The AMP for the VVM of the UMAX system utilises condition monitoring to manage ageing effects of the CEC, Divider Shell and the Closure Lid as set down in the maintenance programme set out in the HI-STORE CIS Licensing Report [79]. All VVMs that contain MPCs identified in the MPC AMP shall be inspected. The visual inspection of the steel components and structures will include inspection of all internal surfaces for corrosion and integrity, and all other surfaces for dents scratches, gouges or other damage. The ISFSI Pad, Support Foundation Pad and Cask Transfer Building slab are examples of reinforced concrete structures at the proposed CIS facility. The AMP includes periodic visual inspections by personnel qualified to monitor reinforced concrete for applicable ageing effects, and to evaluate identified ageing effects against acceptance criteria derived from the design bases. The programme also includes periodic sampling and testing of groundwater, and the need to assess the impact of any changes in its chemistry on the concrete structures underground. Examples of lifting devices used with Holtec systems include the VCT. The visual inspection of the VCT will include all external surfaces for corrosion, dents, scratches, gouges, or other signs of damage which may be adverse to the structural integrity of the component. Tollgates, defined as periodic points within the period of extended operation when licensees would be required to evaluate aggregate feedback and perform and document a safety assessment that confirms the safe storage of spent fuel, are set out in the Licensing Report for the HI-STORE CIS Facility [79].

Appendix H Requirements of the Inter-Industry Guidance on Storage

A cross-industry team assembled by the NDA has developed Inter-Industry Guidance (IIG) on the interim storage of packaged HAW [82]. The IIG comprises of four primary sections covering the key elements of a robust approach to interim storage: package performance and design; store performance and design; operation of the storage system; and provision of assurance of the system over an intergenerational timescale.

Principles of Storage Systems

The following six principles provide a framework for the IIG:

- **Cradle-to-grave lifecycle.** Packages should be managed to protect their overall longevity as part of the lifecycle from container manufacture to disposal facility closure. As interim storage is transitory, packages should be readily retrievable and exportable to a finishing or rework facility (as required), a disposal facility or another store for continued storage.
- **Right Package ↔ Right Store.** Good package design should be matched by appropriate store design with due consideration of the hazards presented by the packages and the quality of storage required. The overall storage system (the waste form, the container, the store environment and its structure) should have limited need for active safety systems, monitoring or prompt human intervention. Overall value for money through both avoiding over- and under-engineering should be demonstrated.
- **Minimising waste generation.** The waste hierarchy should be deployed across the storage system lifecycle, from design through to decommissioning of the store to avoid unnecessary generation of waste while utilising resources sustainably.
- **Prevention is better than cure.** The storage system should be managed to minimise the risk that intervention will be required to maintain safety functions. The storage system should be subject to regular and proportionate monitoring and inspection to demonstrate performance and enhance the understanding of how the system may evolve in the future.
- **Foresight in design.** The storage system design should be flexible to meet likely future needs, taking into account uncertainties and incorporating proportionate contingencies.
- **Effective knowledge management.** The experiences and lessons learned from existing store operations should be shared between store operators to inform development of store standards and designs. Learning from relevant overseas storage facilities should also be utilised effectively through collaboration. In the UK, the Store Operators Forum (SOF) is the primary knowledge exchange forum.

Storage Strategy/ Implementation Plan

The IIG recommends that waste packagers should develop a Storage Strategy which defines the key safety functions of the storage system, how these will evolve and how this evolution will be monitored and controlled. The IIG also proposes a Storage Implementation Plan which defines how monitoring, inspection and intervention plans will be implemented for the specific store.

Generic SMR-300 Storage Strategy

The key parts of the storage strategy are:

- Identifying the package 'groups' likely to evolve in similar ways within the store.
- Identifying the package safety functions over the lifecycle for each package group.
- Identifying evolutionary processes that may affect the performance of the package safety functions and measurable indicators of these processes and calibrating the indicators, where practicable, to provide indicative package performance zones in order to guide appropriate actions in response to any measured or inferred evolution.
- Identifying the storage system safety functions and how these will evolve and be monitored i.e. the performance of any crane, lifting equipment, store structure, etc.

Identification of Generic SMR-300 Spent Fuel Package Groups

There are three generic SMR-300 package groups proposed for storage in the UMAX system:

- Spent fuel stored in MPCs.
- Damaged FAs in DFCs stored in MPCs with other spent fuel.
- NFW stored in NFWCs.

