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2.1 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 United Kingdom (UK) to fulfil the future licensee's legal duties to be safe, secure and protect people and the environment, as defined in Preliminary Safety Report (PSR) Part A Chapter 1 Introduction [1].

The Fundamental Purpose is achieved through the Fundamental Objective of the PSR which is to summarise the safety standards and criteria, safety management and organisation, claims, arguments and 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 A Chapter 2 of the PSR presents the Claims, Arguments and Evidence (CAE) for the general design aspects and site characteristics that underpin the design of the generic SMR-300. Holtec International are the Requesting Party and intend to develop the SSEC with a potential Licensee's legal duties in mind, so that it is fit for use as the starting point for a future Licensee's site-specific project.

The SSEC is the logical and hierarchical set of documents that describe risk in terms of the hazards presented by the SMR-300 on the generic site (see Figure 1), and those reasonably practicable measures that need to be implemented to prevent or minimise harm to workers, the public and the environment.

[REDACTED]

Figure 1: A Twin-Unit SMR-300 at a Generic Site

2.1.1 Purpose and Scope

The Overarching SSEC claims are presented in Part A Chapter 3 Claims, Arguments and Evidence [2].

This chapter (Part A Chapter 2) links to the overarching claim through Claim 1:

Claim 1: The generic Holtec SMR-300 design, and safety case are developed using integrated safety management arrangements that take cognisance of relevant good practice in the context of the UK regulatory regime.

As set out in Part A Chapter 3 [2], Claim 1 provides an overarching justification of the design and safety principles, applicable codes and standards, safety management arrangements, historic development of the design, and the approach to ALARP. It is decomposed across the following PSR chapters: Part A Chapter 2 General Design and Site Characteristics, Part A Chapter 4 Lifecycle Management of Safety and Quality Assurance [3], and Part A Chapter 5 Summary of ALARP and SSEC [4].

This chapter describes the design evolution of the SMR-300 (see sub-chapter 2.2), presents the description of the United States (US) Reference SMR-300 Plant that forms the basis of this GDA (see sub-chapter 2.3), defines the generic SMR-300 GDA design scope, Design Reference Point (DRP), design changes that have occurred during the GDA, design challenges as a result of the SSEC at the GDA stage, and UK prospective design changes (see sub-chapter 2.4), describes the Holtec SMR-300 safety and design principles and philosophies, including US Nuclear Regulatory Commission (NRC) regulations that the SMR-300 has been designed to (see sub-chapter 2.6), outlines the UK approach to safety demonstration for the generic SMR-300 (see sub-chapter 2.7) and defines the Great Britain (GB) Generic Site Envelope (GSE) for the purposes of this GDA (see sub-chapter 2.8).

The GDA Structures, Systems and Components (SSC) scope of the generic SMR-300 Reference Design is presented within this chapter (see sub-chapter 2.4). Appendix B outlines the SSCs that are excluded from the GDA scope.

The generic design aspects and site characteristics chapter is common across the PSR and Preliminary Environmental Report (PER) and supports the Generic Security Report (GSR) and Preliminary Safeguards Report (PSgR) submissions.

Further discussion on how Claim 1 is decomposed into further Level 2 and Level 3 claims and how this chapter supports the fulfilment of these claims is provided in sub-chapter 2.5.

2.1.2 Assumptions

Assumptions which relate to the definition of the Generic Site are identified within sub-chapter 2.8.1 and have been formally captured in the Commitments, Assumptions and Requirements process [5]. Further details of this process are provided in Part A Chapter 4 [3].

This revision of the PSR assumes that a Site Justification Report or similar will be produced at the site-specific stage to characterise the prospective site and confirm that the site is bounded by the GB GSE defined in the Generic Site Envelope Report (GSER) [6]. This report should review the screening analysis presented in the GSER and undertake a detailed hazard analysis, including those External Hazards screened out of the scope of GDA.

There are no other assumptions recorded in this document.

2.1.3 Interfaces with Other Chapters

The generic design aspects and site characteristics PSR chapter interfaces with multiple areas across the Project.

Part A Chapter 2 provides the basic description of the SMR-300 and the generic site for the PSR, PER and the GSR. All other chapters refer to these descriptions. The Reference Design and Reference Plant are stated in Part A Chapter 2. Part A Chapter 4 Lifecycle Management of Safety [3] will address the design arrangements and lifecycle management of this reference design and its interactions with the reference plant. Part A Chapter 5 [4] presents the ALARP methodology and ALARP justifications for the SMR-300.

Part B Chapter 9 Description of Operational Aspects and Conduct of Operations [7] describes the operational modes and specific parameters. Part B Chapter 10 Radiological Protection [8] details the dose assessments in normal operation with Part B Chapter 14 Design Basis Analysis (Fault Studies) and Part B Chapter 16 Probabilistic Safety Assessment (PSA) covering accident dose. Part B Chapter 11 Environmental Protection [9] summarises the environmental impacts described in the PER. Part B Chapter 13 Radioactive Waste Management [10] summarises the solid, liquid, and gaseous wastes generated by the operation of the SMR-300. External Hazards defined for the generic site envelope are used as a key input to Part B Chapter 21 External Hazards [11].

Part A Chapter 2 presents the overall approach to the UK Categorisation and Classification assessment (see sub-chapter 2.7.6). The detailed approach is presented within Part B Chapter 14 Design Basis Analysis (Fault Studies) which identifies some preliminary UK SSC classifications. Discussion of the appropriateness of codes and standards is presented throughout the Part B engineering chapters.

Part A Chapter 2 interfaces with all PER chapters and all Part B chapters.

2.1.4 Project Definitions and Abbreviations

Abbreviations used throughout the SSEC are presented in Appendix A.

2.2 US REFERENCE SMR-300 PLANT DEVELOPMENT

2.2.1 US Reference SMR-300 Plant Design Status

This sub-chapter covers the historic evolution of the Holtec SMR design to its present 300 Megawatts Electric (MWe) configuration. It identifies the history of the design evolution, the current design status of the SMR-300 and introduces the deployment of the Palisades SMR-300 as the first dual-unit plant at an existing nuclear site, Palisades, in Covert, Michigan in the US.

2.2.1.1 Historic Evolution of the Design

Starting in 2011, shortly after the accident at Fukushima Daiichi, Holtec International embarked on a SMR design programme to bring a compact safe, affordable, clean nuclear power system to world markets, that would cure shortcomings in contemporary large Light Water Reactor (LWR) offerings.

The original concept yielded a preliminary design specification for a 160 MWe Nuclear Power Plant (NPP) with an innovative nuclear steam supply system comprising a co-joined and offset Reactor Pressure Vessel (RPV) and once-through Steam Generator (SGE) relying entirely on gravity driven natural circulation for power operations. This offset Reactor Coolant System (RCS) configuration, paired with passively[1] actuated and operated engineered safety systems, resulted in a design significantly safer than large LWRs licensed and operating around the world when comparing calculated core damage frequencies.

The design possessed a novel RCS geometry and novel forging welded to the RPV upper shell and lower steam generator inlet plenum not previously endorsed or licensed. Accordingly, Holtec perceived the regulatory risk too great to proceed with the design, opting for a more conventional RCS geometry that would minimise the potential consequences of a limiting double-ended guillotine large break Loss of Coolant Accident (LOCA), while dramatically increasing the power plant efficiency and economics with a high mass flux flow solution. Additionally, this design preserved the engineered passive safety features and safety characteristics of the design.

On September 1, 2023, direction was given by the Holtec Executive Leadership to upgrade the SMR-160 design from approximately 160 MWe to 300 MWe using reactor coolant pumps and external RCS piping.

2.2.1.2 Design Freeze

HPP-160-3037, Holtec's Design Evolution and Freeze Process [12], laid out the steps taken to transition the design from the SMR-160 to new conceptual layouts of the SMR-300 RCS, Engineered Safety Features (ESF), Containment Structure (CS), Annular Reservoir (AR), Containment Enclosure Structure (CES) and Reactor Auxiliary Building (RAB), Control Building (CB) and all other SSCs affected by the design evolution. Early in GDA Step 1, the SMR design was modified to a 300 MWe design incorporating pumped circulation. This

[1] The actuation by battery power is categorised as a Category D Passive Safety System in International Atomic Energy Agency TCS-69 [112].

procedure was used to control transition of the design from the SMR-160 to SMR-300 to establish a new reference design at a conceptual level. Further details of the Design Freeze process are provided in Part A Chapter 4 [3].

The US Reference SMR-300 Plant is the GDA Input Reference Design. The generic SMR-300 GDA design scope is described in sub-chapter 2.4.

2.2.2 Palisades US SMR-300 Plant Programme

Holtec have started a programme to build the first SMR-300 reactor units at the Palisades site in Michigan, US. The addition of a Holtec SMR-300 power plant near the existing plant will nearly double the Michigan site's total carbon-free generation capacity.

Siting the first SMR-300 power plant at Palisades reduces time when compared to erecting the plant at an undeveloped property, and confers the many benefits of synergy that accrue from the presence of a co-located operating plant - including shared infrastructure and operational expertise, enhancements to grid stability, and resource optimisation.

A US NRC Construction Permit Application (CPA) grants permission to construct the reactor. The documentation needed to support this includes (but is not limited to):

- Environmental Report (ER) (in accordance with 10 Code of Federal Regulations (CFR) 51.50 [13]).
- Preliminary Safety Analysis Report (PSAR), preliminary security plan, preliminary emergency plan (in accordance with 10 CFR 50.34 [14]).
- Technical Specifications (in accordance with 10 CFR 50.36 [14]).

Comprehensive safety assessments are documented in the US PSAR, such that:

- The PSAR is organised in a comprehensive format and content defined by US NRC's NUREG-0800 [15].
- Safety conclusions of the PSAR are substantiated by design information developed.
- Provides the necessary vehicle for independent comprehensive peer review of total assessment.
- Provides a structured assessment package in a format which enables well-informed decisions.

Similarly, for an Operating Licence Application (OLA), the documentation to support the operation of the reactor includes (but is not limited to):

- Supplement to ER for operating license (in accordance with 10 CFR 51.53 [13]).
- Final Safety Analysis Report (FSAR), physical security plan, emergency plan (in accordance with 10 CFR 50.34 [14]).

[REDACTED]

The Palisades build programme will provide valuable Learning from Experience (LfE) to be used in any future UK deployment of the SMR-300 beyond the GDA process. This information flow is captured in Part A Chapter 4 [3].

2.3 US REFERENCE SMR-300 PLANT DESCRIPTION

The SMR-300 is an advanced, Pressurised Water Reactor (PWR) NPP, incorporating two reactors in a single power plant layout, with a design informed by decades of operating reactor experience and industry lessons-learned, able to provide clean and affordable power with passive[2] safety systems and improved safety compared to presently operating nuclear plants. The plant design specification is risk- and value engineering-informed to facilitate a readily licensable and competitive power plant product embodiment, planned for deployment both in the US and international markets.

An SMR-300 reactor is a two-loop PWR designed with forced circulation in normal operation, utilising two cold legs each with a vertically mounted Reactor Coolant Pump (RCP), two hot legs, and a single once-through SGE with an integral pressuriser stacked on top of the SGE. The plant design is simplified relative to operating plants and incorporates passive and robust safety systems to enhance its safety, construction, operation, and maintenance. The use of passive[2] safety systems results in a highly reliable, safe design, which protects workers, the public and the plant. Additionally, the SMR-300 is designed to eliminate or simplify inspections, testing, and maintenance, which reduces operating costs. The following sections provide a summary of SMR-300 Plant Overview [16].

2.3.1 Site Layout and Main Buildings Description

The SMR-300 is a compact plant arrangement, as shown in Figure 1, occupying approximately [REDACTED].

2.3.1.1 Nuclear Island

The Nuclear Island (NI) for the twin-unit SMR-300 consists of the following buildings:

- Containment Enclosure Structure, housing the Containment Structure (1 for each reactor unit).
- Reactor Auxiliary Building (including the Control Room Area).
- Electrical Building for NI.
- Intermediate Building (IB).

Each SMR-300 CS and CES share a common basemat and are connected to the shared RAB. The CES surrounds the CS. The site layout ensures that nuclear assets, including the reactor core, are deeply embedded for protection against hazards. Unauthorised access is prevented by multiple barriers. All critical assets are located inside the site boundary fence, which also defines the emergency planning zone.

2.3.1.2 Conventional Island

The Conventional Island (CI) for the twin-unit SMR-300 consists of the following buildings:

- Electrical Building for Turbine Island.

[2] The actuation by battery power is categorised as a Category D Passive Safety System in IAEA TCS-69 [112].

- Turbine Building.
- Unit Auxiliary Transformer.
- Main Step-Up Transformer.
- Station Service Transformer.

2.3.1.3 Balance of Plant

The Balance of Plant (BOP) is all infrastructural facilities except for the main product producing facilities in a plant. A BOP is generally used in a power project to define all supporting facilities and auxiliary systems of the power plant needed to deliver the electricity, other than the generating unit itself. In the power plant, a BOP includes transformers, inverters, supporting structures, and control and monitoring systems of the entire plant, but not the turbine, generator, and generator step-up transformer, and all its elements. The BOP is generally excluded from the scope of the GDA (see sub-chapter 2.4 for further details).

If an assumption related to the performance of the BOP SSCs is required to bound the safety assessment presented, then the assumption will be clearly stated and recorded in the safety assessment and captured in the project assumptions register. This is captured in the Holtec procedure: Holtec SMR-300 Generic Design Assessment Capturing and Managing Commitments, Assumptions and Requirements [5].

2.3.1.4 SMR-300 Twin-Unit Shared Facilities

The following facilities are shared in the twin-unit arrangements:

- RAB, including Main Control Room (MCR) and Remote Shutdown Facility (RSF).
- Radioactive waste systems.
- The water and wastewater treatment systems such as demineralised water, raw water, and sanitary systems.
- Site drainage, sewer, and similar utilities.
- The non-safety stand-by diesel generator system.
- Security, communications, and Information Technology (IT)-related systems.
- Technical support centre.
- Dry spent fuel storage system.
- Operations (security staff, office building, parking lots, and other common features).

2.3.2 Operating Envelope and Plant States

The SMR-300 is designed with a wide operating envelope to provide stable operation. For example, during a 10% step change in steam demand (or reactor power) the pressuriser is sized to be sufficiently large to maintain the pressuriser level in the normal operating band. This increase in operational margins results in a more reliable plant with fewer reactor trips that can challenge plant equipment and operators. The large pressuriser also eliminates the need for power operated relief valves, which are sources of RCS leakage.

Normal operation of the SMR-300 is defined as when the plant is within its specified operating conditions / limits. The typical plant operating modes are defined in the Standard Technical Specifications to comply with 10 CFR 50 [14] paragraph 50.36 Technical specifications. As the operating modes have not yet been fully defined, the typical Mode 1 Power Operations mode has been split to coincide with the licensing basis events definitions in the Standard Review Plan NUREG-0800 Chapter 15 [15].

The alignment with UK plant condition classes is shown in Part B Chapter 14 Design Basis Analysis (Fault Studies) [17]. Holtec have defined the operating state Abnormal Operations to align with those Anticipated Operational Occurrences (AOO) occurring once per reactor lifetime.

The following plant states and conditions are considered in the design of the SMR-300 in accordance with NUREG-0800 Chapter 15 [15].

1. **Normal Operation (NO):** Operation within specified operational limits and conditions. For a NPP, this includes startup, power operation, shutdown, maintenance, testing, and refuelling. More details of these operations are provided in Part B Chapter 9 [7].
2. **Anticipated Operational Occurrence:** condition of normal operation which is expected to occur one or more times during the life of the nuclear power unit.
3. **Design Basis Accident (DBA):** A postulated accident that a nuclear facility must be designed and built to withstand without loss to the systems, structures, and components necessary to ensure public health and safety.
4. **Design Extension Condition (DEC):** A subset of beyond-design-basis accidents that are considered in the design process of the facility in accordance with best-estimate methodology to keep releases of radioactive material within acceptable limits. DECs could include severe accident conditions.

Those accidents considered beyond design basis accidents, not defined within the subset of DECs, are not considered in the design process, however the consequences of these beyond design basis accidents are assessed. Very low frequency faults beyond the design basis are not considered deterministically but are considered via the PSA, to provide a full understanding of the residual risks associated with NPP operations. Sub-chapter 2.6.5 describes the US safety analyses undertaken in support of the SMR-300, and sub-chapter 2.7 describes the UK safety analyses undertaken within this GDA.

Development of the conduct of operations to meet UK regulatory requirements, including the definition of Normal Operating States, is presented in Part B Chapter 9 [7].

2.3.3 Plant Parameters

Table 1 below provides a comprehensive set of plant parameters for the SMR-300. Section 7 of the SMR-300 Plant Overview [16] provides further details on the design parameters.

Table 1: SMR-300 Plant Parameters

[REDACTED]

2.3.4 Plant Breakdown Structure

This Holtec procedure identifies the Numbering, Tagging and Plant Breakdown Structure (PBS) for the SMR-300 [18]. The PBS identifies the system groups that define the plant, which are given in Table 2. Using this as a starting point, the PBS for the SMR-300 is visualised in Appendix B with each of the systems that are in each system group, and they are described in Part B of the PSR according to the table in Appendix C. The SMR-300 Plant Overview [16] provides further details on key systems within the SMR-300 main structures.

Table 2: SMR-300 System Groups

System Group
Reactor Systems
Reactor Coolant System
Auxiliary Systems
Engineered Safety Features
Instrumentation & Controls
Mechanical Handling Systems
Electrical Systems
Radioactive Waste Systems
Water and Waste Treatment Systems
Heating, Ventilation, and Air Conditioning (HVAC)
Power Conversion
Dry Fuel Storage

There are differences between the PBS outlined in Appendix B and the SSCs listed in the Numbering, Tagging and PBS for the SMR-300 [18]. These differences are described in detail within the Numbering, Tagging and PBS for the SMR-300 [18] however, the delta's have been summarised below:

- RCP has been eliminated as an independent system and is now covered as part of the RCS.
- Radioactive Waste Building (RWB)[3] Crane removed and RWB HVAC system removed.
- Hydrogen cooling system added for generator cooling.
- A number of differences relating to the HVAC systems interfacing with the control room, due to ongoing development of the US design. For the purpose of GDA the MCR Habitability system (MCH) and Control Room Normal Ventilation are still considered as independent systems, as system design descriptions have not yet matured to reflect potential changes to the HVAC architecture.
- Plant Safety System (PSS) and Diverse Actuation System (DAS) I&C systems changed to unitised.

[3] The RWB and RAB have merged. This is discussed further in 2.4.1.

The SMR-300 twin-unit site consists of the following principal structures, which are identified within the 'Buildings and Structures' group in the PBS.

2.3.4.1 Containment Enclosure Structure and Associated Systems

The CES is a Seismic Category I structure that surrounds the CS and is designed for the following functions[4]:

- Protects the CS from external hazards and threats.
- Provides shielding to the plant and personnel from radioactive sources inside the CS during power operations and postulated accidents.
- Forms the outer wall of the AR to retain the AR water inventory for Passive Containment Heat Removal during a LOCA.
- Provides a vent for the AR to facilitate evaporative cooling.
- The CES interfaces with the Intermediate Building, which protects the main steam and main feedwater lines until their respective safety isolations and seismic restraints. The CES provides support for these lines at the CS penetration.

The CES and associated systems are shown in both Figure 4 and Figure 5 in plan and layout perspectives.

The CES is a modular steel-concrete structure that shares a common basemat with the CS. Two concentric steel shells form the inner and outer faces of the modules, with interconnecting plates providing support. Each section is shop-fabricated and transported to the site where it is welded to adjacent sections to form CES rings, which are stacked and filled with concrete. Containment penetrations and personnel access to the containment are made via penetrations in the below-grade section of the CES.

2.3.4.2 Annular Reservoir

The AR is the SMR-300's ultimate heat sink (as defined in accordance with the NRC guide DG-1275 Ultimate Heat Sink for Nuclear Power Plants [19]).

The AR is the above grade annulus space between the CS and the CES containing a substantial captive body of water and is shown in Figure 5.

Its primary function is to provide a heat sink to remove heat from the core via the Passive Core Cooling (PCC) system, the Spent Fuel Pool (SFP), and the containment via the Passive Containment Heat Removal System (PCH) during Design Basis Accident LOCAs and non-LOCAs.

During a postulated high energy release (e.g., a LOCA), steam is released into the containment atmosphere. The steam condenses on the inside surface of the CS wall as heat is transferred through the CS wall to the AR. The large heat transfer area and high conductance of the steel CS results in rapid heat rejection from the containment atmosphere

[4] A design change to the CES and CS general arrangement is subject to UK Prospective Design Change (see sub-chapter 2.4.2).

to the AR. As the water in the AR is heated it rejects heat to the environment through the discharge of clean vapour (no radioactivity) through the vent at the top of the CES.

The CES vent allows water evaporating from the AR to escape out of the top of the CES during a postulated design basis accident. The vent is equipped with a honey-comb structure that prevents the intrusion of any incident projectiles and is covered to prevent ingress of fauna, rain, and debris. The vent has a hatch for personnel access into the AR. Ladders and platforms are provided to allow for inspection and maintenance of the CS and AR. The vent is aligned with the SGE to support SGE replacement when necessary. Figure 2 shows the arrangement of the AR.

2.3.4.3 Containment Structure and Associated Systems

The CS is a cylindrical steel containment vessel with a domed upper head and steel-lined reinforced concrete base[5]. It is partially embedded below grade. The CS is a Seismic Category I structure. The above-grade portion of the CS is reinforced with internally mounted stiffener rings.

The primary functions of the CS are to:

- Provide an essentially leak-tight barrier to contain fission product releases from the reactor coolant pressure boundary during operational states and design basis events.
- Contain the mass and energy release from a postulated LOCA and secondary-system pipe ruptures.
- Contain and support the RPV, RCS, SFP, and associated SSCs.

The CS has an equipment hatch at ground level to facilitate replacement of major components and for access during refuelling and maintenance outages. The containment equipment hatch is a bolted and gasketed round hatchway, providing access to the CS interior from the RAB. A personnel airlock provides an additional entrance during outages and power operations. The CS is designed to have a clear path above the SGE to the domed upper head; the SGE is aligned with the CES vent to facilitate replacement when needed and to support equipment removal during decommissioning. For most refuelling and maintenance activities, access to the CS is through the ground-level containment equipment hatch.

The CS houses the following systems:

- RCS.
- PCC which consists of:
  - PDH.
  - SDH.
  - Automatic Depressurisation System (ADS).
  - Passive Core Makeup Water System (PCM).
- Containment Building Ventilation System (CBV).
- Combustible Gas Control System (CGC).

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[5] A design change to the CES and CS general arrangement is subject to UK Prospective Design Change (see sub-chapter 2.4.2).

- SFP.
- Light Load Handling System (LLH).
- Overhead Heavy Load Handling System (CSH).

The CS and associated systems are shown in both Figure 4 and Figure 5 in plan and layout perspectives. The RCS is shown in Figure 3 along with the coolant flow paths.

The CS features a polar crane, supported by the CS shell. The crane is equipped with a primary hook and an auxiliary hook. Equipment hatches are provided at each floor elevation to allow for vertical movement of equipment. The floor equipment hatches are aligned such that the polar crane can lift equipment directly from the lowest elevation. The crane access and capacity are designed for refuelling operations, including handling of the reactor internals and equipment needed for dry storage of spent fuel using the HI-STORM Underground Maximum Capacity System (UMAX) System (see sub-chapter 2.3.4.6).

A bridge crane located on the refuelling / operating deck is used for fuel movements in and between the SFP and the RPV. The bridge spans the SFP and runs on rails set into the edge of the SFP. The crane is equipped with a fuel mast for fuel movement and an auxiliary hoist for non-fuel related operations in the pool.

2.3.4.4 Reactor Auxiliary Building and Associated Systems

The RAB is a three-level building with one level at grade and two levels below grade. There are two main sections of the RAB. The non-radiologically controlled area houses the MCR and most of the electrical equipment and switchgear. The Radiologically Controlled Area (RCA)[6] contains systems to support normal primary plant operations. The RAB is a seismic Category I structure and is adjacent to each CES. The RAB is designed to perform the following functions:

- Provides protection and separation for the seismic Category I mechanical and electrical equipment located outside the containment.
- Houses SSCs that are critical for the normal operation of the plant, as described below.

The RAB plan arrangement is shown in Figure 4.

2.3.4.4.1 Radiologically Controlled Area

The RCA of the RAB contains the systems and equipment that are or may be radioactive or contaminated. The RCA extends to all three levels of the RAB, including the Fuel Handling Area (FHA) on the grade floor and below grade elevations housing plant auxiliary systems.

The major equipment, systems, areas, and functions contained within the RCA are:

- RSF.
- Main equipment hatch and personnel hatch for entry to the CS.
- Residual Heat Removal System (RHR).

[6] Note that this RCA definition is based upon the SMR-160 design. PSR Chapter B10 provides further details of the development of the SMR-300 zoning scheme, and zoning of building layouts will not be completed until source terms are defined and shielding and contamination assessments are completed.

- Chemical and Volume Control System (CVC).
- Mechanical equipment rooms.
- Containment mechanical piping penetration areas.
- Containment electrical penetration areas.
- FHA.
- Chilled Water System (CWS) and select ventilation systems.
- SFP Cooling System.
- Some radiological waste handling equipment and functions.

UK defined RCAs will be discussed using the terms External Radiation Controlled Areas (ERCA) and Contamination Controlled Areas (CCA) to cover external-radiation and contamination-controlled areas (see sub-chapter 2.7.4).

2.3.4.4.2 Fuel Handling Area

The FHA is housed in the RCA. The FHA has:

- An area for receiving and inspecting new fuel assemblies.
- An area for storing new fuel assemblies.
- An area for loading new fuel assemblies into a HI-TRAC transfer cask which is then transported into the CS.
- An area to prepare spent nuclear fuel for dry storage.

The FHA features a bridge crane to support new fuel transfer activities. One bay of the FHA is allocated to receive new fuel from trailers or rail cars. Space is provided for inspection of the assemblies. The FHA includes provisions to use a low-profile transporter to transfer casks in and out of the CS through the main equipment hatch. A separate bay is provided to prepare spent fuel for dry storage. Spent fuel is stored in the SFP inside the CS, then transferred to the FHA for processing in preparation for onsite interim dry storage using the HI-STORM UMAX system in the Interim Spent Fuel Storage Installation (ISFSI) (see Section 2.3.4.6). Spent fuel is not stored in the FHA.

2.3.4.4.3 Non-Radiologically Controlled Area

The non-radiologically controlled portion of the RAB is contained in the two upper levels of the RAB (the ground level and the first of the two below-grade levels), adjacent to the RCA. Its primary purposes are to support the MCR and house the safety-related electrical systems.

The non-radiologically controlled area of the RAB contains the control point into the RCA. The control point is located on the ground level and is provided with decontamination equipment. Safety-related instrumentation and control cabinets, motor controllers, switchgear, and battery banks are housed on both levels. The control rod drive system electrical equipment is located on the lower level. The two divisions of safety-related electrical equipment are separated by elevation; one division is located on the lower level, while the other is located on the ground level.

Note that these areas may be classified as radiologically controlled under UK legislation (see sub-chapter 2.7.4).

2.3.4.4.4 Main Control Room

[REDACTED]. The MCR is designed to provide a habitable area from which to safely operate the reactor. A kitchen, restroom facilities, conference room, and office are also provided adjacent to the MCR within the non-radiologically controlled portion of the RAB.

[REDACTED]

The MCR's location below grade improves security and provides better protection from external hazards such as high wind events. [REDACTED]

2.3.4.5 Intermediate Building

The CES interfaces with the IB, which is a structure that protects the main steam and main feedwater lines until their respective safety isolations and seismic restraints. The CES provides support for these lines at the CS penetration. The AR contains a raised section which allows the IB to interface with the CS. The IB contains main steam and main feedwater piping and connects the CS to the Turbine Building.

2.3.4.6 Interim Spent Fuel Storage Installation

The SMR-300 includes an integrated fuel management system for handling and moving new and spent fuel. For spent fuel, SMR-300 utilises onsite interim spent fuel storage within Holtec's HI-STORM UMAX[7] system, containing an optimized Multi-Purpose Canister (MPC). HI-STORM UMAX is an underground Vertical Ventilated Module (VVM) dry spent fuel storage system. 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 overpack, is comprised of a cavity enclosure container and closure lid, as well as interfacing structures.

The MPC is a fully welded stainless-steel canister providing a safe containment of SMR-300 spent fuel onsite or for offsite transport. It utilizes a honey-comb fuel basket comprised of Holtec's proprietary material Metamic HT™ to provide positions for SMR-300 fuel assemblies. The MPC is tailored to balance the SFP size inside containment and 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 onsite storage after as little as 3 years of cooling in the SFP. This continuous integrated fuel management solution from first startup ensures a small volume of spent fuel inside the CS, with an associated small source term, as compared to conventional LWRs. The HI-TRAC is a steel, lead, and water-shielded transfer cask which houses the MPC during onsite transfer prior, to placing the MPC in the UMAX. It provides shielding to workers during loading operations and protects the MPC from DBAs. Lifting, handling, processing, and transportation equipment is designed to efficiently move spent fuel from the SFP to the UMAX at the ISFSI. The ISFSI concept layout is shown in the image in Figure 6.

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[7] Note HI-STORM UMAX GDA scope (sub-chapter 2.4.1.2).

[REDACTED]

Figure 2: Arrangement of the Containment Enclosure Structure, Annular Reservoir and Containment Structure [20]

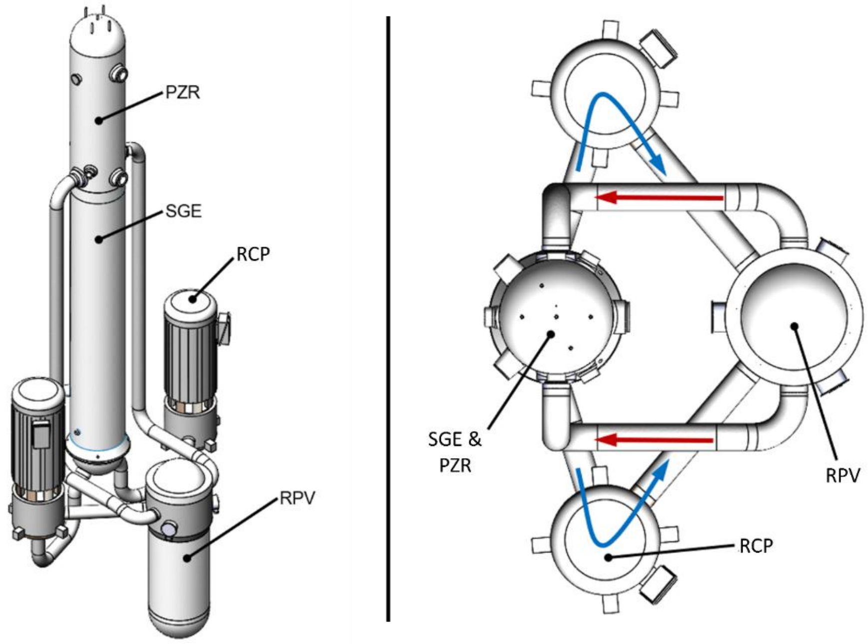

Figure 3: Reactor Coolant System Components and Primary Loop Flow Paths

[REDACTED]

Figure 4: Plan Arrangement of Nuclear Island Structures[8]

[REDACTED]

Figure 5: General Arrangement of the Refuelling Floor of the Containment Structure and Containment Enclosure Structure

[8] Extracted from the Part B Chapter 20 [111].

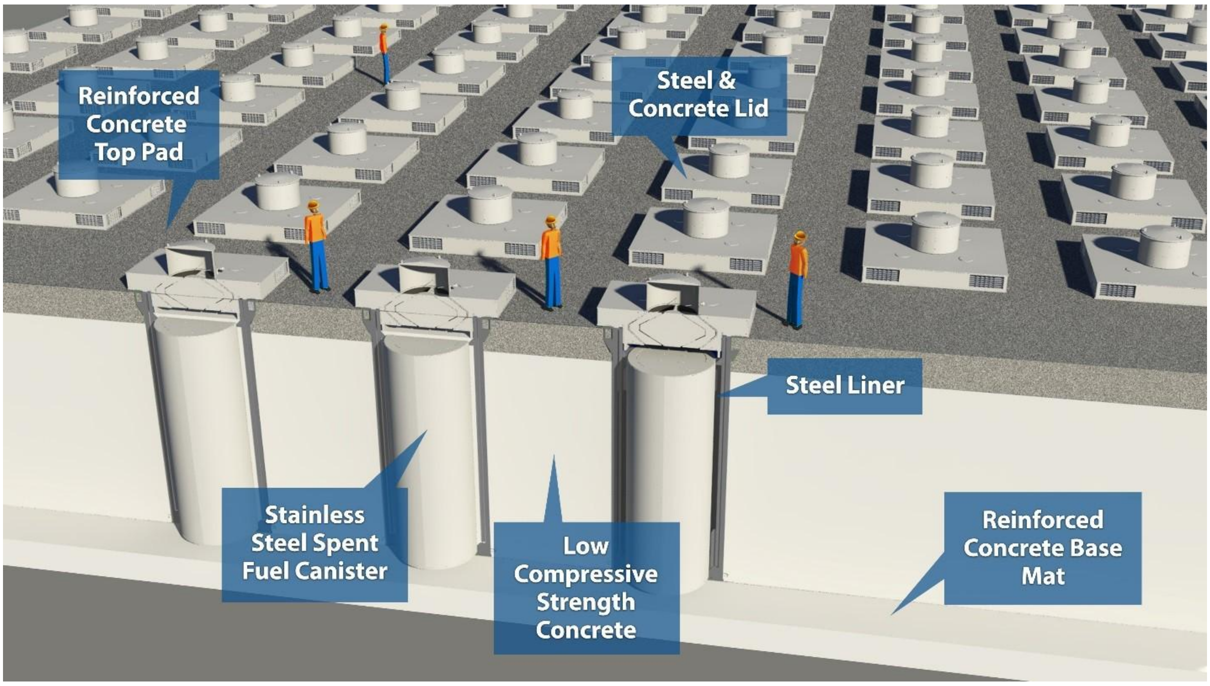

Figure 6: HI-STORM UMAX Arrangement for the Interim Spent Fuel Storage Installation

2.4 GENERIC SMR-300 GDA DESIGN SCOPE

2.4.1 GDA Design Scope

The SMR-300 PBS is introduced in sub-chapter 0, with high level design description. Appendix B presents the PBS and those SSCs that are within the GDA scope, as defined within the GDA Scope Report [21]. The GDA scope comprises those items important to:

- Safety, i.e., SSCs (direct or supporting), that provide a safety function according to the SSC Classification Standard [22].
- Environmental protection, i.e., those SSCs that have role in ensuring environmental protection.
- Security, i.e., SSCs that have a security function.

The development of the GDA Scope Report is a key input to the development of the GDA Reference Design (see sub-chapter 2.4.2), which formally defines the design documentation to be assessed within the GDA process.

During the course of GDA Step 2, and subsequent to issue of the GDA Scope Report in GDA Step 1, there were instances where it was agreed with the regulators to adjust the technical scope of the GDA. These adjustments were managed via the Design Management Procedure [23] and required a dedicated GDA Scope Change document to be produced, providing detail, justification and discussion of the impact on the GDA Scope. Two such changes were agreed to the GDA Scope and are reflected in the presentation across the SSEC v1.

2.4.1.1 Combination of RWB and RAB [24]

Due to ongoing development of the US Design, a decision was taken to move the activities and systems undertaken in the RWB into the RAB. This decision was taken by the SMR-300 Design Authority, soon after the first GDA DRP [25] was defined. The Requesting Party made the decision to trigger a GDA Scope and DRP change to update the GDA DRP, ensuring a more meaningful GDA assessment. A Decision Paper on the Absorption of the Radioactive Waste Building Functions into the Reactor Auxiliary Building [26] provides further detail of the impact this will have on the systems housed within the RWB. The scope change is reflected in relevant PSR chapters, in particular Part B Chapter 13 Radioactive Waste Management [10] and Part B Chapter 22 Internal Hazards [27].

2.4.1.2 Reduction in GDA Scope for the HI-STORM UMAX System [28]

This scope change had no impact on the GDA Design Reference Point being considered in GDA but focused the GDA Scope on establishing a ‘design envelope’ for the generic SMR-300 spent fuel management strategy, demonstrating that the generic SMR-300 design is licensable and permittable, rather than a more detailed technology assessment. The scope change is reflected in relevant PSR chapters, in particular Part B Chapter 24 Fuel Transport and Storage [29].

2.4.2 GDA Design Reference Point

This sub-chapter presents the Design Reference (DR) as defined in the DRP Report [30], which is required by Generic Design Assessment: Guidance to Requesting Parties [31]. The process for establishing and changing the DRP is explained in PSR Part A Chapter 4 Lifecycle Management of Safety and Quality Assurance [3].

Design development is essentially "normal business", where new design information has been issued (e.g. updated Piping and Instrumentation Diagram (P&ID), System Design Descriptions (SDD)) as a result of ongoing design work that can be utilised to support the SSEC. The procedure for controlling SMR-300 and GDA design changes is further explained in PSR Part A Chapter 4 Lifecycle Management of Safety and Quality Assurance [3].

2.4.2.1 GDA Design Reference Point 0

[REDACTED]

2.4.2.2 GDA Design Reference Point 1.0

[REDACTED]

2.4.2.3 GDA Design Reference Point 1.1

[REDACTED]

2.4.3 Ongoing UK Design Challenges and SSEC Alignment

[REDACTED]

[REDACTED]

Figure 7: Management of GDA DRP and SSEC Revision 1 Alignment

2.4.4 GDA Design Challenges

Table 3 provides a summary of all the design challenges identified within the SSEC Revision 1 as a result of the fundamental assessment. These challenges are discussed in further detail, alongside any resultant GDA Commitments to further progress them, within the identified PSR chapters.

Table 3: SMR-300 GDA Design Challenges

[REDACTED]

2.5 GENERAL DESIGN ASPECTS AND SITE CHARACTERISTICS CLAIMS, ARGUMENTS, EVIDENCE

The primary purpose of a CAE approach is to capture the golden thread of a safety case narrative demonstrating how plant and operational evidence is brought together to justify that a high-level or fundamental claim is true. In the context of the generic SMR-300, that is how the Fundamental Purpose of the SSEC (presented in Part A Chapter 1 [1]) is achieved.

The Fundamental Purpose follows a golden thread throughout the SSEC to CAE via the objectives of the PSR, PER, GSR and PSgR. This linkage is holistically presented in Part A Chapter 3 [2].

The PSR Fundamental Objective links to Overarching SSEC Claim 1.

Claim 1: The Generic Holtec SMR-300 design, and safety case are developed using integrated safety management arrangements that take cognisance of relevant good practice in the context of the UK regulatory regime.

This PSR chapter supports Claim 1 through the following Level 2 claims:

Claim 1.1 supports Claim 1 by demonstrating that the design principles, codes and standards used in the design of the SMR-300 have been selected to meet the stringent requirements of the US NRC. These mature and established US codes and standards are internationally recognised and are commensurate with the importance of the safety functions being delivered. The codes and standards applied to the design of nuclear safety related SSCs of the SMR-300 are generally nuclear specific, many of them are from existing practices adopted on UK nuclear licensed sites and / or application in earlier successful GDAs.

Claim 1.2 supports Claim 1 by showing that the design and safety principles being used to develop the US SMR-300 Reference Plant broadly align with UK context expectations in order to provide assurance to a future licensee that it will ultimately be able to demonstrate the generic SMR-300 design against its own safety and design principles; demonstrate that the SSEC has assessed the generic SMR-300 design and demonstrated the equivalency of the codes and standards utilised in the design of the US SMR-300 reference plant and UK codes and standards; that is suitably optimised for location specific (UK regulatory requirements); and to ensure that a robust assessment process is in place such that [REDACTED] risks associated with the SMR-300 deployment in the UK are appropriately assessed in support of the overall ALARP demonstration. The emphasis of Claim 1.1 within the SSEC will likely reduce when the demonstration of Claim 1.2 is complete beyond GDA. The generic SMR-300 is defined by sub-chapter 2.4.

Claim 1.3 supports Claim 1 by defining the GB GSE for the generic SMR-300. In order to support future operation of the generic SMR-300 reactor site at a generic UK location, it is necessary to define the environmental characteristics of the site. For the GDA to be of benefit, the defined site envelope must present characteristics which are suitably bounding of any potential future sites in Great Britain. The definition of the generic site ensures that the generic SMR-300 can be shown to satisfy UK regulatory expectations and legislative requirements and ensures that the generic SMR-300 SSCs will be adequately substantiated.

Claims 1.4 and 1.5 also support Claim 1. Claim 1.4 shows within Part A Chapter 4 [3] that there are appropriate integrated project, quality, design and safety management arrangements, and Claim 1.5 shows within Part A Chapter 5 [4] that an appropriate ALARP methodology is applied to the design change process, to ensure ongoing design decisions support the reduction of risks to ALARP.

Table 4 shows in which section of this PSR chapter these claims are demonstrated to be met.

Table 4: CAE Chapters Sections

Claim No.	Claim	Chapter Section
1.1	The US Reference SMR-300 Plant design is derived from US design and International good practice to demonstrate compliance with US NRC requirements.	2.6 US SMR-300 Reference Plant Safety and Design Principles
1.2	The generic SMR-300 design is developing to ensure compliance with UK nuclear safety and design principles while minimising the impact on the design stability of the global fleet.	2.7 UK Approach to Safety Demonstration for the Generic SMR-300
1.3	An appropriately conservative and bounding GB-context generic site envelope is derived for the generic SMR-300 GDA.	2.8 Generic Site Envelope

A summary of the current CAE route map for Part A Chapter 2 is provided in Appendix D and a further update on claim decomposition, argument development and evidence maturity will be provided in the subsequent update of the chapter. Some claims are supported entirely from evidence within this chapter; however, links have been provided to other chapters where this is not the case.

2.6 US SMR-300 REFERENCE PLANT SAFETY AND DESIGN PRINCIPLES

The SMR-300 design is based on proven technology to reduce first-of-a-kind engineering to minimise technology development and licensing risks. The SMR-300 design draws on the operating experience and lessons learnt from six decades of operating nuclear power plants, resulting in a simplified plant with respect to construction, operation, inspection, and maintenance as compared to Gen-II and Gen-III LWRs.

This sub-chapter explains the basis for the design requirements and principles that have been applied to the US SMR-300. UK regulatory and site specific requirements will be derived for the generic SMR-300 as part of this GDA as these represent location specific requirements, and are discussed further in sub-chapters 2.7 and 2.8.

Claim 1.1. The US Reference SMR-300 Plant design is derived from US design and International good practice to demonstrate compliance with US NRC requirements.

Claim 1.1 has been decomposed into four arguments:

Argument 1.1A-1: The SMR-300 reference plant is designed to be compliant with applicable US regulations.

- Sub-chapter 2.6.2 provides the safety philosophy embedded within the US SMR-300 Reference Plant, such that the design utilises passive operating and safety features to prevent and, if necessary, mitigate the consequences of (US) design basis and beyond design basis accidents. It includes the approach to safety function identification and classification in accordance with US NRC requirements, defence in depth provisions, passive safety features and the approach to grouping and separation within the design. This is supported by:
    - Holtec SMR Top-Level Plant Design Document [32].
    - The Plant-Level Function Identification and Decomposition [33].
    - The classification methodology of the SMR-300 [22].
    - The Design Standard for Grouping and Separation [34].
- Sub-chapters 2.6.3, 2.6.6, 2.6.7 and 2.6.8 provides a summary of the environmental, performance and constructability philosophies that are embedded within the SMR-300 design requirements. This is supported by:
    - Holtec SMR Top-Level Plant Design Document [32].
- Sub-chapter 2.6.4 provides a summary of the Radiation Protection Philosophy and defines the US NRC dose acceptance criteria for the US SMR-300 reference plant design. The US design targets are comparable to other modern reactor designs and are below UK dose limits. This is supported by:
    - Design Standard for Radiation Protection [35].
- Sub-chapter 2.6.5 describes the US safety analysis framework for Deterministic and Probabilistic Safety Analyses, and the safety goals that require demonstration in accordance with US requirements. These analyses will be undertaken to support demonstration of the safety goals in support of the US PSAR that will be prepared in support of the US CPA.
- Sub-chapter 2.6.8 provides a summary of the decommissioning approach for the US SMR-300 reference plant that applies to the SMR-300 and requires designers to

consider decommissioning early in the design phase, incorporate lessons learnt where available from the US NRC, the Nuclear Energy Agency (NEA), International Atomic Energy Agency (IAEA) and the Western European Nuclear Regulators' Association (WENRA). This is supported by:
  - SMR-160 Design Standard for Decommissioning [36].
- Sub-chapter 2.6.9 presents a high-level summary of the codes and standards used within the US SMR-300 reference plant. The US SMR-300 reference plant is designed to be compliant with applicable US regulations with consideration of international regulatory frameworks and recommendations. The codes and standards used in the design of the SMR-300 have been selected to meet the stringent requirements of the US NRC. These mature and established US codes and standards are internationally recognised, and commensurate with the importance of the safety functions being delivered. This is supported by:
  - Step 1 Holtec SMR Codes and Standards Report [37].
- Sub-chapter 2.6.10 provides an overview of the project requirements and standards that originate from the Holtec SMR Top-Level Plant Design Document [32]. SMR-300 project requirements and standards are developed with the intent to comply with NRC requirements. This is supported by the following documents, noting that SMR-160 project level design standards apply to the SMR-300 unless or until superseded:
  - Holtec SMR Top-Level Plant Design Document [32].
  - Classification of SSCs [22].
  - Containment isolation requirements [38].
  - Security and safeguards [39].
  - Cyber security requirements [40].
  - Fire protection [41].
  - External events [42].
  - Human Factors (HF) [43].
  - Grouping and separation [34].
  - Severe accidents [44].
  - Application of single failure criterion [45].
  - Radiation protection [35].
  - Environmental qualification [46].
  - Decommissioning [36].
  - Basic civil structural requirements according to seismic class [47].

Argument 1.1A-2: The Reference US SMR-300 Plant is designed to provide sufficient defence-in-depth and independence between individual levels in line with international good practice.

- Sub-chapter 2.6.2.3 describes the approach to defence in depth provision, and independence between individual levels of defence in depth within the US SMR-300 reference plant. The approach aligns with the IAEA Specific Safety Requirements (SSR) 2/1 [48]. This is supported by:
  - SMR Top-Level Plant Design Document [32].

An initial UK DBAA for the generic SMR-300 is presented within Part B Chapter 14 [17] and further summary of the output of the safety analysis and demonstration that the SMR-300 is designed with adequate defence-in depth, is provided in Part A Chapter 5 [4].

Argument 1.1A-3: The adoption of passive engineered safety measures as a means to mitigate design basis accidents represents relevant good practice, and a logical response to global operational experience examining causal and contributing factors in historical nuclear incidents where failure of active systems and human errors are a common factor. Adoption of these practices will have a net-positive effect on overall risk. Any detrimental risks can be shown to be as low as reasonably practicable.

Sub-chapter 2.6.2.4 describes how IAEA recognise that passive safety systems are desirable method of achieving "*simplification and improved reliability in the of performance in essential safety functions" and* has been the subject of considerable research.

- The SMR-300 design utilises passive operating and safety features to prevent and, if necessary, mitigate the consequences of design basis and beyond design basis accidents. This is supported by:
  - Holtec SMR-300 GDA Passive Systems Report [49].

The US PSAR will demonstrate that the US SMR-300 Reference plant will meet or exceed US NRC General Design Criteria (GDC) and acceptance criteria. An initial UK DBAA for the generic SMR-300 is presented within Part B Chapter 14 [17] and further summary of the output of the safety analysis and demonstration that the adoption of a passive safety philosophy reduces risks to ALARP, is provided in Part A Chapter 5 [4].