Generic SMR-300 Spent Fuel Package Safety Functions

There are five package safety functions that have been identified that cover the lifecycle for all three package groups. These are detailed in the UMAX FSAR [32] where they are called Design Criteria: Structural; Thermal; Shielding; Criticality; and Confinement. The safety functions, a description of the safety function and the examples of the evolutionary processes are summarised in Table 20.

Table 20: SMR-300 Spent Fuel Package Safety Functions

Safety Function (Main Component Delivering Function)	Description	Example Evolutionary Process
Structural – normal operating conditions (MPC design criteria, MPC material types, HI-TRAC design, VVM design).	Provides means of safe handling during all operations.	• Container corrosion (vicinity of lifting features). • Over pressurisation.
Structural – abnormal operating conditions (MPC design criteria and materials, HI-TRAC design, VVM design).	Provides means of safe handling during all abnormal operations.	• Container corrosion (vicinity of lifting features). • Integrity of VVM.
Thermal – normal operating conditions (spent fuel loading patterns in the MPC, peak cladding temperature).	Manage long-term evolution of the spent fuel to support containment of the radionuclides within the waste package during normal operations.	• Formation of radial hydrides. • Expansion of the fuel breaching the cladding. • Rearrangement of fuel within the MPC.
Thermal – accident conditions (spent fuel loading patterns in the MPC, peak cladding temperature, operations of HI-TRAC and VVM).	Manage peak cladding temperatures following failure of HI-TRAC and blockage of cooling system in the VVM.	• Formation of radial hydrides. • Expansion of the fuel breaching the cladding. • Rearrangement of fuel within the MPC.

Safety Function (Main Component Delivering Function)	Description	Example Evolutionary Process
Shielding (HI-TRAC and VVM).	Provide sufficient shielding to meet radiation dose limits	• Corrosion of VVM. • Failure of HI-TRAC.
Criticality (spent fuel).	Provide nuclear criticality safety through control of spent fuel content in the MPC, provision of neutron absorbers and geometric configuration of the fuel.	• Expansion of the fuel breaching the cladding. • Rearrangement of fuel within the MPC.
Confinement - normal operating conditions (spent fuel loading patterns in MPC, peak cladding temperature, MPC design criteria, MPC materials).	Contain radionuclides within the waste package during normal operations.	• Fuel expansion and cladding failure. • Container corrosion.
Confinement – impact accident (MPC design, fuel cladding).	Contain radionuclides within waste package to sufficient extent during impact accident scenario.	• Fuel expansion and cladding failure. • Container corrosion.
Confinement – fire accident (MPC design, fuel cladding, VVM design).	Contain radionuclides within waste package to sufficient extent during fire accident scenario.	• Fuel expansion and cladding failure. • Container corrosion. • Source of a fire.

Store Performance and Design

The IIG states that the primary function of a store is to store packages in a manner that protects workers, the public and the environment from hazards associated with the interim storage of the packages until they are exported. This function may be conveniently divided into two components:

- Maintain the packages through:
  - Provision of safe, secure, reliable and monitorable storage space for packages.
  - Preserving the package safety functions.
  - Ensuring the continued operation of handling, monitoring and other equipment.
  - Ensuring the integrity of key components of the storage system.
  - Retaining knowledge and records of the spent fuel and NFW, equipment and infrastructure.
  - Retaining the ability to promptly retrieve packages for export to other processing plants (e.g. for re-packaging), storage facilities or to the GDF.

- Protect workers, the public and the environment through:
  - Containment of radioactive material.
  - Protection against ionising radiation to ensure doses are ALARP.
  - Protection against criticality, where appropriate.

As can be seen in Table 20, the UMAX system safety functions are an integral part of the package safety functions. The system is passive and does not rely on any active ventilation systems to maintain the package safety functions.

Monitoring and Inspection of Package Safety Functions

Along with the development of a storage strategy which defines what matters and why, the storage strategy should detail how the SSCs that provide the Package Safety Functions will be inspected and monitored to ensure that these safety functions are evolving as expected.

Storage Implementation Plan

Once the storage strategy is agreed the guidance suggests a Storage Implementation Plan should be produced. The implementation plan takes the details of the strategy for the monitoring and inspection of package safety functions and defines how often inspections should be undertaken and the criteria limits for pass/ fail. The Storage Implementation Plan would cover baselining of the packages and the store environment for future comparison for evolution. The implementation plan would be periodically updated with the details of the inspection of the storage system. For some UK sites, good practice is that the strategy and implementation plan are a single document.