Argument 1.1A-4: The initial layout of the nuclear facilities for the SMR-300 reference plant are optimised, to appropriately account for safety.

- Sub-chapter 2.6.11 describes how the SMR-300 Plant Layout is influenced by the top level design requirements, that embody the combined safety, environmental, radiation protection, performance, constructability and decommissioning philosophies, which, in themselves are set to meet US NRC requirements. A description of the management arrangements for plant layout is provided for control and further development of the plant layout through detailed design. This is supported by:
  - Holtec SMR Top-Level Plant Design Document [32].
  - SMR-160 Design Standard for Decommissioning [36].
  - SMR Design Control [50].
  - SMR-160 Design Standard for Human Factors: Maintenance, Inspection and Testing [43].
  - Equipment and Piping Layout Guidelines for Ensuring Radiation Exposures ALARA [51].
  - SMR-300 Design Standard for Radiation Protection [35].
  - SMR-300 Design Standard for Grouping and Separation [34].
  - SMR-300 Specification – Environmental Conditions [52].
  - The Outage Strategy for SMR-300 [53].
  - Part A Chapter 4 [3].

2.6.1 Plant Objectives and Philosophies

The primary objective of the SMR-300 design is as follows.

- To meet the applicable safety, environmental, security and safeguards requirements and goals for advanced light water PWRs with passive safety features.

This primary objective is met by achieving the following technical objectives.

- Redundant and passive ESF.
- Simplified plant design with structures designed to withstand all postulated external events.
- Ability to mitigate design basis accidents with no operator action.
- Ability to cope with an extended loss of all AC power for 72 hours.
- Defence-in-depth approach to beyond design basis accident mitigation.
- Highly reliable active systems to support normal plant operation.
- Secure by design.

The Top Level Holtec SMR design philosophy described in Holtec SMR Top-Level Plant Design Document [32] and summarised in this section provides guiding principles for developing the Reference Design. This Top-Level Philosophy inspires both top-level plant design requirements and lower-tier project specific plant design requirements that are described in Project Design Standards (see sub-chapter 2.6.10).

2.6.2 Safety Philosophy

The design utilises passive operating and safety features to prevent and, if necessary, mitigate the consequences of design basis and beyond design basis accidents. The design of the plant reduces the burden on operators by providing substantial margins to safety limits, allowing increased time to evaluate plant conditions and to decide what, if any, manual operator actions are appropriate.

2.6.2.1 US Safety Function Identification

Sub-chapter 2.7.6.1 describes safety function allocation to the generic SMR-300 for the UK.

A safety function is a specific action that must be accomplished for safety. The Holtec high-level functions comprise safety and non-safety functions, which have been derived from the three basic safety functions identified in Requirement 4 of the IAEA Safety of Nuclear Power Plant Design [48], which are:

- Control of reactivity.
- Removal of heat from the reactor and fuel pool.
- Confinement of radioactive material, shielding against radiation and control of planned radioactive release as well as limitation of accidental releases.

The US expectation defined in 10 CFR 50 [14] Chapter 2 is that:

“Safety-related structures, systems and components” remain functional during and following design basis events to assure:

1. The integrity of the reactor coolant pressure boundary.
2. The capability to shut down the reactor and maintain it in a safe shutdown condition; or,
3. The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to the applicable guideline exposures set forth in paragraph 50.34(a)(1) or 100.11 of this chapter, as applicable.”

Additionally, as per the US NRC's Standard Review Plan Branch Technical Position (BTP) 7-19 [54], the critical safety functions to be managed from the MCR are identified as follows.

- Reactivity control.
- Core heat removal.
- Reactor coolant inventory.
- Containment isolation.
- Containment integrity.

Holtec has distilled these into the high-level functions shown in Table 5. To maintain the physical barriers to the release of radioactivity to the environment, the following functions shall be achieved for all plant states, except where a postulated condition involves the loss of that function:

Table 5: SMR-300 High-Level Functions

Plant Goal	High-Level Function
Ensure Safety	1.1 Control Reactivity
	1.2 Post-Accident Heat Removal
	1.3 Reactor Coolant System Integrity
	1.4 Containment Integrity
Generate Power (i.e., non-safety)	2.1 Heat Generation
	2.2 Primary Heat Transfer
	2.3 Secondary Heat Transfer
	2.4 Energy Conversion
	2.5 Startup
Environmental Protection	Environmental Protection
Security and Safeguards	Security
	Safeguards

The Plant-Level Function and Identification Document [33] presents functional decomposition of plant goals of ensuring safety and generating power, and derives high-level functions to meet these goals with processes to meet these functions. For example, these functions are allocated to reactor SSCs, and the SFP, which are then decomposed to system-level functional requirements, which are presented in the lower-tier documentation (see sub-chapter 2.6.10) in accordance with the instructions for SSCs given in 10 CFR 50 [14] Appendix A General Design Criteria.

It should be noted that US SSCs `important to safety' consist of two subcategories under 10 CFR 50: 'safety-related' and 'non-safety-related'. While safety-related SSCs are defined in paragraph 50.2 of 10 CFR 50 [14] as relating to design basis events, the regulations do not provide an equivalent set of criteria for determining which non-safety-related SSCs are 'important to safety' (i.e. those that do not originate from protection against a design basis event).

2.6.2.2 US Safety Function Classification

Safety significant SSCs are items important to safety within the facility design which provide a safety function. Safety claims can also be made on the operators, which are Human-Based Safety Claims (HBSC).

The US NRC classifies systems and structures based on their functions within a nuclear facility. This functional approach allows for flexibility in design, as long as the intended safety functions are met. As the US has been at the forefront of the development of LWRs technology for over 70 years, it is considered that the regulatory arrangements and requirements for reactor design set out by the NRC represent good practice.

To ensure alignment between safety importance of engineered systems and the associated quality requirements for the design, analysis, manufacture, test, inspection and certification of such systems, the NRC developed a quality classification system in the 1970s to provide licensees with guidance. This was intended to assist licensees in meeting the requirements set out in 10 CFR 50 [14] GDC 1, namely that related to quality standards and records. The resulting Regulatory Guide (1.26) [55] sets out this guidance which comprises four distinct quality (for safety) groups A-D, as applicable to LWRs.

The classification methodology of the SMR-300 [22] aligns and complies with US NRC Regulatory Guide 1.26 [55] and 10 CFR 50 [14] Section 55a ‘Codes and Standards’ subparts (c), (d), and (e). The classification methodology of the SMR-300 addresses these GDC by classifying SSCs in a manner that imparts requirements to ensure safety functions can be reliably performed when needed. The methodology also uses the general classification process described in the US industry consensus standard ANSI/ANS-58.14 [56] and defines classifications appropriate for the SMR-300 design that are similar to other light water pressurised reactors.

Holtec SMR-300 introduces a further classification, the SMR Class, which includes the quality group seismic category, and electrical category in accordance with Regulatory Guide 1.26 [55]. The classifications range in importance from Class A to Class F as shown in Table 6 below.

Table 6: SMR-300 Quality Classes

<table>
<tr><th>SMR Class</th><th>RG 1.26 Quality Group</th><th>Safety Classification</th><th>Seismic Category (C-I, C-II, NS)</th><th>Electrical and I&C Class</th></tr>
<tr><td>A</td><td>A</td><td rowspan="3">Safety-related</td><td rowspan="3">C-I</td><td rowspan="2">N/A</td></tr>
<tr><td>B</td><td>B</td></tr>
<tr><td>C</td><td>C</td><td>Class 1E</td></tr>
<tr><td rowspan="3">D</td><td>D</td><td rowspan="5">Non-safety-related</td><td>C-II</td><td rowspan="2">N/A</td></tr>
<tr><td>D</td><td>NS</td></tr>
<tr><td>N/A</td><td>C-II</td><td>Non-Class 1E</td></tr>
<tr><td>E</td><td>N/A</td><td>Note[9]</td><td>Non-Class 1E</td></tr>
<tr><td>F</td><td>N/A</td><td>NS</td><td>Non-Class 1E</td></tr>
</table>

Generally, safety-related mechanical items are designed to ASME Boiler and Pressure Vessel Code (BPVC) Section III [57] Class 1, 2, or 3 requirements, while non-safety-related mechanical items are designed to ASME BPVC Section VIII and other applicable industry / manufacturers codes / standards requirements.

ANSI/IEEE-603-1991 [58] gives the basic criteria for safety-related electrical and I&C systems and equipment. Electrical and I&C system equipment and components are classified as Class 1E or Non-Class 1E in accordance with definitions stated in the Institute of Electrical and Electronics Engineers (IEEE) 603 Standard. In general, electrical equipment and components that perform safety-related functions are designated as Class 1E and the equipment and components that do not perform any safety-related function are designated as Non-Class 1E. IEEE-603 Standard is endorsed by Regulatory Guide 1.153 [59] as a method acceptable to the USNRC for complying with 10 CFR 50 [14] Appendix A General Design Criteria, 10 CFR 50.49, and 10 CFR 50.55a, with respect to the design, reliability, qualification, and testability of the power, instrumentation, and control portions of safety systems for nuclear power plants.

10 CFR 50 [14], Appendix S, contains the criteria to which the plant design bases demonstrate the capability to function during and after vibratory ground motion associated with the Safe Shutdown Earthquake (SSE) conditions. SSCs that are designated as C-I are designed to withstand the effects of the SSE and remain functional. C-I applies to both functionality and integrity, and C-II applies only to integrity. Non-seismic items are designated as C-II if they are in the proximity of safety-related items and their failure during a SSE could result in loss of function of those safety-related items.

Sub-chapter 2.7.6.2 describes the approach to identify a UK equivalent classification for SSCs and how this has been applied at GDA Step 2.

2.6.2.3 Defence-in-Depth

The design adheres to principles of Defence-in-Depth (DiD) concepts which provide an overall strategy for safety measures and features of nuclear power plants. This ensures that no single

[9] Class E components may have seismic requirements even though they are not categorised as C-I or C-II.

human or equipment failure will lead to harm to the public, and even combinations of failures that are only remotely possible will lead to little or no harm.

The approach to DiD for the SMR-300 is set out in the Holtec SMR Top-Level Plant Design Requirements Document [32]. DiD helps to establish that the three basic safety functions (controlling the reactivity, cooling the fuel, and confining the radioactive material) are preserved, ensuring that radioactive materials do not reach people or the environment.

The approach aligns with the IAEA SSR 2/1 [48] and is structured in five levels. The goal of each level of protection and the essential means of achieving them in existing plants are shown in Table 7.

Table 7: Holtec SMR Defence-in-Depth Philosophy

Level	Goal	Essential Means
Level 1	Prevention of abnormal operation and failures	Conservative design and high quality in construction and operation
Level 2	Control of abnormal operation and detection of failures	Control, limiting and protection systems and other surveillance features
Level 3	Control of accidents within the design basis	ESF and accident procedures
Level 4	Control of severe plant conditions, including prevention of accident progression and mitigation of the consequences of severe accidents	Complimentary measures and accident management
Level 5	Mitigation of radiological consequences of significant releases of radioactive materials	Off-site emergency response

Holtec's approach to achieving DiD considers normal operating systems, passive safety systems, and active non-safety systems for diversity. There will be sufficient functional diversity and redundancy within the safety systems, and they will be available for all modes of operation.

Demonstration that each level of DiD for the SMR-300 is sufficiently reliable and robust is delivered by the totality of the SSEC chapters and is summarised in Part A Chapter 5 [4]. The following sections provide a summary of the key principles which have been accommodated within the SMR-300 design to provide resilience and DiD.

2.6.2.3.1 Level 1: Prevention of Abnormal Operation and System Failures

The foundation of the SMR-300 DiD design strategy is the prevention of initiating events through robust design, high-quality construction, and conservative operational limits.

The fuel cladding, reactor coolant pressure boundary, and the containment structure form multiple physical barriers to fission product release. In addition, the underground siting of the reactor vessel and containment provides further shielding and isolation from the environment.

High component integrity is ensured using high quality code compliant materials, precision manufacturing, and rigorous quality assurance during construction. The SMR-300 is designed with fewer active components (as compared to conventional PWRs), thereby reducing maintenance needs and the risk of failure of equipment during operation. Human Factors

Engineering (HFE) plays a central role, with a high degree of automation and diagnostics, minimising operator burden and the potential for human error.

The reactors' inherent safety characteristics, such as a strong negative temperature and void coefficients, naturally counteract power excursions and ensure self-limiting responses to disturbances. These features significantly reduce the likelihood of abnormal operations or system failures occurring in the first place.

2.6.2.3.2 Level 2: Control of Abnormal Operation and Detection of Failures

If deviations from normal operation occur, the SMR-300 uses I&C control systems and components to detect and respond promptly. The Plant Control System (PCS) provides control and coordination of the plant during startup, ascent to power, power operation, and shutdown conditions and maintains the plant operating conditions within the prescribed limits. The PCS improves plant safety by minimizing the frequency of events during which a protective response is initiated, and it relieves the operator from routine tasks

In addition, regular testing, diagnostics, and surveillance ensure timely identification and correction of faults before they escalate.

2.6.2.3.3 Level 3: Control of Design Basis Accidents

To handle DBAs, such as LOCAs, loss-of-power, or steam line breaks, the SMR-300 utilises passive safety systems that operate without external power or operator action. The reactors simplified primary system and integral vessel layout also minimize the potential for pipe rupture and loss-of-coolant accidents. If a break were to occur, the containment system and isolation valves would immediately respond, preventing progression.

Should a DBA occur, the containment of the SMR-300 is designed to remain intact and sealed during all postulated events, and to reject its internal energy to the water in the AR. The volume of coolant required to achieve a safe-stable state is contained entirely within containment, reducing the reliance on external supplies and minimising the safety importance of SSCs outside of containment.

Shutdown and long-term reactivity control are achieved through gravity driven insertion of control rod assemblies and diverse shutdown of borated water injection using non-active means. Decay heat removal uses diverse passive recirculation of coolant within the primary and secondary circuits to exchange heat into the large bodies of water in the Passive Core Makeup Water Tank (PCMWT) and AR respectively. RCS depressurisation and safety injection are delivered through reliable, multi-stage depressurisation valves ensuring the DBA progression is controlled.

Initiation of these ESFs is through the PSS, backed up by the DAS. The PSS and DAS are capable of automatic actuation of safety functions without human intervention, improving speed and reliability of response (as compared to operator manual actuations). The PSS and DAS are built on redundant, diverse, and fail-safe design principles, including multiple, independently powered sensor channels that monitor key variables such as reactor power, pressure, temperature, and neutron flux.

Electrical supplies to the ESFs are powered by separate divisions of DC electricity, so failure of supplies from one division does not prevent successful operation

The passive ESFs are described in further detail in Part B Chapter 1 Reactor Coolant System and Engineered Safety Features [60]. These features provide high robustness and reliability during DBAs.

2.6.2.3.4 Level 4: Control of Severe Plant Conditions (Beyond-Design-Basis Accidents)

The SMR-300 is engineered to cope with extreme accident scenarios beyond the design basis, including core melt accidents and extended station blackout.[10]

The philosophy for the management of severe accidents has been developed based on the following approach:

a) Understanding Severe Accident Phenomena.
b) Protection against Beyond Design Basis Events.
c) Defining the Safety Features provided to manage Severe Accidents.
d) Defining the Emergency Operating Procedures (EOP) and Severe Accident Management Guidelines (SAMG).

In the extremely unlikely scenario of a severe accident where core melt is imminent or occurring, the SMR-300 safety strategy shifts to manage the accident's progression and mitigate its consequences. In Beyond Design Basis Events (BDBE) involving core melt, the SMR-300 no longer relies solely on PDH, SDH and the ADS, particularly if those systems have failed, are unavailable, or the accident has progressed beyond their effective window. In these cases, alternative measures are incorporated to achieve reactor depressurisation and facilitate core cooling, primarily as part of SAMGs. These include, but are not limited to, using diverse safety and non-safety related systems like CVC letdown, RPV head vent valves, PDH vents, etc.

The list of available accident management strategies and means to implement them will be identified and reasonable assurance that the equipment will survive to perform its function within the severe environment will be provided during future development of SAMG framework.

Design measures also focus on preventing fuel degradation and reducing accident progression likelihood, ensuring that even in the most extreme cases, the consequences are mitigated effectively and safely.

2.6.2.3.5 Level 5: Mitigation of Radiological Consequences of Significant Releases

Even under the unlikely condition of a major release, the SMR-300 incorporates several robust systems and procedures to limit public exposure.

The plant is equipped with real-time radiation monitoring systems and site-wide communication systems to support emergency planning and public notification. Emergency

[10] Note that Loss of Offsite Power (LOOP) and Extended Loss of Grid (ELOG) are considered as design basis frequent faults for the generic SMR-300 (see sub-chapter 2.8.2.5).

response procedures are planned to be integrated with regional and national frameworks, ensuring coordinated evacuation or sheltering, although the SMR-300's low source term and strong containment make such actions extremely unlikely.

Finally, extensive deterministic safety analyses (fault studies, severe accident studies and hazards studies) and PSA support the design, confirming that the frequency and consequences of significant radiological releases are well below international safety limits.

2.6.2.4 Passive Safety Features

The SMR-300 design provides multiple physical barriers to the release of radioactivity to the environment. These include the fuel cladding, the reactor coolant pressure boundary, and the containment structure. It is the intent that during NOs and AOOs the SMR-300 design prevents challenges to the integrity of all these barriers and the design incorporates highly reliable passive ESF. The Holtec SMR-300 GDA Passive Systems Report [49] describes the passive safety features incorporated in the SMR-300 design. These passive safety features are designed to reduce the reliance on operator actions for a period of 72 hours post fault initiation.

The IAEA, as far back as 1991 [61] has recognised that passive safety systems are desirable method of achieving "*simplification and improved reliability in the of performance in essential safety functions" and* has been the subject of considerable research since (within the IAEA and without). Passive engineered systems in this context can be coarsely characterised as utilising one or more of the following traits [62]:

- Reduced reliance on active components for proper actuation.
- Reliance on natural phenomena for proper operation.
- Not requiring support functions for proper operation.
- Not requiring human intervention for actuation and operation.

Nearly all historical nuclear incidents feature HF or dependence of active systems as either a causal or contributing factors. This is further reinforced by the Fukushima Daiichi accident where in maintaining active cooling proved challenging post event.

Furthermore, the IAEA do recognise that that typical features of SMRs such as relatively small core sizes and power lend themselves favourably for the application of passive systems based on natural phenomena (e.g. natural circulation). Which implies that it is good practice to consider the use of passive systems where possible.

Passive systems (like any other system) do come with some aspects that require careful consideration. For instance, reliance, on natural phenomena to provide heat removal, will require that the phenomena is well understood and can be shown to function is as intended in the appropriate conditions. This is what the "Integrated and Separate Effects Testing" (ISET) program seeks to do. It is intended that once residual risks of implementing passive systems are managed, that the net effect on overall risk will be shown to be positive.

The containment of the SMR-300 is designed to remain intact and sealed during all postulated events, and to reject its internal energy to the large body of water surrounding the containment structure. After a postulated event, such as a LOCA, the plant is designed to automatically achieve and maintain a safe shutdown state without operator actions, without external water, without external power, and without active systems. Figure 8 shows the accident progression following a LOCA.

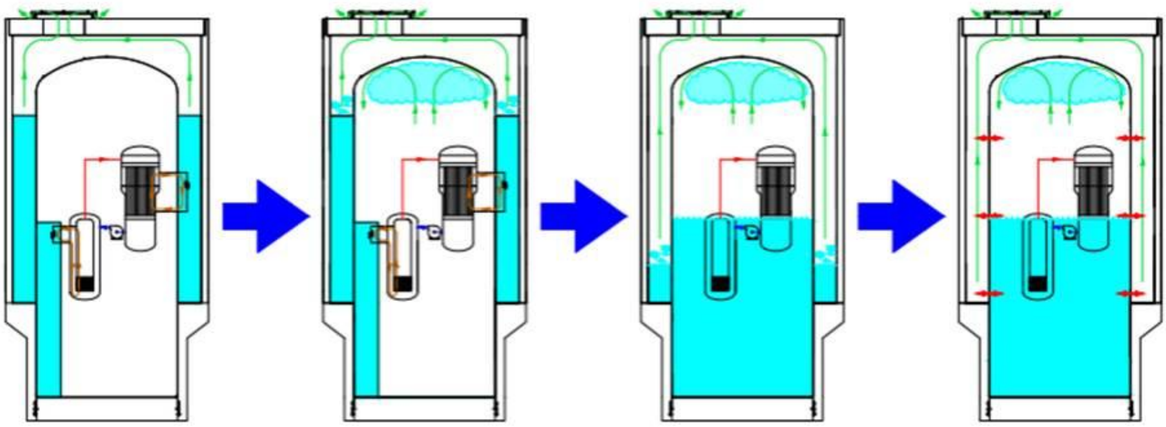

Figure 8: LOCA Accident Progression

The passive ESF are described in further detail in Part B Chapter 1 Reactor Coolant System and Engineered Safety Features [60].

2.6.2.5 Grouping and Separation

Separation is included by design to assure protection for certain credible hazards in the NPP, such as:

- Pipe Breaks.
- Missiles.
- Fires.

Each hazard can have different effects. For example, a pipe break in a high energy fluid system can result in pipe whip, jet impingement, excessive sub compartment pressure and temperature and internal flooding.

The following are general methods of separation:

- **Plant Arrangement (Layout):** A basic approach to segregation and separation in the initial plant layout is to arrange redundant divisions of the safety systems so that they do not share a common area. Redundant safety-related equipment may be placed in different compartments or even on different elevations. The Plant Layout design approach is described in sub-chapter 2.6.11.
- **Barriers:** Where Plant Arrangement cannot be employed to separate the redundant equipment, a protective barrier may be installed.
- **Spatial Separation:** Where both Plant Arrangement and Barriers are not practical, it may be sufficient to separate the redundant trains of the safety systems by distance.

Grouping and separation criteria have been established for the SMR-300. The Design Standard for Grouping and Separation [34] identifies these design criteria and requirements, as well as the project approach to addressing grouping and separation in the design process. It should be noted that [34] is undergoing an update to the SMR-300 design.

The identification of, and approach to assessment of, Internal Hazards for the generic SMR-300 is described within Part B Chapter 22 Internal Hazards [27]. Part B Chapter 14 Design Basis Analysis (Fault Studies) [17] will present the outcome of the Internal Hazards assessment, as recorded in the Preliminary Fault Schedule (PFS) post GDA.

2.6.3 Environmental Philosophy

To ensure impacts to the environment from the operation of the NPP are minimised, the design includes all reasonably practical measures to protect the environment during normal operation and to mitigate consequences from an accident. The design will contain provisions to control, treat, and monitor releases to the environment and strive to prevent and / or minimise the generation of radioactive and hazardous waste. Where the waste generation is unavoidable, the SMR-300 will prevent and / or minimise the risks and impacts of radioactive wastes on workers, the public and the environment. Environmental claims and arguments are presented in PER Chapter 6 Demonstration of BAT [63].

2.6.4 Radiation Protection Philosophy and US Dose Acceptance Criteria

The design approach is to minimise radiation exposure to plant personnel and the public with due consideration for human factors. The ALARA principle is applied throughout the design of the plant to minimise radiation doses and the release of radioactive materials into the environment.

To reduce doses to OSW and MoP to levels that are ALARA and pursue the best practice, design dose targets (constraints in IRR17 [64] parlance) for NOs have been established. These are derived for the SMR-300 in Design Standard for Radiation Protection [35]. UK dose acceptance criteria in accordance with IRR17 [64] are described in sub-chapter 2.7.4 and discussed further in Part B Chapter 10 [8] together with a comparison of UK versus US limits and targets.

A radiation zoning system shall be established to classify non-RCA and RCAs according to anticipated personnel occupancy and access restrictions in all areas of the station during normal conditions. The zoning system shall be used to implement radiation protection controls and to direct the movements of personnel or equipment and thereby to control personnel exposure in the station.

2.6.5 Safety Analysis Philosophy and Safety Goals

To ensure that the nuclear safety and DiD objectives are met and the means of addressing them are adequate, supporting analyses will be performed for event sequences that occur during all plant states.

A deterministic safety analysis framework supplemented by probabilistic safety assessments has been developed to quantify and justify these levels of safety protection. Accident prevention is the first level of protection to minimise the frequency and severity of initiating events which could challenge safety. Structures, systems, and components are designed with features and functions to provide high confidence that initiating events which do occur do not progress to the point of core damage. Lastly, quantitative accident mitigation requirements are developed and justified to meet or exceed NRC safety goals.

The SMR-300 is designed to mitigate the consequences of design basis and severe accidents. During NO and AOO, the design philosophy is to prevent challenges to the integrity of all physical barriers and, if challenged, failure shall be prevented. Additionally, SSCs are robustly designed to provide increased protection from both internal and external events.

Safety assessments are incorporated into design activities via:

- Limited set of bounding events for safety analysis considered to be 'driving' for design, and preliminary results are extensively used to inform design.
- Evaluation of similar designs and industry Operating Experience (OPEX) (e.g., Electric Power Research Institute (EPRI) Utility Requirements Document (URD)) used to inform design and changes.
- Prior Research and Development (R&D) considered as design input and future R&D is determined, in part, by safety assessment needs.
- Proven practices employed, such as industry consensus standards, to ensure a high margin of safety in design, including safety analysis methodologies consistent with Relevant Good Practice (RGP).
- Design analyses to establish Operating Limits and Conditions (OLC), also referred to as Limiting Conditions for Operation. These OLCs will be based on conservative design assumptions and will show that operation of the SMR-300 meets the requirements of all safety codes and standards. The design will minimise the unavailability of safety systems (ESF) for maintenance and testing and consider their unavailability, if applicable, during the analysis. This analysis will define the operating restrictions during normal operation.

2.6.5.1 Safety Goals

Safety goals are established for the effective implementation of the general nuclear safety objective and for demonstrating that the operation of the NPP does not pose any significant additional risk to the public health, safety, security, and to the environment in comparison with other risks to which the public and the environment are normally exposed. These safety goals are for the standard US SMR-300 plant design and comply with US requirements. UK goals or targets for the generic SMR-300 (i.e., ONR Safety Assessment Principles (SAP) [65] targets 1-9) are described in sub-chapter 2.7.4.

2.6.5.1.1 Qualitative Safety Goals

A limit is placed on the societal risks posed by NPP operation as derived by the NRC qualitative safety goals in NUREG-0880 [66].

The following two qualitative goals shall be met by the SMR-300 design:

1. Individual members of the public are provided a level of protection from the consequences of NPP operation such that there is no significant additional risk to the life and health of individuals.

2. Societal risks to life and health from NPP operation are comparable to or less than the risks of generating electricity by viable competing technologies and should not significantly add to other societal risks.

2.6.5.1.2 Quantitative Safety Goals

To demonstrate and achieve the intent of the above qualitative goals, the following quantitative safety goals from Regulatory Guide 1.174 [67] shall be met by the SMR-300 design.

- **Core Damage Frequency:** The sum of frequencies of all event sequences that can lead to core damage is less than 10^{-5} per reactor year. Core damage is defined in Holtec SMR Top-Level Plant Design Requirements [32] as local fuel cladding temperature above 2200 degrees F [1204 degrees C]. This definition of core damage is conservative as at this temperature the fuel may suffer localised damage; however, it will remain intact. By combining this conservative definition for core damage with a conservative frequency limit of 10^{-5} per reactor year, the intent of the qualitative safety goals is met under all regulatory frameworks considered.
- **Small Release Frequency (SRF):** The sum of frequencies of all event sequences that can lead to a release to the environment of more than 10^{15} becquerel of iodine-131 is less than 10^{-5} per reactor year.
- **Large Release Frequency:** The sum of frequencies of all event sequences that can lead to a release to the environment of more than 10^{14} becquerel of caesium-137 is less than 10^{-6} per reactor year.

2.6.5.2 Deterministic Safety Analysis

Deterministic safety analysis will be performed in support of the US PSAR to show that the operation of the SMR-300 does not pose an unacceptable risk or consequences to the workers at the plant, the public or the environment for all AOOs and DBAs.

Part B Chapter 14 Design Basis Analysis (Fault Studies) [17] presents further details of the US deterministic approach and the UK DBAA strategy.

2.6.5.3 Probabilistic Safety Assessment

PSA will be performed using best estimate assumptions and supports the demonstration of Regulatory Guide 1.174 [67] quantitative safety goals.

The Level 1 PSA identifies postulated initiating events by conducting a systematic review of the plant design and evaluates the plant response following these initiating events. It consists of the identification and quantification of accident sequences that could lead to core damage and gives insights into the performance of the safety systems provided to prevent core damage. The Level 1 PSA evaluates the summed core damage frequency and demonstrates the compliance with the safety goal for core damage frequency.

The Level 1 PSA will provide feedback for modifications and refinements to the design, input to technical specifications, outage planning, emergency operating procedures and to the risk-importance decision making process by identifying the systems that contribute the most to total core damage frequency.

The Level 2 PSA extends analyses of reactor core damage performed by the Level 1 PSA to quantify releases of radioactivity into environment. The Level 2 PSA models the plant design provisions that can mitigate the consequences of core damage and provides an estimate of the frequency and magnitude of a release of radioactive material to the environment. The Level 2 PSA determines the small release frequency and large release frequency and demonstrates the compliance with the safety goals for small and large release frequency.

The Level 2 PSA will provide feedback for modifications and refinements of complementary design features, input to severe accident management guidelines, emergency preparedness procedures, off-site emergency procedures and to the risk-important decision making process by identifying the systems that contribute the most to total large release frequency.

The Level 3 PSA is used to evaluate the off-site radiological risks of nuclear power plants, ensuring they meet safety regulations and environmental protection standards and to demonstrate that a new nuclear power plant design is suitable for construction.

Part B Chapter 16 [68] describes the SMR-300 PSA supporting GDA.

2.6.6 Performance Philosophy

Holtec SMR's performance aims to ensure the plant is competitive with other commercial power generation technologies, whilst not compromising the safety philosophy.

Design inputs consider current and projected utility grid needs for the entire plant design life. Plant design life is backed by economic assessments and reflect what is practically achievable in the specification and selection of components and materials. The design intent includes significant margin to ensure:

- There is sufficient designed-in capability and capacity to accommodate operating transients without causing initiation of engineered safety features.
- Operators have significant time to assess and deal with upset conditions with minimal potential for damage to plant equipment.
- System and component reliability minimizes the potential for exceeding operating limits, derating output, or unplanned shutdown.
- There is sufficient design assurance that plant design life can be met through a combination of design-for-life components, maintenance strategies, and planned replacements.

Plant availability is improved compared to existing plants by achieving a higher capacity factor, preventing losses in generating capacity due to aging, designing for adequate cycle length, and reducing the duration of planned outages. The design life objective for the plant is to achieve a design life of at least 80 years.

Where practicable, plant design employs modern digital technology for monitoring, control, and protection functions. Modern digital communications are used where it is appropriate to reduce the cost and complexity of cable runs throughout the plant.

A Man-Machine Interface System (M-MIS) promotes error-free NO and quick, accurate diagnosis of off-normal conditions. Control room operations are improved using proven, highly integrated digital control systems with advanced Human System Interfaces (HSI).

The design philosophy is to minimise staffing levels. Where minimum staffing is required by regulation, sensible alternatives are proposed and justified consistent with best available technology and technique. The role of the operator is that of an intelligent overseer in the event of off-normal conditions. The plant is designed to allow the operator significant time to evaluate the plant condition and decide what, if any, manual action is needed. Plant operation is highly automated, but the plant is not designed to lock out the operator at any time.

Design life is incorporated into the planning of the preventive maintenance and inspection programme such that inspection, test, and maintenance requirements are simplified or eliminated.

Due consideration is provided for operational experience from existing plants. Operating procedures and system designs are informed by operational experience and make use of computer-based procedures.

2.6.7 Constructability Philosophy

The Holtec SMR-300 emphasises simplicity in plant design and construction for improved safety and economy. The plant design considers replacement of design features and equipment of existing LWR plant designs with both simplified and passive safety systems. Design includes a minimum number of systems and equipment, consistent with essential functional requirements.

Specifically, material, system, and operation simplification efforts include:

- A minimum number of types and grades of materials, where practical, consistent with service conditions and design performance requirements.
- A minimum amount of instrumentation, control functions and control loops, consistent with the essential functional requirements of the systems, availability, maintainability, and testing capability. The number of divisions and channels for electrical, instrumentation, and control systems should reflect the minimum required to achieve licensing and other plant safety objectives while reducing maintenance requirements.
- A minimum number of valves, pumps, heat exchangers, snubbers, and other mechanical components, consistent with the essential functional requirements of the systems, availability, maintainability, and testing capability.
- Simplified operations during all modes of operation, including operator actions to diagnose and manage abnormal and accident conditions.

The Holtec SMR-300 design uses standardised component sizes, types, and installation details to improve construction, maintenance, and operations. “Off-the-shelf” components as opposed to “special-order / design” components are utilised to the maximum extent practical.

Proven technology from the experience base of existing LWRs has been heavily considered in plant design and component selection to minimise risk to the plant owner and assure credibility and control of schedule and cost. The plant is designed using SSC’s proven through several years of acceptable service in LWR plants. Materials used have been selected from materials proven in service in LWR plants. Laydown space, for maintenance and staging of equipment, has been incorporated into the plant design, with means for removal and replacement of components.

Prefabrication, preassembly, and modularisation is used to the maximum extent practical to reduce capital cost and construction time. Provisions for simplification and facilitation of construction and startup include provisions of good crane and material handling access, adequate space and access for construction activities, and provision for temporary construction buildings and equipment.

In addition to construction considerations, the Holtec SMR design considers full design life of the plant and the future decommissioning and dismantling activities. The full design life of the plant has been used in planning for on-site interim spent fuel storage in a dry fuel storage installation.

2.6.8 Decommissioning Philosophy

The SMR-160 Design Standard for Decommissioning [36] (this project level design standard applies to the SMR-300 unless or until superseded) specifies that it is a US regulatory requirement that:

- Materials and processes are to be chosen to minimise the generation of radioactive waste.
- Plant layouts are designed to limit the spread of contamination and to facilitate access for decommissioning.
- Interim and final waste storage facilities must be given consideration.

The Design Standard requires designers to consider decommissioning early in the design phase, incorporate lessons learnt where available from the US NRC, the NEA, IAEA and the WENRA, and to incorporate design features that:

- Prevent the spread of radioactive material during normal operation and decommissioning.
- Provide for the containment of spilled or leaked radioactive material to prevent the spread of contamination.
- Enhance access to contaminated material or equipment to facilitate its removal.
- Enhance structural decontamination through improved surface preparation.
- Minimise the quantity of radioactive waste generated during decommissioning efforts.
- Ensure that radiation exposure of both decommissioning personnel and the public is ALARA.
- Provide protection of the health and safety of the public, and the environment.
- Provide storage facilities, surveillance monitors, and site preparation.

The Design Standard also specifies the following design features to be applied by the designer:

- Reduction of the radiation source (including selection of materials in and around the reactor core that will minimise the radionuclide inventory at the end of the operating life of the NPP).
- Plant layouts that limit the spread of contamination (including grouping equipment to enable segregation of higher and lower radiation areas, airflow away from areas of lower radioactivity to higher, etc.).
- Plant layouts that facilitate dismantling and decontamination of radioactive equipment (for example, design and placement of pipes, ducts, sumps, and drains, minimisation

of areas that could be exposed to contamination, avoidance of crud traps, ventilation systems, and radiological zoning, etc.).
- Simplification of waste management systems (including waste process design features and space requirements).
- Design for Storage with Surveillance or deferred decommissioning (including considerations for corrosion allowance, and systems that may be required during different stages of decommissioning).
- Radiological data and design information management to facilitate decommissioning (including a documentation programme to collect and update drawings, design specifications, and any modifications).

The SMR-300 design considers the full design life of the plant in planning for on-site interim spent fuel storage in a dry fuel storage installation to facilitate decommissioning.

Part B Chapter 26 Decommissioning Approach [69] presents a decommissioning strategy assessment lays out the groundwork for future evidence to be developed for the decommissioning approach topic. Supported by the qualitative decommissioning inventory, Nuclear Waste Services (NWS) expert view and Integrated Waste Strategy (IWS). It currently shows how the design and strategy of the decommissioning topic is aligned with UK expectations and demonstrate that all risks associated with decommissioning can be reduced ALARP.

2.6.9 Fundamental Regulatory Requirements

The SMR-300 reference plant is designed to be compliant with applicable US regulations with consideration of international regulatory frameworks and recommendations. Protection of the public health and safety, the environment, and that of plant workers is paramount. These regulations inform the project requirements and standards.

2.6.9.1 US Nuclear Regulations

The codes and standards used in the design of the Holtec SMR-300 have been selected to meet the stringent requirements of the US NRC. These mature and established US codes and standards are internationally recognised, and commensurate with the importance of the safety functions being delivered. The codes and standards applied to the design of nuclear safety related SSCs of the SMR-300 are generally nuclear specific. Many of them represent good practice adopted on UK nuclear licensed sites and / or application in earlier successful GDAs.

Sub-chapter 2.7 introduces the UK approach to safety demonstration for the generic SMR-300.

2.6.9.2 International Requirements and Guides

The international requirements represent another level of requirements that the SMR-300 project considers. The international requirements are used as source of RGP where the requirements can be demonstrated within the Canada Nuclear Safety and Control Act: Nuclear Non-proliferation [70].

2.6.9.2.1 IAEA and other International Requirements

The SMR-300 is designed to be licensable in the US. The US is a United Nations (UN) member state and signatory to the convention on nuclear safety.

The US NRC oversees and facilitates application of IAEA safeguards at NRC licensed facilities.

Discussion of the fundamental regulatory requirements applicable for the generic SMR-300 design in the UK is provided in sub-chapter 2.7.

2.6.9.3 Selection of Codes and Standards

The codes and standards used in the design of the SMR-300 have been selected to meet the requirements of the US NRC and CFRs, specifically title 10 CFR Part 50 [14]. They are mature, established and internationally recognised US and international codes and standards, developed by standardisation organisations such as:

- The American Society of Mechanical Engineers (ASME).
- The American National Standards Institute (ANSI).
- The American Nuclear Society (ANS).
- The American Concrete Institute (ACI).
- The American Institute of Steel Construction (AISC).
- The American Society of Civil Engineers (ASCE).
- The International Electrochemical Commission (IEC).
- The Institute of Electrical and Electronics Engineers (IEEE).
- The International Society of Automation (ISA).
- The Electric Power Research Institute (EPRI).
- The American Society for Testing and Materials (ASTM).
- The National Fire Protection Association (NFPA).
- The American Society for Non-destructive Testing (ASNT).
- The International Organisation for Standardisation (ISO).

Within the US, the NRC specifies the design codes and standards that must be used, although exceptions are permitted subject to justification. Endorsed codes, together with the versions of the codes (which may not be the latest versions) are promulgated through NUREG-0800 [15]. This requires NRC to have substantial engagement with code committees and the development of individual codes.

An advantage to the US industry is that these codes are automatically well matched to the NRC regulatory approach, and in many cases, written specifically to respond to NRC regulatory concerns. A disadvantage to the UK industry, and to US designs moving into the UK, is that these US codes do not fully necessarily reflect UK nuclear practice and the use of the risk ALARP principle. Nevertheless, US codes are used extensively around the world, including in the UK.

The codes and standards applied to the design of nuclear safety related SSCs are generally nuclear-specific. Many of them represent good practice adopted on UK nuclear licensed sites and / or application in earlier successful GDAs. This is from existing practices adopted on UK nuclear licensed sites, application in earlier and successful GDAs, as well as recognition as RGP by ONR Technical Assessment Guides (TAG).

The principal codes and standards of the SMR-300 are summarised in Table 8 below. The detailed list of the codes and standards (including the versions) applied to the SMR-300 is provided in the Step 1 Holtec SMR Codes and Standards Report [37] and their appropriateness justified in relevant PSR Part B Chapters.

Table 8: SMR-300 Principal Codes and Standards

Technical Area	SMR-300 Codes and Standards
Mechanical Engineering	ASME BPVC & ASME Standards Heat Exchange Institute Standards ANSI/ANS Standards
Structural Integrity	ASME BPVC
Civil Engineering	ACI 349 & ACI Standards ANSI/AISC N690 ANSI/ANS Standards ASCE 4
Electrical & Instrumentation and Controls	IEC Standards IEEE Standards ANSI Standards ANSI/ANS Standards ANSI/ISA Standards
Probabilistic Safety Analysis	ASME/ANS Standards EPRI Standards US NRC NUREG
Fire Protection	ASTM Standards NFPA Standards
Quality Assurance	ANSI/ANS Standards ASME NQA-1 ASNT Recommended Practice ISO 9001 Quality Management Systems

Appendix A of 10 CFR 50 [14] presents GDC that are the minimum requirements for the principal design criteria used in water-cooled nuclear power plants, similar in design and location to plants for which construction permits have been issued by the US NRC. GDC 1, 2, 3, and 4 specify that SSCs important to safety are constructed in a manner that will assure safety functions can be performed reliably for design basis conditions, such as environmental effects, fire, natural phenomena, and interactions to other systems which may impact their functions.

The selection of codes and standards applied for the development and design of the SMR-300 is commensurate with the importance of relevant safety functions delivered. SSCs of the SMR-300 are classified according to their importance to safety. The selection of codes and standards is imparted from the classification of SSCs based on US NRC requirements. The classification assigned determines requirements for design and construction of the SSCs, including compliance to codes or standards. See sub-chapter 2.7.6.2 for a description of the equivalent categorisation and classification approach to be applied to the generic SMR-300 for the UK.

2.6.10 Project Requirements and Standards

The hierarchical structure of the design documentation is defined within the SMR-300 Design Control Procedure [50]. Figure 9 depicts the hierarchy of SMR-300 project requirements and standards, and is described in more detail in Part A Chapter 4 [3].

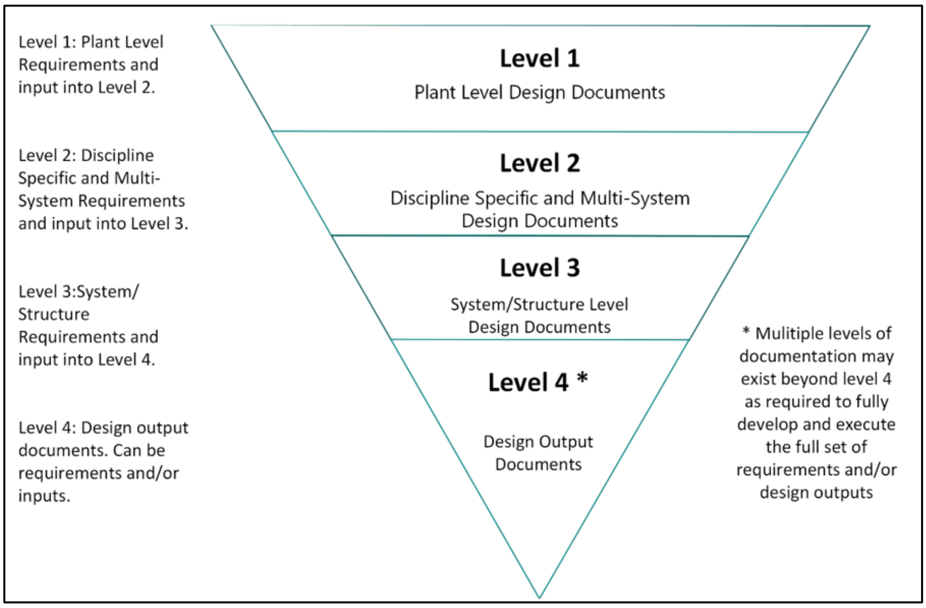

Figure 9: Hierarchy of Holtec SMR Design Documentation

SMR-300 project requirements and standards are developed with the intent to comply with the regulatory requirements described in the previous sections. The project requirements and standards are developed in a hierarchal manner, with lower tier documents intended to convey specific instructions or interpretation for higher tier documents.

2.6.10.1 Top-Level Plant Design Document

This document serves as the design philosophy and high-level requirements for all Holtec SMR designs. This document comprises the first tier of the Holtec SMR design document architecture. Content for this document was primarily adopted from Holtec's interpretation of tier 1 and tier 2 chapter 1 of the EPRI URD [71]. EPRI URD guidance encompasses both industry experience and current US regulations to present a clear, complete statement of requirements for the next generation of nuclear plants. More detailed requirements covered in follow-on sections of the EPRI URD are captured in specifications and lower-level design documents.

The design philosophy included in the Top-Level Plant Design Requirements [32] serves as the guiding principles for the intended design of the Holtec SMR plant. These guiding principles govern the approach to plant design and serve as inspiration to design requirements. The top-level design requirements are specific actionable statements that shall be met to ensure the philosophy is achieved.

This document does not state all the requirements for licensing compliance. However, Holtec recognises that there are differences between the UK and US regulatory frameworks, as outlined in the UK / US Regulatory Framework & Principles Report [72]. Sub-chapter 2.7.4

describes a review of the alignment of the respective UK and US regulatory expectations within the SAPs / SyAPs / ONMACS that are relevant to the SMR-300 design.

2.6.10.2 Specification Documents

Specification documents comprise the second and third tier of the Holtec SMR document architecture. Discipline-specific specifications, at Level 2, comprise design intent, requirements, and basis for each discipline (i.e., Mechanical, Civil, Electrical). General specifications comprise the design intent, requirements, and basis for multi-disciplinary topics (i.e., environmental protection, equipment, fire protection).

2.6.10.3 Project Design Standards

Project Design Standards are lower tier documents developed for each specific SMR-300 topical discipline and specify the design requirements, engineering methodologies and practices for use by designers to ensure consistency and standardisation across the design. The Design Standards are used by each technical discipline, as described in SMR-300 Design Control Procedure [50] . The Design Standard shall provide a descriptive representation of the discipline's scope of work, scheduling information, major deliverables, including Milestones, and the workflow entailing sequences, key interfaces, computer programs used for drawing, modelling and/or analysis, applicable standards and procedures (non-exhaustive), and assigned responsibilities

Design Standards for the following topical disciplines are applied in the development of the SMR-300 design (SMR-160 project level design standards apply to the SMR-300 unless or until superseded):

- Classification of SSCs [22].
- Containment isolation requirements [38].
- Security and safeguards [39].
- Cyber security requirements [40].
- Fire protection [41].
- External events [42].
- Human Factors [43].
- Grouping and separation [34].
- Severe accidents [44].
- Application of single failure criterion [45].
- Radiation protection [35].
- Environmental qualification [46].
- Decommissioning [36].
- Basic civil structural requirements according to seismic class [47].

2.6.11 Plant Layout

Sub-chapter 2.3.1 presents a high-level description of the SMR-300 Site Layout and Main Buildings description. This sub-chapter describes how the key requirements have influenced the high-level architecture of the SMR-300 plant layout, identifying some key examples, and the processes and design considerations that will influence the continuous development of the plant layout beyond GDA, through detailed design.

The initial approach to layout, is instructed primarily by the design philosophy (*Safety, Performance, Constructability)* high-level requirements and the objectives of SMR-300 project. These are set out in the Holtec SMR Top-Level Plant Design Requirements [32], which presents the Design Philosophy, along with mostly EPRI URD derived plant requirements. Generally, these layout plant requirements are captured initially in the civil design of structures and buildings. Using the high-level information from the Requirements Paper, these are laid out as design specification documents, which inform the preparation of the building General Arrangement (GA) drawings where equipment is located, and rooms are sized accordingly.

A key plant requirement that establishes the capability of the SMR-300 to respond to postulated design basis events, beyond design basis events, and to other hazards is safety requirement #1001 [3.1.1][11].

These requirements place emphasis on the primacy of the design of the NSSS and the PCC systems. The layout therefore prioritises these systems. The NSSS and PCC are shaped by their thermo-hydrodynamics, a symmetrical design of hot and cold legs for consistent flow performance in the RCS, Examination, Inspection, Maintenance and Testing (EIMT) concerns, and the requirement for a large inventory of water for flooding the designated containment volume. It must also be located above the volume to be flooded to enable gravity to drive the flooding mechanism. The PCMWT is required to reach a substantial elevation to provide the water volume above the primary decay heat removal system and the driving head for depressurised core make-up capability via PCM.

The AR is required for achieving an Ultimate Heat Sink (UHS) that meets the passive safety requirements. Penetrations through this volume must be minimised, which has resulted in the RCS and its support systems being located below grade so they can be routed under the UHS. Other requirements lead to a design that locates the core below grade, as this is a SMR-300 design objective (requirement #1033 [3.1.27]) for increased security and protection from external threats.

Another feature of the passive cooling requirement is for the below grade layout design to contain the “floodable volume” from LOCAs, act as the evaporative basin for flood-up events, and provide the hydraulic head above the core for passive Direct Vessel Injection (DVI) injection from the makeup tank. All the large primary fluid systems (RCS, DVI, RHR, Spent Fuel Cooling (SFC) etc.), except for the Accumulator injection tanks, are located within this internal flood-proof basin.

Additionally, to best achieve the Plant design requirement #1040 [3.1.33][12], the spent fuel pool was included within the containment volume, otherwise a separate passively cooled fuel building would have been required, which would fall out of the design philosophy outlined in

[11] The plant design shall rely on passive means to mitigate design basis accidents. Passive means shall only require a one-time actuation of valves to place in service. Passive means shall be consistent with EPRI’s definition of natural forces such as gravity and natural circulation, stored energy such as batteries and compressed fluids, check valves, and non-cycling powered valves.

[12] The prevention of fuel damage for spent fuel contained within the Spent Fuel Pool cooling for postulated accidents shall be via passive means, such as evaporative cooling and gravity fed make-up.

the Holtec SMR Top-Level Plant Design Requirements [32] (requirement #1115 [3.2.29])[13]. Additionally, the gravity fed make-up is combined with the passive cooling system in a singular reactor cavity / spent fuel pool volume as the condensate guttering sump that completes the long-term passive cooling cycle.

Design life requirements also influence the containment layout, requirement #1052 [3.2.1], calls for all major plant equipment with a planned replacement capability be designed to accommodate that replacement. For the steam generator for example, this is achieved by incorporating an opening in the CES lid which is vertically aligned with the steam generator location.

A constructability requirement within Holtec SMR Top-Level Plant Design Requirements [32] is that construction considerations must also consider the full life of a SMR-300 plant and the future decommissioning and dismantling activities. This requirement has been captured in SMR-160 Design Standard for Decommissioning [36], which details guidance on Plant Layouts that limit the spread of contamination and facilitate dismantling and decontamination of radioactive equipment. This standard will be revised when the SMR-300 version becomes available.

Design control of Layout follows the processes as laid out in SMR Design Control [50]. A summary of these configuration management objectives and is discussed further in Part A Chapter 4 [3].

The Civil Engineering team by being responsible for the production of Design specification documents of buildings followed by the GA drawings, are owners of the layout of the SMR-300. When the civil team creates the initial building layouts, they use the best available information such as equipment sizing, systems and components, number of tanks and use this to locate and size rooms. The civil models are then imported into SmartPlant 3D. Updating and revising these general arrangement drawings for the general layout of the plant must conform to the configuration processes laid out in SMR Design Control [50]. Additionally, equipment drawings / models are also imported when available into the 3D model by their respective responsible design teams.

During or after the initial layout of a systems routing, the layout is reviewed with a cross disciplinary team (Integrated Design Reviews: Design Development, Civil, Mechanical, Electrical / I&C, Operations, etc.) to provide any immediate feedback, which is incorporated into the various layouts in the 3D model. It is important to state that the 3D model is not “Quality Assured” – rather the GA drawings used to provide inputs into it are subject to Quality Assurance (QA), and the drawings produced from the model will be subject to quality and configuration control. A summary of this process is shown in Figure 10. An example of this would be Piping Isometrics, which as the Palisades refence design approaches the detailed design phase will be produced from the 3D layout. These drawings are subject to the Nuclear

[13] All spent fuel on-site shall be stored either within the containment structure or within the containment vessels of the dry cask storage technology to minimise the source term to be considered for plant events and reduce the overall risk of release of radionuclides.

Quality Assurance (NQA-1) Certification design process and configuration controls as set out in SMR Design Control [50].

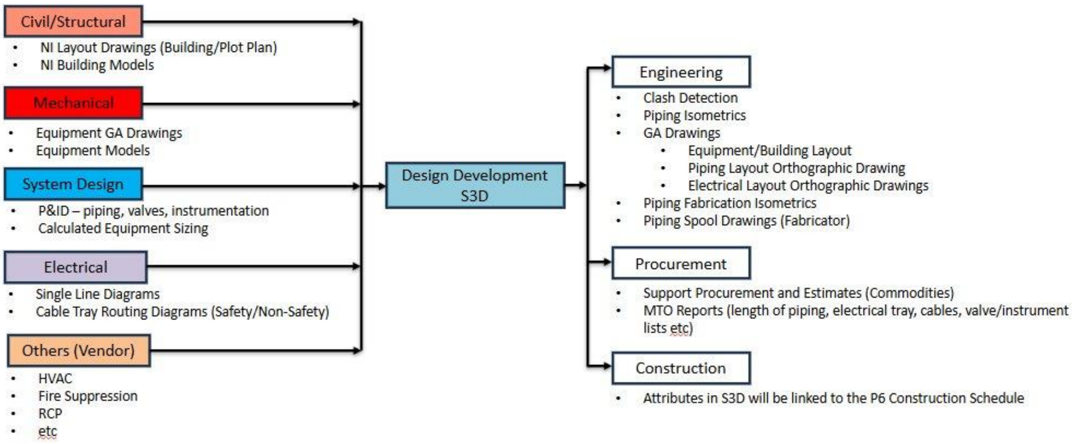

Figure 10: Layout Considerations in 3D Design Development

The Layout design will progress throughout detailed design and will be further supported by detailed analyses, for example, stress analysis work will be conducted for pipework, which will then incorporate new layout changes due to flexibility calculations, pipe supports etc. After this process the Design Integration Reviews (DIR), which consider Constructability, Operability, Maintainability and Safety (COMS), provide the formal cross disciplinary optioneering of the layout.

The following documents (non-exhaustive) also influence the detailed SMR-300 layout:

- HF guidance as set out in SMR-160 Design Standard for Human Factors: Maintenance, Inspection and Testing [43]. These HF design standards then inform the requirements to produce the relevant general arrangement drawings by the civil engineering team. Specific operational and HF input, is solicited in the pre-job brief for a drawing and then is captured in the redlines and comments of a document / drawing in line with the Holtec International NQA-1 design process.
- Equipment and Piping Layout Guidelines for Ensuring Radiation Exposures ALARA [51].
- SMR-300 Design Standard for Radiation Protection [35].
- SMR-300 Design Standard for Grouping and Separation [34] the primary purpose of which is for implementing appropriate grouping and separation methodologies to ensure that plant SSCs can perform their safety functions during and following a design basis event.
- SMR-300 Specification – Environmental Conditions [52].

The Outage Strategy for SMR-300 [53] documents the high-level plant outage execution strategy, including maintenance schedules, activity schedules, lifting and handling routes and equipment. Insights from previous design reviews and operational experiences are utilised to shape outage strategy. The Outage Strategy provides analysis on operations required for EIMT and refuelling. Lifting paths, laydown space allocation, capturing access requirements and work sequencing. While operations and EIMT feedback is already considered in the NQA-

1 design process, the outputs from the Outage strategy will further inform detailed design as maintenance requirements are assessed in greater detail, and provide requirements, recommendations and optimisations for the SMR-300 layout. EIMT is discussed in more detail in Part B Chapter 9 [7].

A Nuclear Site Health and Safety (NSHS) GDA Phase 1 Plan of Work has been prepared to address Phase 1 of the Holtec Construction, Design and Management (CDM) Strategy [73]. This is outlined within the Safety Management System Report [74], which sets the overall Holtec Britain safety management system. This includes the Office Health and Safety Manual [75] and the Design Safety Management Plan (DSMP). Further detail is provided in Part B Chapter 12.

SMR-300 design control and review is described in further detail in Part A Chapter 4 [3].

2.6.12 CAE Summary

The SMR-300 design is based on proven technology as far as is reasonably practicable to reduce first-of-a-kind engineering to minimise technology development and licensing risks. The SMR-300 design draws on the operating experience and lessons learnt from six decades of operating nuclear power plants, resulting in a greatly simplified plant with respect to construction, operation, inspection, and maintenance as compared to Gen-II and Gen-III LWRs.

Safety, environmental, radiation protection, safety analysis (including safety goals), performance, constructability and decommissioning philosophies have been outlined within this chapter that have derived the SMR-300 design. It is demonstrated that these philosophies satisfy US NRC requirements and are consistent with wider (e.g. IAEA) international guidance and best practice. The SMR-300 Plant Layout is influenced by the top-level design requirements, that embody the combined safety, environmental, radiation protection, performance, constructability and decommissioning philosophies, which, in themselves are set to meet US NRC requirements.

The design has followed an integrated design approach to safety in which accident resistance, core damage prevention, and accident mitigation are considered. The design utilises passive operating and safety features to prevent and, if necessary, mitigate the consequences of (US) design basis and beyond design basis accidents. Central to this, the SMR-300 design does not require operator action or reliance on off-site or on-site AC power for accident mitigation.

The approach to defence in depth provision aligns with the IAEA SSR 2/1 [48]. The SMR-300 design utilises passive operating and safety features to prevent and, if necessary, mitigate the consequences of design basis and beyond design basis accidents The SMR-300 utilises passive safety systems that operate without external power or operator action. Natural circulation within the reactor coolant system ensures that decay heat can be removed without pumps, while heat exchangers submerged in 1) a large gravity-fed water tank and 2) the AR provide long-term cooling capacity. The passive decay heat removal systems are designed with sufficient thermal capacity to maintain core cooling for a minimum of 72 hours without external power, and it can be replenished using low-pressure means, such as gravity refill from onsite water storage tanks. The US PSAR will demonstrate that the US SMR-300 Reference plant will meet or exceed US NRC GDC and acceptance criteria.

The US safety analysis framework is identified for Deterministic and Probabilistic Safety Analyses, and the safety goals that require demonstration in accordance with US requirements. These analyses will be undertaken to support demonstration of the safety goals in support of the US PSAR that will be prepared in support of the US CPA. The US PSAR will support the Pre-Construction SSEC stage.

The codes and standards used in the design of the SMR-300 have been selected to meet the stringent requirements of the US NRC. These mature and established US codes and standards are internationally recognised, and commensurate with the importance of the safety functions being delivered. The codes and standards applied to the design of nuclear safety related SSCs of the SMR-300 are generally nuclear specific. Many of them represent good practice adopted on UK nuclear licensed sites and / or application in earlier successful GDAs.

2.7 UK GENERIC SMR-300 SAFETY, SECURITY AND SAFEGUARDS DEMONSTRATION APPROACH

Claim 1.2: The generic SMR-300 design will be shown to be compliant with UK nuclear safety and design principles while minimising the impact on the design stability of the global fleet.

An essential objective of the GDA process is that duty holders demonstrate that risks have been reduced or are capable of being reduced ALARP.

For the GDA, the location specific regulatory framework is necessary to consider. The site-specific requirements are not applicable at GDA, however the requirements of the generic site are, which are introduced in sub-chapter 2.8.

The SMR-300 has been designed for initial deployment in the US but with cognisance of wider deployability requirements. There differences in expectations between the UK and US regulatory environments (identified in the US / UK Regulatory Framework and Principles Report [72]), but also areas where regulatory expectations are equivalent or identical.

Claim 1.2 is decomposed into four further claims:

Claim 1.2.1 (see sub-chapter 2.7.4) is to show that the design and safety principles being used to develop the US SMR-300 Reference Plant broadly align with UK context expectations in order to provide assurance to a future licensee that it will ultimately be able to demonstrate the generic SMR-300 design against its own safety and design principles.

Claim 1.2.2 (see sub-chapter 2.7.7) is to demonstrate that the SSEC has assessed the generic SMR-300 design and demonstrated the equivalency of the codes and standards utilised in the design of the US SMR-300 reference plant and UK codes and standards. This will show that the generic SMR-300 design will be capable of compliance with UK codes and standards.

Claim 1.2.3 (see sub-chapter 2.7.8) shows that suitable design management arrangements are in place to support wider deployment of the SMR-300, that is suitably optimised for location specific (UK regulatory requirements). This is to demonstrate an ALARP assessment methodology that ensures the risks associated with the design are minimised whilst also supporting wider deployability.

Claim 1.2.4 (see sub-chapter 2.7.9) is to ensure that a robust assessment process is in place such that metrication risks associated with the SMR-300 deployment in the UK are appropriately assessed in support of the overall ALARP demonstration.

In support of the demonstration of these claims, this sub-chapter contains further context, including:

- A high-level overview of the UK regulations, including the Security and Safeguards frameworks, which contextualises the demonstration of Claim 1.2.1.
- A description of the UK Numerical Targets and Dose Acceptance Criteria, followed by the approach to the definition of UK safety functions and identification of preliminary SSC classifications. This provides context for the demonstration of Claim 1.2.2, when considering UK context potential SSC classification requirements.

2.7.1 UK Regulations

There are marked differences in general approach between the regulators in the UK and the US, and many differences of detail. The ones of major importance at the GDA stage are those relating to the UK safety case process. It is these differences that need consideration in moving a US design into the UK. There are also differences in the licensing and permitting regime that will need to be considered in due course, as the licensing process occurs after GDA when the generic SMR-300 is offered for construction on a GB site.

While differences in regulatory style exist between the US and UK, it is noted that both regimes are very mature. The development of nuclear regulation in both countries has responded to extensive international co-operation, especially through organisations like the IAEA, to which both the US and UK regulators have been major contributors. On this basis, good engineering in the US should imply good engineering in the UK and it is noted that:

- Holtec International has used sound and historically successful engineering principles to develop the SMR-300 design in the US, using codes and standards that are, in many instances, already familiar to the UK regulatory system.
- The US is a signatory to various UN nuclear treaties and conventions, therefore the NRC regulatory system is consistent with IAEA nuclear safety guidance, as is the ONR system.

It is likely that both US and UK regulatory frameworks will settle on similar engineering solutions for a given technical problem as both UK and US systems are consistent with the requirements and guidance of the IAEA.

The nuclear safety regulatory framework in GB is explained in A Guide to Nuclear Regulation in the UK by the ONR [76]. There are other ONR guides explaining in detail the many different aspects of nuclear safety regulation in the UK but, in addition to the Guide to Nuclear Regulation [76], a general understanding of safety, security, safeguards, environmental protection and waste management, sufficient for the needs of this report is provided by the following:

- ONR, Security Assessment Principles for the Civil Nuclear Industry 2022 Edition, Version 1 [77].
- ONR Nuclear Material Accountancy, Control and Safeguards Assessment Principles [78].
- ONR, ONR-GDA-GD-006, Issue 1, New Nuclear Power Plants: Generic Design Assessment Guidance to Requesting Parties [31].
- ONR, ONR-GDA-GD-007, Revision 0, New Nuclear Power Plants: Generic Design Assessment Technical Guidance [79].
- ONR, Licensing nuclear installations [80].
- ONR, Safety Assessment Principles for Nuclear Facilities, 2014 Edition, Revision 0 [65].
- ONR, Permissioning inspection – Technical Assessment Guides [81].
- EA, New Nuclear Plants: Generic Design Assessment guidance for Requesting Parties [82].
- EA, Process and Information Document for Generic Design Assessment of Candidate Nuclear Power Plants Designs Version 3 [83].

- NWS guidance: High Heat Generating Waste (HHGW) Specifications, Low Heat Generating Waste (LHGW) Specifications [84].
- Radioactive Substance Regulations (RSR) generic developed principles: regulatory assessment [85].

Both the UK and the US systems have compliance (prescriptive) aspects and goal-setting (non-prescriptive) aspects. The main difference is that in the UK, safety case and design aspects are regulated as non-prescriptive and subject to the Risk ALARP principle. In the US, custom and practice is that these aspects are heavily prescribed in regulatory guidance including the use of specific codes and standards.

The US approach to safety cases, captured by NUREG-0800 Chapter 15 [15], is comparable in status to a (very extensive) Approved Code of Practice (ACoP) in the UK. The ALARP principle doesn't apply but the companion principle of ALARA is used for radiation dose control.

2.7.2 UK Nuclear Regulatory Security Framework

The GSR [86] presents the overall nuclear security case and how the evolving design is compliant with the UK nuclear security framework. Nuclear security for the generic SMR-300 will be delivered via the following series of activities which, taken together, provide a structured, clear, and logical approach to the development of the conceptual security arrangements. The key steps of this approach are:

- The nuclear inventory comprising Nuclear Material (NM) and Other Radioactive Material (ORM) at the generic SMR-300 facility is identified.
- The Sensitive Nuclear Information on the SMR-300 site is identified.
- An appropriate threat is used to define the physical and cyber threat at the SMR-300 site and is regularly reviewed to accommodate developments in the threat and understand certain credible beyond Design Basis Threat (DBT) scenarios.
- The assets and areas within the SMR-300 facility requiring protection to prevent the sabotage of the nuclear material inventory are identified, and any Vital Areas are categorised.
- The assets and areas requiring protection to prevent the theft of the NM / ORM or Sensitive Nuclear Information (SNI) are identified (including categorisation for theft of NM / ORM).
- Protection against sabotage and theft is provided by a blend of protective physical, cyber and procedural measures to provide defence in depth.
- Areas within the nuclear facility are security zoned to facilitate the provision of graded protection, which is delivered by an Integrated Security Solution (ISS).
- The site security operations deliver the ISS, which is regularly tested and reviewed to confirm its ongoing validity and effectiveness during the plant lifecycle, within an effective security culture.

These activities form the basis for the generic SMR-300 nuclear security case.

Other assets of importance to nuclear safety (e.g., emergency response systems and equipment) may require protection from sabotage. However, these are not within the scope of the GSR but will be considered post GDA during the development of the site-specific security arrangements.

The identification of SNI is included in the above activities for completeness and will be assessed during future site-specific assessments.

2.7.3 International and UK Safeguards Framework

The Preliminary Safeguards Report [87] demonstrates Holtec's understanding of safeguards requirements at the generic (international and UK) level and how they are being accommodated in the generic design of the SMR-300. Its objectives are to:

- Present Holtec's understanding of the safeguards requirement at the generic (international and UK domestic) level and of relevant good practice.
- Outline at a high level the generic SMR-300 safeguards programme, i.e. how the safeguards requirements will be delivered for the generic SMR-300 through life, and progress in its implementation during the GDA.
- Present an outline of the generic SMR-300 safeguards case and the main safeguards claims, showing how these claims integrate with the SSEC, and progress on the development of the safeguards case.
- Provide the basis for the accommodation of safeguards requirements in the generic SMR-300 design, including information on the development of the safeguards design objective, safeguards design principles, and the implementation of safeguards by design.
- Present progress on the development of conceptual safeguards arrangements, including Qualifying Nuclear Material Flow and potential Material Balance Areas / Key Measurement Points.
- Outline the evolution from GDA Step 2 to site licensing in the safeguards area, in accordance with the SMR-300 safeguards programme.

2.7.4 Generic SMR-300 Safety, Security and Safeguards Principles Alignment Review

Claim 1.2.1. The design and safety principles being used to develop the generic SMR-300 are broadly aligned with relevant UK context expectations, with significant differences identified and used to inform design development.

Claim 1.2.1 is demonstrated at the GDA stage by the following argument.

Argument 1.2.1-A1: In order to provide assurance that a future licensee will be able to demonstrate the generic SMR-300 design against its own safety and design principles, alignment with a set of principles derived from the UK SAPs is presented within this GDA with gaps and risks identified to inform design development.

This sub-chapter presents the results of a review of the respective UK and US regulatory expectations that are relevant to the SMR-300 design. This is supported by:

- Holtec SMR-300 Safety, Security and Safeguards Principles Alignment Review [88].

The SMR-300 design has been developed by Holtec International principally to meet the US regulatory expectations. These expectations take the form of mature and established requirements, codes and standards that are internationally recognised. Many of these requirements, codes and standards reflect existing practices either adopted on UK nuclear licensed sites or have formed a part of earlier successful GDA applications.

Demonstrating that RGP is being used in the design of the SMR-300 is key to showing that risks can be reduced to ALARP. To this end the SAPs and SyAPs are considered by ONR to represent RGP. The SAPs support ONR when shaping their regulatory judgements on whether reducing risks to ALARP has been achieved.

It is a requirement of GDA Step 2 that the Requesting Party submits sufficient detail to ONR to be satisfied that the relevant SAPs and the SyAPs are likely to be fulfilled. Consequently, for the purposes of GDA, the SMR-300 is being assessed against the ONR SAPs / SyAPs and supporting documentation along with other relevant UK legislation such as the Ionising Radiations Regulations 2017 [64].

Holtec consider that the US requirements, codes and standards being used in the design of the SMR-300 represent good practice that is relevant to its deployment in the UK. Holtec also consider that this can provide confidence that the controls and safety measures prevailing in the US are at least as effective as those considered relevant good practice in the UK.

While the differences in the approach to regulation between the US and UK have been reviewed and identified, it is noted that both regimes are very mature. The development of nuclear regulation in both countries has responded to extensive international co-operation and has a common basis in the form of the safety standards and the security and safeguards guides of the IAEA.

This common basis is displayed through both countries being Contracting Parties to the Convention on Nuclear Safety [89]. Regarding the IAEA safety standards, one response of the Convention on Nuclear Safety to the Fukushima Daiichi accident was the adoption by the Contracting Parties of the Convention on Nuclear Safety in 2015 of the Vienna Declaration on Nuclear Safety [90]. Principle 3 of the Declaration states:

> "*National requirements and regulations for addressing this objective*[14] *throughout the lifetime of nuclear power plants are to take into account the relevant IAEA Safety Standards".*

The UK applies the IAEA safety standards and ensures that its own regulations, regulatory requirements and guidance are consistent with them, including the SAPs. The US NRC Regulatory Guides cite or reference relevant IAEA safety standards and guides.

It is on these grounds that the regulatory expectations in both countries can be broadly seen to have a common basis. Hence good engineering practice in the US should, in general terms,

[14] *Vienna Declaration on Nuclear Safety, 2015 Principle* 2. Comprehensive and systematic safety assessments

be seen to be equivalent to good engineering practice in the UK. However, while there may be a common basis for regulation there may also be divergence in some areas of detail.

Holtec SMR-300 Safety, Security and Safeguards Principles Alignment Review [88] presents the results of a review of the respective UK and US regulatory expectations that are relevant to the SMR-300 design. The report identifies areas where, in Holtec's judgement, there is broad alignment between the US SMR-300 design and safety principles and the Office for ONR SAPs [65] and SyAPs [77] and ONMACS [78].

The report provides a high-level route map to SSEC evidence to support Holtec's judgement on alignment with the SAPs. The location of evidence to support the security and safeguards alignment is also provided. For any identified potential misalignment the report identifies the key respective UK Design Challenges (identified in Table 3) which have been, or are being, raised against these at-risk areas. The aim is to provide confidence that the current principles and criteria against which the SMR-300 design is being developed will be aligned with UK regulatory expectations.

This work provides an additional means of ensuring that Design Challenges identified during GDA Step 2, address these potential risk areas. The Design Challenges have been mapped against the relevant SAP thematic areas. These Design Challenges will be progressed in accordance with the Holtec SMR-300 Design Management Process [23].

The criteria used to review the alignment of the SMR-300 design principles and criteria with the ONR SAPs and SyAPs are described in Table 9, including a summary of the conclusions of the SMR-300 Safety, Security and Safeguards Principes Alignment Review [88].

Table 9: ONR SAP SyAPs and ONMACS Alignment Review

Category Description	Summary
Aligned: Good alignment between the US and UK principles and the design expected to develop to fully meet the UK principle.	As a reflection of the common basis for the regulatory frameworks in the US and the UK, Holtec made the judgement that many of the SMR-300 design principles and criteria are well aligned with the relevant principles in the ONR SAPs / SyAPs.
Aligned Once Resolved (AOR): Differences in alignment between the US and UK principles. Design at risk of not meeting UK principle but Design Challenges / GDA Commitments raised to address risk	For each area assigned the AOR category, a discussion of the potential shortfall in alignment is presented in Table 5 of the SMR-300 Safety, Security and Safeguards Principles Alignment Review [88] against the relevant SAP thematic area. The associated Design Challenges or GDA Commitments if appropriate is identified in response to the potential shortfall. It is recognised that further Design Challenges or GDA Commitments could still arise within GDA Step 2 (or beyond) as a result of the need for better alignment. However, items are judged as being AOR, where actions are underway, even if the outcome of the Design Challenge (or future challenges) are not yet clear.
Not Aligned: Differences in alignment in US / UK principles. Design at risk of not meeting UK principle and no / limited Design Challenges / GDA Commitments raised to address risk	No thematic areas were judged by Holtec for the Not Aligned category at GDA, though it is noted that this is dependent on the outcome of the Design Challenges identified in Table 3. Several themes may be at risk of being categorised as gaps if suitable safety justification and design decisions are not provided to resolve the identified Design Challenges.
Not Assessable: Principle not considered relevant to the SMR-300 or maturity of the safety case does not currently enable consideration of alignment against this principle, and this will be assessed against future safety cases.	Holtec judged that some of the ONR SAPs and SyAPs were not relevant to the SMR-300 or not relevant to the maturity of the design at this stage of GDA, Step 2 the Fundamental Assessment. These SAPs include, for example, the Engineering principles for Chemical Engineering and Graphite Reactor Cores.

In Holtec's judgement, good alignment has been shown between the SMR-300 design principles and relevant UK regulatory expectations in a large number of cases. However, in some instances it has not yet been possible to fully demonstrate alignment taking into account the maturity of SMR-300 design at the Fundamental Assessment stage. Potential risks in alignment are being progressed as specific Design Challenges (see Table 3) or are being captured as GDA Commitments to progress beyond Step 2.

Regarding post GDA considerations, any future UK nuclear site licensee for the SMR-300 will need to consider developing dedicated UK SMR-300 design, safety security and safeguards principles, by supplementing the US SMR-300 Top-Level Plant Requirements with other relevant UK context aspects. Further assessment of alignment and demonstration that the design meets these principles, will then be necessary to demonstrate that any SMR-300 to be deployed in the UK, is fully aligned with UK expectations.

2.7.5 UK Numerical Targets and Dose Acceptance Criteria

This sub-chapter describes the ONR SAP Numerical Targets that are necessary to comply with in the UK. The targets quantify ONR's risk policy and have been set to demonstrate whether radiological hazards are being adequately controlled, and risks reduced to ALARP.

SAPs NT.1-NT.3 support the application of the numerical targets defined below. These targets are applicable to the generic SMR-300 and cover both dose uptake and risks of death from radiation exposure, and are applicable to NO, AOOs, DBAs and Beyond Design Basis Accidents (BDBA), and to the assessments conducted for normal operating conditions and in DBA analysis, PSA and Severe Accident Analysis (SAA). Compliance with these targets is not discussed here but is discussed within the relevant chapters within Part B and summarised in Part A Chapter 5 [4].

2.7.5.1 Normal Operations

For NO, the dose targets given in Numerical Target 1-3 in Table 10, Table 11 and Table 12 are derived from the IRR17 [64]. Further information about their derivation and the compliance against these targets is demonstrated in Part B Chapter 10 [8].

Part B Chapter 10 [8] presents the strategy to ensure that external and internal exposures to OSWs and MoP during NOs of the generic SMR-300 are within legal limits and reduced to ALARP. It also discusses the UK expectations regarding the classification / zoning of areas containing radiation.

A radiation zoning scheme is to be established to classify ERCAs and CCAs according to anticipated personnel occupancy and access restrictions in all areas of the station during normal conditions.

The targets and a legal limit for effective dose in a calendar year for any person on the site from sources of ionising radiation are shown in Table 10:

Table 10: Target 1: Normal Operations Dose Targets – Individuals on Site (UK)

Exposed Group	Threshold	Dose Target (mSv)
Employee working with ionising radiation	Basic Safety Level (BSL) (Legal Limit)	20
	Basic Safety Objective (BSO)	1
Other employees on site	BSL	2
	BSO	0.1

The targets for average effective dose in a calendar year to defined groups of employees working with ionising radiation are shown in Table 11:

Table 11: Target 2: Normal Operations Dose Targets – Groups on Site (UK)

Exposed Groups	Thresholds	Dose Target (mSv)
Any group on site	BSL	10
	BSO	0.5

The target and a legal limit for effective dose in a calendar year for any person off the site from sources of ionising radiation originating on the site are shown in Table 12:

Table 12: Target 3: Normal Operations Dose Targets – Any Person off Site (UK)

Exposed Group	Threshold	Dose Target (mSv)
Any person off site	BSL (Legal Limit)	1
	BSO	0.02

2.7.5.2 Accident Conditions and Accident Analysis

The ONR SAP Numerical Targets for DBA represent criteria for assessing the safety of the facility's design and operations for faults that could have significant consequences. They are based on initiating fault frequencies and so take no account of the reliability of the claimed safety measures. Further information about their derivation and the compliance against these targets is demonstrated in Part B Chapter 14 [17].

The targets for the effective dose received by any person arising from a design basis fault sequence are presented in Table 13.

Table 13: Target 4: Design Basis Fault Sequences – Any Person (UK)

Exposed Group	Threshold	Dose Target (mSv)	Frequency Range (per annum)
On site	BSL	20	$>10^{-3}$
		200	$10^{-3} – 10^{-4}$
		500	$10^{-4} – 10^{-5}$
	BSO	0.1	-
Off site	BSL	1	$>10^{-3}$
		10	$10^{-3} – 10^{-4}$
		100	$10^{-4} – 10^{-5}$
	BSO	0.01	-

Targets 5 (see Table 14) and 7 (see Table 16) are set in terms of the overall (summated) risk impact to individuals from all the facilities on a site. Targets 6 (see Table 15) and 8 (see Table 17) for accidents apply to individual nuclear facilities rather than whole sites. Target 6 sets out frequency based BSLs and BSOs for a person on the site from a single accident. Target 8 is for any person off the site and provides BSLs and BSOs that represent the total frequency of all the accidents in each dose band.

The targets for the individual risk of death to a person on the site, from accidents at the site resulting in exposure to ionising radiation, are shown in Table 14:

Table 14: Target 5: Individual Risk of Death from Accidents – Any Person on the Site (UK)

Exposed Group	Threshold	Frequency (per annum)
Any person on site	BSL	10^{-4}
	BSO	10^{-6}

The targets for the predicted frequency of any single accident in the facility, which could give doses to a person on the site are shown in Table 15.

Table 15: Target 6: Frequency Dose Targets for any Single Accident – Any Person on the Site (UK)

Exposed Group	Threshold	Dose Target (mSv)	Predicted Frequency (per annum)
Any person on site	BSL	2-20	10^{-1}
		20-200	10^{-2}
		200-2000	10^{-3}
		>2000	10^{-4}
	BSO	2-20	10^{-3}
		20-200	10^{-4}
		200-2000	10^{-5}
		>2000	10^{-6}

The targets for the individual risk of death to a person off the site, from accidents at the site resulting in exposure to ionising radiation, are shown in Table 17:

Table 16: Target 7: Individual Risk of Death to People off the Site from Accidents (UK)

Exposed Group	Threshold	Predicted Frequency (per annum)
Any person off site	BSL	10^{-4}
	BSO	10^{-6}

The targets for the total predicted frequencies of accidents on an individual facility, which could give doses to a person off the site are shown in Table 18.

Further information about the derivation of PSA results and demonstration of compliance against these targets is presented in Part B Chapter 16 [68], which supports Part A Chapter 5 [4].

Table 17: Target 8: Frequency Dose Targets for Accidents on an Individual Facility – Any Person off the Site (UK)

Exposed Group	Threshold	Dose Target (mSv)	Predicted Frequency (per annum)
Any person off site	BSL	0.1-1	1
		1-10	10^{-1}
		10-100	10^{-2}
		100-1000	10^{-3}
		>1000	10^{-4}
	BSO	0.1-1	10-2
		1-10	10-3
		10-100	10-4
		100-1000	10-5
		>1000	10-6

SAA considers major but very unlikely accidents and provides information on their progression, both within the facility and beyond the site boundary. As the SAA forms an input to the PSA, it does not have a separate Numerical Target. Societal risks from severe accidents are addressed by Target 9. The targets for the total risk of 100 or more fatalities, either immediate or eventual, from accidents at the site resulting in exposure to ionising radiation are shown in Table 18.

Table 18: Target 9: Total Risk of 100 or More Fatalities

Exposed Group	Threshold	Predicted Frequency (per annum)
Any person on site	BSL	10^{-5}
	BSO	10^{-7}

Further discussion of likely compliance with Targets 1-9, based on preliminary safety analysis results, is provided in Part A Chapter 5.

2.7.6 UK Safety Functions and SSC Classification Expectations and Approach

2.7.6.1 UK Safety Functions and Safety Function Alignment

A safety function is a specific purpose that must be accomplished for safety. SSCs are items important to safety within the facility design which provide a safety function. Safety claims can also be made on the operators, which are HBSCs.

The Holtec high-level functions shown in Table 5 include both safety and non-safety functions, which have been derived from the three basic safety functions identified in Requirement 4 of the IAEA's Safety of Nuclear Power Plant Design [48], the US expectation defined in 10 CFR 50.2 [14] and the US NRC's Standard Review Plan BTP 7-19 [54]. These are then further broken down into system-level safety and non-safety functional requirements in the lower-tier documents in the Plant-Level Function Identification and Decomposition [33] (see sub-chapter 2.6.10) in accordance with the instructions for SSCs given in 10 CFR 50 [14] Appendix A GDC.

There are four fundamental safety functions expected for a nuclear site in the UK: three that are analogous with the IAEA's safety functions for reactor facilities (recorded in SAP ERC.1 'Design and operation of reactors') and a fourth that is usually used for facilities that handle sources of radiation (recorded in SAP RP.7 'Radiation Protection').

1. Control of reactivity.
2. Removal of heat from the core.
3. Confinement of radioactive material.
4. Control of radiation exposure.

The ONR SAPs [65] state that a UK safety case should 'provide sufficient information to demonstrate that engineering rules have been applied in an appropriate manner (e.g., all SSCs have been designed, constructed, commissioned, operated and maintained in such a way as to enable them to fulfil their safety functions for their projected lifetimes)'.

It should be noted that SSCs 'important to safety' consist of two subcategories under 10 CFR 50: 'safety-related' and 'non-safety-related'. While safety-related SSCs are defined in paragraph 50.2 of 10 CFR 50 [14] as relating to design basis events, the regulations do not provide an equivalent set of criteria for determining which non-safety-related SSCs are 'important to safety' (i.e. those that do not originate from protection against a design basis event). Nevertheless, the structured approach in 10 CFR 50 fulfils the UK expectation in the SAPs [65] EKP.4 that 'the safety functions should be delivered by a structured analysis', albeit the scope of the safety function identification is defined for design basis events (that being a subset of all accident conditions), whereas the UK expectation is that both NO and accident conditions are considered (see SAP [65] ECS.1). The concept of a controlled state and safe shutdown state in the standard US approach to categorisation and classification does need to consider how this is then applied to facilities and activities away from the reactor for which the definition of a controlled and safe shutdown state may not directly apply.

The approach adopted for UK deployment is centred around demonstrating equivalency between the 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. This formal UK aligned safety assessment has commenced during GDA Step 2, through the development of a PFS and a limited set of UK DBAA, which is described further below.

The categorisation and classification expectations which are derived from this UK aligned assessment, can then be compared with the existing SMR-300 design and its corresponding US classification. Work has commenced via relevant safety analysis and engineering disciplines, to demonstrate equivalency between the US and UK expectations and confirm that for all aspects, the SMR-300 design meets UK expectations. Where equivalency is at risk of not being demonstrable, then this may lead to a UK design challenge, potentially resulting in a modification to the design or requiring supplemental safety justification to demonstrate the current design reduces risks to ALARP. This equivalency demonstration is still in progress and is discussed further throughout the Part B chapters. UK design challenges identified to date are reported in Table 3 and a dedicated GDA Commitment (C_Faul_01) is identified to complete the safety assessment work beyond Step 2.

The detailed UK aligned safety assessment approach is set out in the Safety Assessment Handbook (SAH) [91] and this has been developed to be fully compliant with UK context expectations. This SAH sets out the UK DBAA approach, including guidance on the categorisation of UK safety functions.

The suggested scheme set out in the SAH [91] makes use of the three categories recommended in ONR SAP ECS.1 [77]:

- **Category A:** Safety functions that play a principal role in ensuring nuclear safety in that they are associated with the removal of intolerable radiological risks from design basis faults, either by prevention of the risks or reduction of the risks to broadly acceptable levels.
- **Category B:** Safety functions that make a significant contribution to nuclear safety in that they are associated with the removal of radiological risks outside the design basis by either preventing the risks or reducing the risks to broadly acceptable levels for foreseeable events and beyond design basis faults, which are identified in fault studies. Functions whose failure would lead to a demand on a Category A safety function are also categorised as B.
- **Category C:** Safety functions that do not fall into either of Categories A or B. They are mainly associated with the support of Category A or B safety functions or identified from ALARP or BAT analyses.

Part B Chapter 14 [17] presents an initial UK DBAA of the generic SMR-300. A PFS has been produced to support PSR v1 [92]. The development of the PFS in support of the GDA Step 2 process for the SMR-300 has been in two stages:

- **Stage 1:** Revision 0 which was focussed on ‘in-reactor’ design basis faults and a limited set of DEC events.
- **Stage 2:** Revision 1 covers a wider set of DEC events and a preliminary set of external hazards. The Consolidated Fault List (CFL) is also incorporated into the PFS for Rev 1 and additionally consideration has been made of how Internal Hazards would be incorporated into future Fault Schedule development.

The PFS has been informed by a limited (OPEX-based) fault and hazard identification study which has examined international and relevant PWR projects (including other GDA projects) and any novel or unique features of the SMR-300 design in order to identify a credible and complete set of faults.

The fault analysis process for the SMR-300 has been configured to demonstrate that UK context has been adequately addressed. UK DBAA refers to the full scope of fault analysis, not just operational occurrences and 'accidents' within the design basis (i.e. transients, internal events, internal and external hazards), and will provide a robust demonstration of the fault tolerance of the facility and of the effectiveness of its safety measures.

The initial faults selected for UK DBAA are design basis accidents that utilise the vast majority of the safety systems identified for the generic SMR-300. These have been specifically selected for the following reasons:

1. To provide confidence to the Requesting Party stakeholders that it has the necessary skills and competence to conduct a full UK DBAA and understands the fundamental construct of a UK DBAA including the application of categorisation of safety functions and classification of SSCs.
2. To provide confidence that the transient and accident analyses that have been conducted against the US NRC context contain adequate detail to be able to draw likely meaningful comparisons against the requirements of the UK regulatory framework and support a UK DBAA.
3. To provide confidence to the Requesting Party stakeholders that the novel aspects of the design that have been included within scope (notably the AR, PCC and PCH) are likely to meet UK expectations and therefore likely to be licensed within the UK.
4. To propose candidate Operating Rules (OR) and the methodology for deriving the Safe Operating Envelope (SOE) for the safe operation of the SMR-300 plant, as derived via the new discrete analyses.

Beyond GDA timescales, the scope of the UK DBAA will be widened to cover all fault scenarios and all modes of operation. Safety functions will be identified in a more detailed manner for faults in areas other than the reactor and at-power operation, including for shutdown modes, for the SFP, for fuel handling activities and for waste treatment and storage. Radiological consequence analysis for fault sequences will also be undertaken and evaluated against relevant acceptance criteria. Source term analysis will inform the derivation of normal operation safety functions for control of radiation exposure where required.

Completion of the full UK DBAA will identify all faulted condition safety functions in accordance with UK regulatory requirements for the four fundamental safety functions. The full DBAA will also identify the normal operation safety functions, where loss of the normal operating function results in a faulted condition.

Part B Chapter 19 [93] presents the approach to the development of a preliminary Engineering Schedule, which will ultimately include the safety functions identified by the UK DBAA. The Engineering Schedule will continue to develop to cover all SSCs during site specific development of the Pre-Construction Safety Report (PCSR) and to support later project lifecycles. The Engineering Schedule specifies the Safety Functional Requirements (SFR) that SSCs satisfy and is developed iteratively throughout design development and GDA.

2.7.6.2 UK Categorisation and Classification

The safety categorisation and classification methodology currently defined for the US SMR-300 Reference Plant by Holtec is explained in sub-chapter 2.6.2.2. It should be noted that the US approach leads only to a classification of the SSC, not a categorisation of the safety function per se. The existing safety classifications of the SMR-300 SSCs using the Holtec scheme (outlined in sub-chapter 2.6.2.2) are reported in the Part B chapters of the PSR and summarised in Appendix C.

Holtec acknowledge the existence of differences in the approach to safety categorisation and classification between the NRC regulatory guides and the UK expectations. Within the UK, the approach to the definition of safety functions and their categorisation and the subsequent classification of relevant SSCs is integrated with the assessment of hazards and faults, in that they are an extension of the overall approach to safety assessment, redundancy / diversity requirements and a demonstration of defence-in-depth.

SSC classification is the process by which SSCs are classified on the basis of their significance in delivering associated safety functions. The SAH sets guidance on the UK classification of SSCs which deliver of UK Safety Functions, using the three classifications recommended in ONR SAP ECS.2 [77]:

- **Class 1:** Any SSC that forms a principal means of fulfilling a Category A safety function.
- **Class 2:** Any SSC that makes a significant contribution to fulfilling a Category A safety function or forms a principal means of ensuring a Category B safety function.
- **Class 3:** Any other SSC contributing to a categorised safety function.

The main expectation for the safety system or SSC classification is that it is based on the safety function category that needs to be delivered by the system and its relative importance in delivering that safety function. This permits the classification process to include principal and secondary (or back-up) safety systems as part of the DiD provision (see Table 19).

Table 19: Initial Classification of SSCs

Safety Function Category	SSC Classifications		
	Principal Means	Secondary Means	Other Means
Category A	Class 1	Class 2	Class 3
Category B	Class 2	Class 3	Class 3 (if needed)
Category C	Class 3	Class 3 (if appropriate)	Class 3 (if appropriate)

For SSCs that are included within the initial UK DBAA, preliminary UK safety classifications have been identified and these are shown in Table 20.

Table 20: Preliminary UK SSC Safety Classifications

SSC	Preliminary UK Safety Classification	SSC Classification Equivalency Demonstration
SCRAM (RTB / CRP / CDM)	Class 1	Part B Chapter 2 Reactor
PDH / SDH	Class 1	Part B Chapter 1 RCS and ESFs

SSC	Preliminary UK Safety Classification	SSC Classification Equivalency Demonstration
RCPB	Class 1	Part B Chapter 1 RCS and ESFs Part B Chapter 2 Reactor Part B Chapter 5 Reactor Supporting Facilities
PSVs	Class 1	Part B Chapter 1 RCS and ESFs
PCM	Class 1	
ADS	Class 1	
PCH	Class 1	
CS / CIVs	Class 1	Part B Chapter 1 RCS and ESFs
CES	Class 1	Part B Chapter 20 Civil Engineering
MCH	Class 1	Part B Chapter 1 RCS and ESFs
PSS	Class 1	Part B Chapter 4 Control and Instrumentation Systems
DCE	Class 1	Part B Chapter 6 Electrical Engineering

Based upon the preliminary UK SSC safety classifications determined by the initial UK DBAA, Part B Chapter 14 [17] identifies and discusses in detail several areas where additional evidence will be needed to meet UK expectations. These are reflected as design challenges in Table 3.

Beyond GDA timescales, the scope of the UK DBAA will be widened to cover the full scope of safety assessment and all modes of operation. Safety measures will be identified in a more detailed manner for faults in areas other than the reactor and at-power operation, including for shutdown modes, for the SFP, for fuel handling activities and for waste treatment and storage. Radiological consequence analysis for fault sequences will also be undertaken and evaluated against relevant acceptance criteria. Work will continue via relevant safety analysis and engineering disciplines, to identify UK expectations and demonstrate equivalency between the US and UK expectations and confirm that for all aspects, the SMR-300 design meets UK expectations with the outputs of the UK DBAA being used to inform the detailed design.

At the generic Pre-Construction SSEC maturity, the documentation created at the PSR stage will be expanded upon to have a fully developed safety assessment, including a full set of design basis faults, whereby all credible faults have been identified, and their fault sequences developed such that suitable and sufficient safety measures are identified. This will identify the full set of UK aligned safety classifications for normal operations and faulted conditions for the generic SMR-300 SSCs, against which equivalency of the SMR-300 design will need to be demonstrated.

2.7.7 UK Codes and Standards

Claim 1.2.2. Equivalency of Codes and Standards selected for the generic SMR-300 design, with relevant UK Codes and Standards, is demonstrated across the SSEC.

Claim 1.2.2 is demonstrated at the GDA stage by the following argument.

Argument 1.2.2-A1: The capability of the SMR-300 design to comply with UK Codes and Standards has been assessed at GDA to a sufficient level to support a fundamental assessment, with gaps and potential shortfalls identified to inform design development.

This sub-chapter describes the codes and standards evaluation strategy undertaken during GDA to demonstrate the applicability, adequacy and sufficiency of the codes and standards supporting US SMR-300 reference plant design aspect or SSEC topic area, that are applicable to identified UK RGP.

An essential objective of the GDA process is that duty holders demonstrate that risks have been or will be capable of being reduced to ALARP. A key element of ALARP is the demonstration of the application of appropriate design codes and standards. It is worthwhile to note that, for the overall safety case, there is a requirement for adequate emphasis on ALARP and BAT, which underpins all activities within the scope of the UK Health and Safety legislation. The application of prescriptive codes alone is insufficient to satisfy UK legislative requirements for the overall safety case. ALARP is addressed within each chapter of the PSR and summarised in Part A Chapter 5 [4]. The demonstration of BAT to the generic SMR-300 is demonstrated in PER Chapter 6 Demonstration of BAT [63] consistent with PSR maturity.

The codes and standards used in the design of the SMR-300 have been selected to meet the stringent requirements of the US NRC (see sub-chapter 2.6.9.3). The codes and standards applied to the design of nuclear safety related SSCs of the SMR-300 are generally nuclear specific. Many of them recognised as RGP in the UK nuclear industry, from existing practices adopted on UK nuclear licensed sites and / or application in earlier successful GDAs.

The Codes and Standards Report [37] established the strategy for the evaluation of the relevance, applicability, adequacy and sufficiency of codes and standards of the SMR-300 in the UK context during GDA. Evaluation can be necessary even when an applied code or standard is recognised as UK RGP and it is commensurate with the relevant classification, safety functions and required reliability. This is because the approach of applying the code or standard alone may not be consistent with UK RGP and may need to be supplemented or modified. Furthermore, UK RGP evolves with time. Therefore, it is necessary to establish whether the applied codes and standards represent current RGP. Further, it is possible that the version of an applied code or standard may not be the most recent version. In this case, it is necessary to demonstrate that the version applied is appropriate or consistent with RGP.

All relevant Part B chapters and PER SSEC chapters include specific sub-chapters outlining the applicable UK Codes, Standards, Methodologies, Regulations or Legislation. These sub-chapters present evaluation of the applicability, adequacy and sufficiency of the codes and standards supporting SMR-300 design aspect or SSEC topic area that are applicable to its respective scope, against the identified UK RGP.

Part A Chapter 5 [4] presents a summary of the Codes and Standards evaluation undertaken across the SSEC. A comprehensive assessment has been undertaken across the SSEC where it has been shown in general terms, that the generic SMR-300 design will be capable of being demonstrated against UK RGP codes and standards. There are instances where specific topic area codes and standards assessments have identified design challenges at GDA. These are identified within the relevant Part B SSEC chapters and PER chapters, summarised in Part A Chapter 5 [4] and also presented in Table 3. Further, risks have been identified to be managed in detailed design or are subject to GDA Commitment(s).

Topic areas associated with undertaking safety assessment of the generic SMR-300 design have demonstrated appropriate assessment methodologies that are in accordance with UK

RGP. These have been implemented to differing levels across the SSEC to what is considered suitable for each topic area at the GDA stage (e.g. an initial UK DBAA in Part B Chapter 14 [17] – see sub-chapter 2.7.6). These will be implemented in full during detailed design and inform the generic SMR-300 design process where necessary.

2.7.8 ALARP and SMR-300 Design Stability

Claim 1.2.3. The impact of location specific requirements on global fleet deployment of the generic SMR-300 are minimised, such that it is optimised for safety and environmental aspects.

[REDACTED]

Holtec are implementing a pragmatic and holistic approach to ALARP and BAT which builds on the design that is developing in the US. The design stability process is intended to focus effort on addressing the greatest risk and supporting the safety (nuclear safety, conventional safety, environmental protection) and operational factors associated with a deployable reactor design. This is important as lower-level risks and ALARP judgements being made on a topic by topic, fault by fault, system by system basis, are at risk of challenging the broader benefits of a pragmatic cost-effective approach, which supports the deployment of the SMR-300. Security and safeguards have been specifically excluded from these considerations as the concept of ALARP does not similarly translate to security and safeguards.

This sub-chapter summarises the key aspects of the ALARP demonstration and the design stability process, and links it to demonstration of application of the process across the SSEC and is supported by:

- Design Stability Toolkit [94].
- Design Management Process [23].
- ALARP Guidance document [95].
- Part A Chapter 5 [4].

2.7.8.1 Design Stability Toolkit

Where the SSEC assessment identifies design risks against the DRP, initial considerations of the significance of the risk are undertaken in accordance with the Risk Management Plan [96]. A Design Stability Toolkit [94] supports this part of the Design Management Process [23] and provides guidance to topic leads on the key aspects to consider when making judgements on design stability and whether the benefits of not changing the design outweigh the negatives of the design risk.

[REDACTED]

Holtec plan to utilise the arguments above, supported by evidence, as part of the overall demonstration that risks for the SMR-300 are reduced to ALARP.

2.7.8.2 Overview of ALARP Design Process

Part A Chapter 5 [4] specifically addresses the methodology associated with applying the ALARP principle. It presents the overall SSEC summary of ALARP.

These align with the four tests of the ALARP demonstration, shown in Figure 11 below:

Figure 11: The Four Tests of ALARP

The ALARP principle requires the following tests to demonstrate ALARP:

- Ensuring that legislative / regulatory requirements are complied with.
- The design follows RGP and that it has been used in the development of the design.
- Assessment of the Tolerability of Risk (ToR) through comparison with the nine numerical Targets in ONR SAPs which translate the ToR framework [65] demonstrating that the risk is at least in the tolerable region i.e. meets Basic Safety Levels.
- Options should only be disregarded if the sacrifice is considered to be grossly disproportionate to the benefits of risk reduction that would be gained.

An introduction to the ALARP principle and its legislative status within the UK is provided in Part A Chapter 5 [4]. Part A Chapter 5 [4] also provides a summary from the Part B chapters for the design stage to show that the risks to workers and the public are tolerable and ALARP.

Each Part B chapter summarises any safety significant option evaluations relevant to the scope of that chapter, to support the demonstration that there are no further options to reduce risk. This includes potential design challenges that have been derived by the PSR SSEC at GDA. Relevant GDA Commitments, and any proportionate optioneering, will also be discussed within the relevant PSR chapter, to provide the appropriate reference documentation as part of the evidence to demonstrate that there are no further options to reduce risk. There is therefore a high level of alignment between this part of the Part B chapters and the Design Management Process.

The ALARP Guidance document [95] includes the regulatory principles of ALARP and provides an optioneering methodology for undertaking ALARP assessment. The document is applicable to the design phase of the lifecycle as part of the GDA and for the design development post-GDA for the generic SMR-300 into the construction, commissioning, operation and decommissioning phases

Part A Chapter 4 [3] supports the lifecycle elements of the ALARP demonstration, with respect to the management of safety and quality assurance for the construction, commissioning, operation and decommissioning of the SMR-300.

2.7.9 Metrication

Claim 1.2.4 Metrication risks associated with the UK SMR-300 deployment shall be identified, assessed and justified through the Holtec safety management arrangements.

Claim 1.2.4 is demonstrated at the GDA stage by the following argument.

Argument 1.2.4-A1: A methodology has been developed to ensure the metrication risk associated with the generic SMR-300 design will inform the overall ALARP demonstration.

This sub-chapter presents an overview of the strategy for metrication of the generic SMR-300, such that metrication risks associated with the UK SMR-300 deployment shall be identified, assessed and justified through the Holtec safety management arrangements. These arrangements will ensure that any potential issues will be identified and assessed as part of their contribution to the overall ALARP justification of the generic SMR-300. This is supported by:

- Metrication Safety Strategy Overview [97].
- Metrication Hazard Assessment Process and Affected Areas report [98].
- Metrication Pilot Study Analysis Report [99].
- Design Management Process [23].

The entirety of the SMR-300 design is in imperial units. However, the UK regulator expects all designs and safety case to be presented in Standard International units, specifically noting ONR Guidance to Requesting Parties [31] below:

> *“Where existing documentation is used, ONR requires that the generic safety and security cases are presented in, and the NPP will be built and operated using SI (International System) Units. The safety and security cases should be written in English.”*

Previous GDAs for reactors designed in imperial units required supplemental work to demonstrate that the relevant Requesting Party had adequately justified its position on a quasi-metric approach to the design, construction, and operation of their reactor (GDA RI GI-AP1000-ME-02 [100]).

[REDACTED]

2.7.9.1 Metrication Affected Systems

[REDACTED]

2.7.9.2 Metrication Hazard Analysis Technique and Pilot Study

[REDACTED]

[REDACTED]

Figure 12: Metrication Risk Systems Hierarchy

[REDACTED]

2.7.9.3 Metrication Assessment Post GDA

It is recognised that the potential risks associated with metrication should be adequately understood and assessed in order that the proposed design solution can be tested and amended if necessary. A process to assess potential metrication risks for the UK deployment of the SMR-300 has therefore been developed and piloted. The Requesting Party commits to utilising this process to ensure metrication risks are clearly understood and considered, such that risks are reduced to ALARP. This is raised as GDA Commitment C_Metr_113 (see sub-chapter 2.9.3)

[REDACTED]

[REDACTED]

Figure 13: Post GDA Metrication Assessment Process

[REDACTED]

2.7.10 CAE Summary

The SMR-300 has been designed for the US market. There are areas of difference in expectations between the UK and US regulatory environments, but also areas where regulatory expectations are equivalent or identical. Broad alignment has been shown between the US SMR-300 design and safety principles and the ONR SAPs [65], SyAPs [77] and ONMACS [78]. However, in some instances it has not yet been possible to fully demonstrate alignment when accounting for the maturity of SMR-300 design at the Fundamental Assessment stage. Where there is a risk of the design not meeting UK regulatory expectations, to support a fundamental assessment, specific Design Challenges (see Table 3) have been raised or are being captured as GDA Commitments to progress beyond Step 2.

The safety assessment approach adopted for UK deployment of the generic SMR-300 is centred around demonstrating equivalency between the US SMR-300 Reference Plant design, and UK expectations for the categorisation of safety functions and classification of SSC. This is achieved through the application of formal safety assessment techniques, which are consistent with UK context expectations. The SSEC at PSR presents an initial UK DBAA of the design basis accidents that utilise the vast majority of the safety systems identified for the generic SMR-300. The categorisation and classification expectations which are derived from this UK aligned assessment, can then be compared with the existing SMR-300 design and its corresponding US classification. Where equivalency is at risk of not being demonstrable, then this may lead to a UK design challenge, potentially resulting in a modification to the design or requiring supplemental safety justification to demonstrate the current design reduces risks to ALARP. This equivalency demonstration is still in progress and is discussed further throughout the Part B chapters. UK design challenges identified to date are reported in Table 3. A dedicated GDA Commitment (C_Faul_01) is identified to complete the safety assessment work beyond Step 2.

A comprehensive assessment has been undertaken across the SSEC where it has been shown in general terms, that the generic SMR-300 design will be capable of being demonstrated against UK RGP codes and standards. There are instances where specific topic area codes and standards assessments have identified design challenges at GDA. These are identified within the relevant Part B SSEC chapters and PER chapters, summarised in Part A Chapter 5 [4] and also presented in Table 3. Further, risks have been identified to be managed in detailed design or are subject to GDA Commitment(s).

The SSEC provides the overall ALARP assessment of the generic SMR-300 design. All Part B chapters, and PER chapters provide assessments of the design against UK RGP. Any potential design risks are identified and considered in accordance with the ALARP design management arrangements in place. These arrangements also consider the deployability of the SMR-300 design within the optioneering process. This is where the design is stable and meets US requirements, but there is a gap to meet UK specific legal duties or UK specific deployment factors, where supplemental design optimisation will be undertaken to demonstrate ALARP and BAT. Progression of design risks and design challenges are ongoing beyond GDA.

[REDACTED]

2.8 GENERIC SITE ENVELOPE

Claim 1.3: An appropriately conservative and bounding GB-context generic site envelope is derived for the generic SMR-300 GDA.

In order to assess the environmental impact of the operation of the generic SMR-300 reactor site at a generic UK location, it is necessary to define the environmental characteristics of the site. For the GDA to be of benefit, the defined site envelope must present characteristics which are suitably bounding of any potential future sites in GB. The EA GDA guidance for Requesting Parties using the Initial Radiological Assessment Tool 2 (IRAT2): Part 2 [101] states that the Requesting Party must provide:

- A description, and the characteristics, of the generic site (or sites) that the Requesting Party will use for its assessment of radiological impacts and conventional impacts in EA's GDA Guidance for Requesting Parties [82] on people and the environment.

Claim 1.3 has therefore been decomposed into the following argument:

Argument 1.3-A1: The defined site envelope presents characteristics which are suitably bounding of any potential future sites in GB marine, atmospheric, freshwater, direct shine and short-term discharge parameters, ground conditions, grid connections, local population and definition of credible External Hazards.

This sub-chapter defines the GB GSE for the purposes of this GDA. Sub-chapter 2.8.1 provides a description of the generic site, including the assumptions and characteristics that define the coastal sites and the freshwater generic site. Sub-chapter 2.8.2 provides further information, including grid connection requirements, and the density and distribution of the local population. Sub-chapter 2.8.3 defines the generic site external hazards. This is supported by:

- Generic Site Envelope Report [6].
- Grid Code Compliance Strategy [102].
- Part B Chapter 6 [103].

2.8.1 Generic Site Description

Figure 1 presents a conceptual site layout for the generic SMR-300. Two generic sites have been selected to provide bounding Radiological Impact Assessments (RIA), considering liquid effluent discharges to a coastal site and to a lake.

The Generic Site Envelope Report [6] presents detailed assumptions for both a Generic Coastal Site and a Generic Freshwater Site. The assumptions include topography, nearest human receptors and habits of exposed groups, reference organisms (wildlife) applicable to the terrestrial and marine environments, discharge routes (including atmospheric and marine), assumed site specific meteorological conditions and distances from external exposure sources to site boundary or dwellings.

The Generic Site Envelope Report [6] defines the following parameters: marine, atmospheric, freshwater, direct shine, short term discharge and detailed site assumptions.

Demonstration that these parameters are bounding for a potential reactor site in England or Wales is presented within sensitivity analyses. As the discharge fingerprints are yet to be defined, OPEX of radionuclides typically recorded for discharges from PWR sites was obtained. The radionuclides were selected from a sample of data for European PWR reactors recorded on the European Commission Europa Radiation Discharges Database (RADD) [104].

These assumptions and parameters inform RIA based upon the generic SMR-300 at the generic site, which are presented in PER Chapter 3 [105].

2.8.2 Generic Site Information

Generic site information includes features of a site that can be defined on a qualitative basis.

2.8.2.1 Ground Conditions

The Requesting Party acknowledges that the ground conditions will vary across all of the prospective sites considered within the GB GSE and that the design of the foundations and embedded retaining structures may need to be adapted at the site-specific stage due to the broad envelope defined in the GSER.

All sites considered in the GB GSE have previously supported NPPs providing confidence that a geotechnical solution can be developed for each site.

At the site-specific stage a full comprehensive ground investigation will be undertaken to determine the properties and characteristics of the ground conditions present.

2.8.2.2 Heat Sink

Seven of the sites considered within the development of the GB GSE are located on the coast with access to seawater. However, Oldbury is situated in Gloucestershire on the River Severn estuary and Trawsfynydd is located in the Snowdonia National Park with a man-made reservoir previously used for cooling water. The SMR-300 design has the flexibility to be adapted to suit the water sources available at a prospective future site and can be sited in locations with severe water restrictions utilising hybrid or fully Air-Cooled Condenser (ACC) configurations.

For the purposes of assessing the impact of the generic SMR-300, it is assumed that the design uses a forced mechanical draft cooling tower configuration to remove normal operation heat loads. This does not foreclose the final design of the generic SMR-300 utilising once-through or air-cooled cooling technology, or a hybrid solution at the site-specific stage.

2.8.2.3 Ultimate Heat Sink

The AR is the generic SMR-300’s ultimate heat sink (as defined in accordance with the NRC guide DG-1275 [19]) and is described in sub-chapter 2.3.4.2.

In the event of a postulated accident, the Passive Containment Heat Removal System maintains the containment atmospheric pressure and temperature within the design limits by utilising the metal CS and the water inventory within the AR.

2.8.2.4 Grid Connections

The generic SMR-300 will be adapted to suit the UK Grid and the National Electricity Transmission System (NETS). Holtec is aware that adaptions will be required in order to deploy the SMR-300 to the UK electrical market. A Grid Code Compliance Strategy has been developed [102] which is based on a bounding set of requirements from UK and other Countries' Grid Codes. This is discussed in more detail in Part B Chapter 6 [103].

2.8.2.5 Loss of Offsite Power Event

A Loss of Offsite Power (LOOP) event is defined as the failure of both the main and alternative sources of electrical grid connection and supply of offsite power.

The UK National Grid is typically a reliable source of power for NPPs. The infrastructure continues to be developed undergoing upgrades through the planned installation of renewable energy sources and further modes of optimisation. However, the NPP licensee has no control over the National Grid and severe weather events can result in a LOOP event such as the storm that led to the Dungeness B LOOP event in 2013. Furthermore, the frequency of these types of weather events is projected to increase with climate change.

Current RGP in the UK is to consider an Extended Loss of Grid (ELOG) event of five days with a frequency of occurrence of 10^{-2}/yr. For the purposes of the generic SMR-300 GDA, the two LOOP events presented in Table 21 are considered as design basis frequent faults in the fault analysis. A long-term LOOP in excess of 72 hours is assessed within the initial UK DBAA (see Part B Chapter 14 [17]).

Table 21: Generic SMR-300 Initiating LOOP Events

Initiating LOOP Event	Frequency
LOOP up to 72 hours	10^{-2}/year
LOOP greater than 72 hour (ELOG)	10^{-2}/year

2.8.2.6 Density and Distribution of Local Population and Emergency Arrangements

An understanding of the density and distribution of the local population is required to inform an assessment on the expected off-site radiological dose. The operational discharges of the reactor should be assessed on a site-specific basis. However, at this stage, the Requesting Party is confident that all sites considered within the GB GSE are suitable for deployment.

EN-6 [106] has already assessed the demographics of each of the eight EN-6 sites concluding that the sites do not exceed the ONR's 'semi-urban' demographic siting criteria as described in NS-LUP-GD-001: Land Use Planning and the Siting of Nuclear Installations [107]. It's understood that the current proposal is to retain the 'semi-urban' demographic site criterion in the upcoming EN-7 National Policy Statement for Nuclear Power [108].

The Requesting Party acknowledges that a detailed assessment will still be required at the site-specific stage, as demographics may have changed since the publication of EN-6 [106].

All sites considered within the GB GSE have previously hosted NPPs and will have had emergency arrangements appropriate to NPP operations in place, although the extent of these arrangements may have been reduced for those sites currently hosting stations undergoing decommissioning or long-term care and maintenance. At the site-specific stage, these arrangements will be reviewed and updated where required to ensure compliance with the Radiation (Emergency Preparedness and Public Information) Regulations 2019.

2.8.3 Generic Site External Hazards

External Hazards are defined as, “natural or man-made hazards to a site and facilities that originate externally to both the site and its processes” [65]. The ONR requires External Hazards to be identified and treated as events that could give rise to possible initiating faults [79].

The scope of the GSER [6] considered the following topics related to External Hazards:

- The External Hazard identification process.
- The External Hazard screening methodology.
- The derivation of the GB GSE parameters to define the GB GSE.

2.8.3.1 External Hazards Methodology

Part B Chapter 21 [11] presents the External Hazard identification and screening methodology undertaken in the GSER [6] to identify credible External Hazards that are relevant to the GB context, can affect nuclear safety, and can be considered on a generic basis.

Step 1 of this methodology involved a comprehensive literature review of UK and International Regulatory Guidance documents, RGP, previous GDA submissions and previous work undertaken by Holtec. Step 2 of the methodology involved screening the extensive list of External Hazards into the scope of the GDA based on criteria aligned with RGP.

The result of this process is presented in Part B Chapter 21 [11] with the complete output of the External Hazard identification and screening process presented in Appendix A of the GSER [6].

2.8.3.2 Derivation of GB GSE External Hazards Parameters

Design Basis Events (DBEs) have been conservatively derived for each of the credible External Hazards identified in the GSER [6] to establish the GB GSE parameters. This derivation has been undertaken in accordance with the ONR SAPs [65]. Conservatism has been included in the approach for the derivation of the DBEs. The GB GSE Parameter has then been determined by adopting the DBE which bounds all the nine sites considered within the envelope, using a reasonable assessment of publicly available information. This approach is an implicitly conservative process, as it accounts for the worst-case site.

At this stage of the GDA, uncertainty analysis has not been conducted as part of the approach to derive GB GSE parameters, this will be possible at the site-specific stage with access to site-specific datasets, to allow conservative definition of hazard values with reasonable confidence intervals.

The Requesting Party has built on the success of previous GDA submissions by employing the same methods to quantitatively derive DBEs for each of the nine sites whilst ensuring those previously accepted methods are compliant with the latest industry guidance and RGP. For those hazards considered in the SMR-300 GDA Reference Design (e.g. seismic vibration, extreme wind, precipitation), the majority of actual hazard values used for GDA Reference Design are expected to be substantially in excess of those DBE values derived for the GB GSE, thus incorporating an additional level of conservatism to the overall approach.

The accepted methods in UK and European design codes have been used to derive values for the meteorological based External Hazards such as wind and ambient air temperature, these values have then been assessed against relevant meteorological record data where available.

There are several External Hazards such as lightning where a DBE cannot be accurately derived at this stage for each specific site. In these cases, a value has been adopted based on RGP which conservatively bounds the nine sites considered within the envelope. The derivation of each External Hazard GB GSE parameter is presented in Part B Chapter 21 [11] with further detail provided in the GSER [6].

The operational design life of the generic SMR-300 is 80 years. The UK Climate Projections (UKCP) [109] tools have been used to determine GB GSE parameters that account for reasonably foreseeable climate change throughout the design life of the facility.

Part B Chapter 21 [11] identifies those External Hazards where a margin or risk exists between the GB GSE parameters versus the GDA Reference design parameters. These hazards are further addressed in the ‘External Hazards US-UK Gap Analysis’ report [110]. Further information can be found in Part B Chapter 21 [11].

2.8.3.2.1 UK Climate Projections

The use of the UKCP [109] is considered RGP for estimating the effects of climate change. UKCP is a climate analysis tool developed by the Met Office for the Hadley Centre Climate Programme. UKCP18 are the latest set of projections released in 2018 which have built on the success of the previous projections released in 2009 (UKCP09).

The approach for the GB GSE has been to adopt values from relevant UKCP18 reasonably foreseeable climate change scenarios, by considering the median value for the high emission scenario and an extreme value from the medium emission scenario. Further detail and justification for the selected scenarios are provided in the GSER [6].

The Requesting Party acknowledges that the UKCP18 projections do not go beyond the year 2100 and that the full design life of the facility will exceed this date. The Requesting Party will recommend to future UK Licensees wishing to site the generic SMR-300 that the managed adaptive approach should be adopted for the full design life of the facility for affected hazards, especially sea flooding. It is expected that any future projections will be covered during the Periodic Safety Reviews by a future site Licensee.

Further explanation on the approach to climate change is provided in Section 4.4 of the GSER [6].

2.8.3.3 Summary of GB GSE External Hazards Parameters

Table 22 presents the External Hazard GB GSE parameters derived to establish the GB GSE and the characteristics of the generic site derived within the GSER [6] to be used as the basis for the safety analysis.

Part B Chapter 21 [11] demonstrates that in the preliminary evaluation of the GDA Reference Design, for many of the External Hazards, significant margin is present between the GDA Reference Design parameters and the GB GSE Parameters providing confidence in the robustness of the design in the context of a UK deployment.

Table 22: Summary of GB GSE External Hazards Parameters

[REDACTED]

2.8.4 CAE Summary

Nine prospective sites in Great Britain have been considered in the development of the GB GSE. This approach results in a broad conservative envelope providing confidence that the generic SMR-300 is suitable for deployment at the existing civil nuclear power station sites in Great Britain. Two generic sites have been selected to provide bounding RIA which consider liquid effluent discharges to a coastal site and to a lake. The assumptions that constitute these sites are defined. The parameters identified within these assumptions are demonstrated to be bounding for a potential reactor site in England or Wales. Exposure pathways for atmospheric discharges and liquid discharges are also defined.

[REDACTED]

DBEs have been derived for each of the credible External Hazards identified for the generic site, which are of a magnitude that is bounding of the sites considered within the development of the GB GSE to establish the GB GSE parameters.

2.9 CHAPTER SUMMARY AND CONTRIBUTION TO ALARP

This sub-chapter provides an overall summary and conclusion of the General Design Aspects and Site Characteristics and how this chapter contributes to the overall demonstration of ALARP for the generic SMR-300. Part A Chapter 5 [4] 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 RGP.
    - Evaluation of Risk and Demonstration Against Risk Targets.
    - Options Considered to Reduce Risk.
    - GDA Commitments.
- Conclusion.

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

2.9.1 Technical Summary

This chapter directly supports Claim 1.

Claim 1: The Generic Holtec SMR-300 design, and safety case are developed using integrated safety management arrangements that take cognisance of relevant good practice in the context of the UK regulatory regime.

Claim 1 has been decomposed in sub-chapter 2.5 into three further claims, with two further Level 2 claims supporting Claim 1 also covered in PSR Chapters A4 and A5.

Claim 1.1. The US Reference SMR-300 Plant design is derived from US design and International good practice to demonstrate compliance with US NRC requirements.

Claim 1.1 supports Claim 1 by demonstrating that the design principles, codes and standards used in the design of the SMR-300 have been selected to meet the stringent requirements of the US NRC. These mature and established US codes and standards are internationally recognised and are commensurate with the importance of the safety functions being delivered. The codes and standards applied to the design of nuclear safety related SSCs of the SMR-300 are generally nuclear specific, many of them are from existing practices adopted on UK nuclear licensed sites and / or application in earlier successful GDAs.

The SMR-300 design is based on proven technology as far as is reasonably practicable to reduce first-of-a-kind engineering to minimise technology development and licensing risks. The SMR-300 design draws on the operating experience and lessons learnt from six decades of operating nuclear power plants, resulting in a greatly simplified plant with respect to construction, operation, inspection, and maintenance as compared to Gen-II and Gen-III LWRs.

Safety, environmental, radiation protection, safety analysis (including safety goals), performance, constructability and decommissioning philosophies have been outlined within this chapter that have derived the SMR-300 design. It is demonstrated that these philosophies satisfy US NRC requirements and are consistent with wider (e.g. IAEA) international guidance and best practice.

The design has followed an integrated design approach to safety in which accident resistance, core damage prevention, and accident mitigation are considered. The approach to defence in depth provision aligns with the IAEA SSR 2/1 [48]. The design utilises passive operating and safety features to prevent and, if necessary, mitigate the consequences of (US) design basis and beyond design basis accidents. Central to this, the SMR-300 design does not require operator action or reliance on off-site or on-site AC power for accident mitigation. The US PSAR will demonstrate that the US SMR-300 Reference plant will meet or exceed US NRC GDC and acceptance criteria.

Claim 1.2: The generic SMR-300 design will be shown to be compliant with UK nuclear safety and design principles while minimising the impact on the design stability of the global fleet.

Claim 1.2 is decomposed into four further claims (described in further detail in sub-chapter 2.7). Claim 1.2 supports Claim 1 by showing that the design and safety principles being used to develop the US SMR-300 Reference Plant, broadly align with UK context expectations in order to provide assurance to a future licensee that it will ultimately be able to

- demonstrate the generic SMR-300 design against its own safety and design principles;
- demonstrate that the SSEC has assessed the generic SMR-300 design and demonstrated the equivalency of the codes and standards utilised in the design of the US SMR-300 reference plant and UK codes and standards;
- demonstrate that suitable design management arrangements are in place to support wider deployment of the SMR-300, that is suitably optimised for location specific (UK regulatory requirements);
- ensure that a robust assessment process is in place such that metrication risks associated with the SMR-300 deployment in the UK are appropriately assessed in support of the overall ALARP demonstration.

The SMR-300 has been designed for the US market. There are areas of difference in expectations between the UK and US regulatory environments, but also areas where regulatory expectations are equivalent or identical. Broad alignment has been shown between the US SMR-300 design and safety principles and the ONR SAPs [65], SyAPs [77] and ONMACS [78]. However, in some instances it has not yet been possible to fully demonstrate alignment when accounting for the maturity of SMR-300 design at the Fundamental Assessment stage. Potential shortfalls in alignment, to support a fundamental assessment, are being progressed as specific Design Challenges or are being captured as GDA Commitments to progress beyond Step 2.

A comprehensive assessment has been undertaken across the SSEC where it has been shown in general terms, that the generic SMR-300 design will be capable of being demonstrated against UK RGP codes and standards. There are instances where specific topic area codes and standards assessments have identified design challenges at GDA. These are identified within the relevant Part B SSEC chapters and PER chapters, summarised in Part A Chapter 5 [4] and also presented in Table 3. Further, risks have been identified to be managed in detailed design or are subject to GDA Commitment(s).

The safety assessment approach adopted for UK deployment of the generic SMR-300 is centred around demonstrating equivalency between the US SMR-300 Reference Plant design, and UK expectations for the categorisation of safety functions and classification of SSC. This is achieved through the application of formal safety assessment techniques, which are consistent with UK context expectations. Where equivalency is at risk of not being demonstrable, then this may lead to a UK design challenge, potentially resulting in a modification to the design or requiring supplemental safety justification to demonstrate the current design reduces risks to ALARP. The application of this equivalency demonstration has been shown at GDA and has resulted in a number of design challenges, presented in Table 3 and a dedicated GDA Commitment (C_Faul_01) is identified to complete the safety assessment work beyond Step 2.

Any potential design risks are identified and considered in accordance with the Design Management Process, which utilises supporting ALARP guidance to consider the deployability of the SMR-300 design, within the optioneering process. This is where the design meets US requirements, but there is a requirement for further justification to meet UK specific legal duties or UK specific deployment factors, where supplemental design optimisation will be undertaken to demonstrate ALARP and BAT.

At the GDA stage, areas of the generic SMR-300, including plant systems, safety case area or future lifecycle process which may be impacted by metrication have been identified. [REDACTED]

Claim 1.3: An appropriately conservative and bounding GB-context generic site envelope is derived for the generic SMR-300 GDA.

Claim 1.3 supports Claim 1 by defining the GB GSE for the generic SMR-300. In order to support future operation of the generic SMR-300 reactor site at a generic UK location, it is necessary to define the environmental characteristics of the site. For the GDA to be of benefit, the defined site envelope must present characteristics which are suitably bounding of any potential future sites in Great Britain. The definition of the generic site ensures that the generic SMR-300 can be shown to meet UK regulatory and legislative requirements and ensures that the generic SMR-300 SSCs will be adequately substantiated.

A generic site has been defined by the GB GSE for the purposes of this GDA. Nine prospective sites in GB have been considered in the development of the GB GSE, resulting in a broad conservative envelope providing confidence that the generic SMR-300 is suitable for deployment at the existing civil nuclear power station sites in Great Britain. The parameters identified within these assumptions are demonstrated to be bounding for a potential reactor site in England or Wales. DBEs have been derived for each of the credible External Hazards identified for the generic site, which are of a magnitude that is bounding of the sites considered within the development of the GB GSE to establish the GB GSE parameters.

Claim 1 is considered met with evidence appropriate to a PSR, noting GDA Commitment C_Metr_113 related to metrication.

2.9.2 ALARP Summary

2.9.2.1 Demonstration of RGP

The general design aspects and site characteristics have been described within this chapter.

Sub-chapter 2.6 identifies that the SMR-300 design process is following identified international good practice and is following the requirements of the US NRC. The SMR-300 design is based on proven technology as far as is reasonably practicable to reduce first-of-a-kind engineering to minimise technology development and licensing risks. Safety, environmental, radiation protection, safety analysis (including safety goals), performance, constructability and decommissioning philosophies that define the design are in accordance with US NRC requirements and are consistent with wider (e.g. IAEA) international guidance and best practice. The approach to defence in depth provision aligns with the IAEA SSR 2/1 [48]. The SMR-300 design utilises passive operating and safety features to prevent and, if necessary, mitigate the consequences of design basis and beyond design basis accidents.

Sub-chapter 2.7 presents the approach to safety demonstration for the generic SMR-300. The safety assessment approach adopted for UK deployment of the generic SMR-300 is centred around demonstrating equivalency between the US SMR-300 Reference Plant design, and UK expectations for the categorisation of safety functions and classification of SSCs. This is achieved through the application of formal safety assessment techniques, which are consistent with UK context expectations.

The SSEC provides the overall ALARP assessment of the generic SMR-300 design. All Part B chapters, and PER chapters provide assessments of the design against UK RGP. ALARP implications, including design challenges are described within Part A Chapter 5 [4] and it is shown by the summary of design challenges within Table 3 that the assessment of the generic SMR-300 design against UK RGP at the fundamental assessment stage is ensuring that the design will ultimately be demonstrable against UK RGP.

2.9.2.2 Evaluation of Risk and Demonstration Against Risk Targets

Sub-chapter 2.7.4 presents the UK ONR SAP Numerical Targets risk targets that the SSEC demonstrates compliance against. Safety analyses results will be compared with these targets, and further information about their derivation and the compliance against them is provided in Part A Chapter 5 [4].

2.9.2.3 Options Considered to Reduce Risk

The process for the assessment of risk reduction options is presented in the Design Management Process [23]. PSR Part A Chapter 5 [4] considers the holistic risk-reduction process for the generic SMR-300.

This chapter demonstrates how risk reduction has been included as a fundamental part of the SMR-300 design process. Its design evolution shows that it has been developed to consider many international requirements and has been designed to demonstrate compliance with US NRC regulatory requirements.

Sub-chapter 2.7 presents the approach to safety demonstration for the generic SMR-300. Where design risks have been identified by the SSEC for demonstration of the generic SMR-

300 against the non-prescriptive UK regulatory regulations, design challenges, design risks and GDA Commitments have been raised as appropriate within Part B chapters, and PER chapters. Those considered significant to safety are summarised within Part A Chapter 5 [4]. A summary of the design challenges raised at the fundamental assessment stage is presented in Table 3 as evidence that the arrangements currently in place throughout the SSEC can deliver a generic SMR-300 design that can be shown to be ALARP.

2.9.3 GDA Commitments

At Revision 1 there is one GDA Commitment identified for Part A Chapter 2 General Design Aspects and Site Characteristics.

C_Metr_113: It is recognised that the potential risks associated with metrication should be adequately understood and assessed in order that the proposed design solution can be tested and amended if necessary. A process to assess potential metrication risks for the UK deployment of the SMR-300 has therefore been developed and piloted. This process has identified that the assessment of potential metrication risks is expected to require consideration of plant-wide consistency, consideration of individual system / plant areas and how metrication risks could impact safety case claims. A Commitment is therefore raised to ensure metrication risks are clearly understood and considered, both at a plant-wide and system / plant area level, such that risks are minimised SFAIRP beyond GDA. Target for Resolution - Issue of Pre-Construction SSEC.

GDA Commitments have been formally captured in the Commitments, Assumptions and Requirements process [5]. Further details of this process are provided in Part A Chapter 4 [3].

2.9.4 Conclusion

The conclusion of this chapter of the PSR is that:

- The chapter claims have been met with evidence appropriate to the GDA stage.
- The SMR-300 design is based on proven technology as far as is reasonably practicable to reduce first-of-a-kind engineering to minimise technology development and licensing risks. The SMR-300 design draws on the operating experience and lessons learnt from six decades of operating nuclear power plants, resulting in a greatly simplified plant with respect to construction, operation, inspection, and maintenance as compared to Gen-II and Gen-III LWRs.
- The Holtec safety, environmental, radiation protection, safety analysis (including safety goals), performance, constructability and decommissioning philosophies have been outlined within this chapter that have derived the SMR-300 design. These philosophies satisfy US NRC requirements and are consistent with wider international guidance and best practice.
- The codes and standards used in the design of the SMR-300 have been selected to meet the stringent requirements of the US NRC. These mature and established US codes and standards are internationally recognised, and commensurate with the importance of the safety functions being delivered. Many of them represent good practice adopted on UK nuclear licensed sites and / or application in earlier successful GDAs.
- A comprehensive assessment has been undertaken across the SSEC where it has been shown in general terms, that the generic SMR-300 design will be capable of being demonstrated against UK RGP codes and standards.

- The design and safety principles being used to develop the generic SMR-300 have been shown to be broadly in alignment with relevant UK context expectations within the ONR SAPs [65] and SyAPs [77] and ONMACS [78].
- [REDACTED]
- A generic site has been defined by the GB GSE for the purposes of this GDA.

Overall development of Part A Chapter 2 is considered appropriate for a PSR.

2.10 REFERENCES

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

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

[5] Holtec Britain, "HPP-3295-0013, Holtec SMR-300 Generic Design Assessment Capturing and Managing commitments, Assumptions and Requirements," Revision 1, January 2025.

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

[10] Holtec Britain, "HI-2240344, Holtec SMR GDA PSR Part B Chapter 13 Radioactive Waste Management," Revision 1, July 2025.

[11] Holtec Britain, "HI-2240350, Holtec SMR GDA PSR Part B Chapter 21 External Hazards," Revision 1, July 2025.

[12] Holtec International, "HPP-160-3037, Design Evolution and Freeze Process for SMR 300," Revision 2, February 2024.

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2.11 LIST OF APPENDICES

Appendix A Definitions and Abbreviations A-1

Appendix B SMR-300 Plant Breakdown Structure B-1

Appendix C Generic SMR-300 Mapping of SSCs to PSR Chapters C-1

Appendix D Chapter A2 CAE Route Map D-1

Appendix A Definitions and Abbreviations

Table 23: Definitions and Abbreviations used Across the SSEC

Acronym / Abbreviation	Definition	Acronym / Abbreviation	Definition
1LD	1-Line Diagrams	LPT	Low Profile Transporter
A / D	Analogue-to-Digital	LPZ	Low / Limited Population Zone
ABWR	Advanced Boiling Water Reactor	LR	Large Release
AC	Alternating Current	LRF	Large Release Frequency
ACC	Air-Cooled Condensers	LRW	Liquid Radioactive Waste System
ACI	American Concrete Institute	LSP	Low Support Plate
ACOP	Approved Code of Practice	LTOP	Low Temperature Overpressure Protection
ACR	Audible Count Rate	LTR	Licensing Topical Reports
ADS	Automatic Depressurisation System	LVE	Low Voltage Alternating Current Distribution System
ADV	Atmospheric Dump Valve	LWA	Limited Work Authorisation
AFoE	Annual Frequency of Exceedance	LWR	Light Water Reactor
AHJ	Authority Having Jurisdiction	LWR	Liquid Radwaste System
AIA	Aircraft Impact Assessment	MAAP	Modular Accident Analysis Programme
AIC	Silver-indium-cadmium	MAC	Multi-division Algorithm Conversion
AIM	Analogue Input Modules	MACE	Material Accountancy and Control Expectations
AIP	Agreement in Principle	MB	Maintenance Bypass
AISC	American Institute of Steel Construction	MB-LOCA	Medium Break Loss of Coolant Accident
ALARA	As Low As Reasonably Achievable	MCC	Motor Control Centre
ALARP	As Low As Reasonably Practicable	MCE	Maximum Credible Event
ALWR	Advanced Light Water Reactor	MCH	Main Control Room Habitability System
AM	Accident Management	MCNP	Monte Carlo N-Particle Transport
AMP	Accident Management Program	MCR	Main Control Room
ANS	American Nuclear Society	MCS	Minimal Cut Sets
ANSI	American National Standards Institute	MDNBR	Minimum Departure from Nucleate Boiling Ratio
ANT	Advanced Nuclear Technology	MDSL	Master Document Submission List
AOA	Axial Offset Anomaly	MELCO	Mitsubishi Electric Corporation
AOM	Analogue Output Modules	MELTAC	Mitsubishi Electric Total Advanced Control (I&C Plant Controls Technology)
AOO	Anticipated Operational Occurrence	MEWP	Mobile Elevated Working Platforms

Acronym / Abbreviation	Definition	Acronym / Abbreviation	Definition
AOR	Aligned Once Resolved	MFCV	Main Feedwater Control Valve
AOV	Air-Operated Valve	MFICV	Main Feedwater Isolation Check Valve
AP	Additional Protocol	MFIV	Main Feedwater Isolation Valve
AP1000	Westinghouse Advanced Pressurised 1000 MW Reactor	MFS	Main Feedwater System
API	American Petroleum Institute	MG	Motor Generator
AR	Annular Reservoir	MHA	Maximum Hypothetical Accident
AR5	Fifth Assessment Report	MIC	Memory Integrity Checks
ARP	Alarm Response Procedure	MIS	Manual Initiation Switch
ASAMPSA	Advanced Safety Assessment Methodologies: Extended PSA	MLD	Master Logic Diagram
ASCE	American Society of Civil Engineers	M-MIS	Man-Machine Interface System
ASEP	Accident Sequence Evaluation Programme	MML	Mott MacDonald Limited
ASHRAE	The American Society of Heating, Refrigerating and Air-Conditioning Engineers	MoM	Minutes of Meeting
ASME	American Society of Mechanical Engineers	MoP	Member of the Public
ASNT	The American Society for Non-destructive Testing	MoU	Memorandum of Understanding
		MOV	Motor Operated Valve
ASPM	Applications Software Programme Manual	MPB	Main Power Block
ASTM	American Society for Testing and Materials	MPC	Multi-Purpose Canister
ATWS	Anticipated Transient Without Scram	MRSR	Maximum Recirculating Steaming Rate
AV	Assurance of Validity of data and models	MS	Leadership and Management for Safety
A-VDU	Alarm Visual Display Unit	MSIV	Main Steam Isolation Valve
A-VDU-P	Alarm Visual Display Unit Processor	MSLB	Main Steam Line Break
AVL	Approved Vendors List	MSQA	Management of Safety and Quality Assurance
AVT	All Volatile Treatment	MSR	Moisture Separator Reheater
AWG	American Wire Gauge	MSS	Main Steam System
AWS	Automatic Welding System	MSSV	Main Steam System Valve
AWWA	American Water Works Association	MSU	Main Step-Up Transformer
AXS	Auxiliary Steam System	MSV	Mean Square Voltage
B-10	Boron-10	mSv	Milli Sievert
BAP	Breathing Air and Pressurisation System	MTC	Moderator Temperature Coefficient
BAST	Boric Acid Supply Tank	MTS	Main Turbine System

Acronym / Abbreviation	Definition	Acronym / Abbreviation	Definition
BAT	Best Available Techniques	MVE	Medium Voltage Alternating Current Distribution System
		MW	Megawatt
BAT C	Best Available Techniques Conclusions	MWe	Megawatts Electric
BB	Balfour Beatty	MXS	Manual Transfer Switch
BDA	Beyond Design Accidents	N2S	Nitrogen Supply System
BDB	Beyond Design Basis	NCSC	National Cyber Security Centre
BDBA	Beyond Design Basis Accident	NDA	Nuclear Decommissioning Authority
BDBE	Beyond Design Basis Event	NDAWG	National Dose Assessment Working Group
BE	Best Estimate	NDE	Non-Destructive Examination
BF_3	Boron Trifluoride	NDT	Non-Destructive Testing
BFPL	Brittle Fracture Prevention Limit	NEA	Nuclear Energy Agency
BIM	Building Information Modelling	NEC	National Electrical Code
BMP	Best Management Practices	NEI	Nuclear Energy Institute
BMS	Business Management System	NEMA	National Electrical Manufacturers Association
BOC	Beginning of Cycle	NESO	National Energy System Operator
BOP	Balance of Plant	NETS	National Electricity Transmission System
BPRA	Burnable Poison Rod Assembly	NFA	New Fuel Assemblies
BPVC	Boiler and Pressure Vessel Code	NFPA	National Fire Protection Association
BREF	Best Available Techniques Reference Document	NFSR	New Fuel Storage Rack
BS	British Standards	NFV	New Fuel Vault
BSI	British Standards Institute	NFW	Non-Fuel Waste
BSL	Basic Safety Limit (Legal Limit)	NFWC	Non-Fuel Waste Canister
BSL	Basic Safety Level	NFWS	Non-Fuel Waste Storage
BSO	Basic Safety Objective	NI	Nuclear Island
BSPM	Basic Software Program Manual	NIA	Nuclear Installations Act
BSS	Basic Safety Standard	NISCI	Nuclear Industry Safety Culture Inventory
BSSD	Basic Safety Standards Directive	NISS	Nitrogen Supply System
BTC	Basic Technical Characteristics	NLR	Nuclear Liabilities Regulation
BTP	Branch Technical Position	NM	Nuclear Material
BWR	Boiling Water Reactor	NMACS	Nuclear Material Accountancy and Control Expectations
C&S	Codes and Standards	NO	Normal Operation
CAE	Claims, Arguments, and Evidence	NOAA	National Oceanic and Atmospheric Administration

Acronym / Abbreviation	Definition	Acronym / Abbreviation	Definition
CAFTA	Computer Aided Fault Tree Analysis	NODA	Normal Operations Dose Assessment
CAI	Instrument and Service Air System	Non-LOCA	Non-Loss of Cooling Accident
CAR	Commitments, Assumptions and Requirements	NOP	Normal Operating Pressure
CAS	Condenser Vacuum System	NOT	Normal Operating Temperature
CB	Control Building	NPP	Nuclear Power Plant
CBA	Cost Benefit Analysis	NPSA	National Protective Security Authority
CBP	Computer-Based Procedure	NPSH	Net Positive Suction Head
CBV	Containment Ventilation System	NQA – 1	Nuclear Quality Assurance
CCA	Contamination Controlled Area	NR	No Release
CCDF	Conditional Core Damage Frequency	NRC	Nuclear Regulatory Commission
CCF	Common Cause Failure	NRPB	Nuclear Radiological Protection Board
CCFP	Containment Conditional Failure Probability	NRV	Non-Radiologically Controlled Area Heating, Ventilation, and Air Conditioning System
CC-I	Capability Category level	NRW	Natural Resource Wales
CCP	Component Control Processors	NS	Non-Seismic
CCW	Component Cooling Water System	NSA	Neutron Source Assemblies
CD	Core Damage	NSD	Near-Surface Disposal
CDF	Core Damage Frequency	N-SFR	Non-Safety Functional Requirement
CDM	Construction, Design and Management	NSHS	Nuclear Site Health and Safety
CDM 2015	Construction, Design and Management Regulations 2015	NSL	Nuclear Site Licence
CDS	Control Rod Drive System	NSR	Nuclear Safeguards Regulations
CE	Conformité Européenne	NSR19	The Nuclear Safeguards Regulations 2019
CED	Committed Effective Dose	NSSP	Nuclear Site Security Plan
CES	Containment Enclosure Structure	NSSS	Nuclear Steam Supply System
CET	Containment Event Tree	NT	Numerical Target
CF	Capable Faulting	NTS	Nuclear Transportation Services
CFL	Consolidated Fault Listing	NUREG	United States Nuclear Regulatory Commission Technical Report Designation
CFR	Code of Federal Regulations	NWS	Nuclear Waste Services
CFS	Chemical Feed System	O&M	Operation and Maintenance
CFSS	Concrete Filled Steel Structures	OB	Operating Bypasses

Acronym / Abbreviation	Definition	Acronym / Abbreviation	Definition
CGC	Combustible Gas Control	OBE	Operating Basis Earthquake
CGC	Combustible Gas Control System	OC	Operator Console
CGD	Commercial Grade Dedication	ODS	Ozone-Depleting Substance
CGN	China General Nuclear	OECD	Organisation for Economic Co-operation and Development
CHF	Critical Heat Flux	OER	Operational Experience Review
CI	Conventional Island	OFGEM	Office of Gas and Electricity Markets
CID	Criticality incident Detection	OJT	On-the-Job Training
CIDI	Central Index of Dose Information	OLA	Operating License Application
CIGRE	International Council on Large Electric Systems	OLC	Operating Limits and Conditions
CILC	Crud Induced Localised Corrosion	ONMACS	ONR Nuclear Material Accountancy, Control and Safeguards Assessment Principles
CIS	Containment Isolation System	ONR	Office For Nuclear Regulation
CIV	Containment Isolation Valve	OOS	Out of Scope
CLDP	Contaminated Land and Groundwater	OPEX	Operating Experience
CLP	European Regulation (EC) No 1272/2008 on classification, labelling and packaging of substances and mixtures	OR	Operating Rule
CLRF	Conditional Large Release Frequency	ORE	Operational Radiation Exposure
CLSM	Controlled Low Strength Material	ORM	Other Radioactive Material
CME	Control Mass Ejections	OSD	Operational Sequence Diagram
CMF	Common Mode Failure	OSE	Off-Site Emergency
CMI	Coronal Mass Injections	OSHA	Occupational Safety and Health Administration
CMVC	Configuration Management and Version Control	OSW	On Site Worker
CNS	Condensate System	OTS	Operating Technical Specification
CNSC	Canadian Nuclear Safety Commission	OTSG	Once-Through Steam Generator
COBRA-TF	Coolant Boiling in Rod Arrays - Two Fluid	O-VDU	Operational Visual Display Unit
COL	Combined License	P&ID	Piping and Instrumentation Diagrams
COMAH	Control of Major Accident Hazards	PAM	Post Accident Monitoring System
COMAH 2015	Control of Major Accident Hazards Regulation 2015	PAR	Passive Autocatalytic Recombiner
COMS	Constructability, Operability, Maintainability and Safety	PARI	Purdue Applied Research Institute

Acronym / Abbreviation	Definition	Acronym / Abbreviation	Definition
Con Ops	Concept of Operations	PAT	Power Ascension Testing
COP	Code of Practice	PBS	Plant Breakdown Structure
COTS	Commercial-Off-The-Shelf	PC	Personal Computer
CPA	Construction Permit Application	PCC	Passive Core Cooling System
CPG	Containment Performance Goal	PCER	Pre-Construction Environmental Report
CPO	Condensate Polisher System	PCH	Passive Containment Heat Removal System
CPPNM	Convention on the Physical Protection of Nuclear Material	PCM	Passive Core Makeup Water System
CPS	Counts Per Second	PCMWT	Passive Core Makeup Water Tank
CPU	Central Processing Unit	PCS	Plant Control System
CRA	Control Rod Assembly	PCSR	Pre-Construction Safety Report
CRC	Control Rod Control System	PC-SS	Pre-Construction Site Specific
CRDM	Control Rod Drive Mechanism	PC-SS-SSEC	Pre-Construction Site Specific Safety, Security and Environment Case
CREZ	Control Room Emergency Zone	PDA	Personal Digital Assistant
CRS	Chemical and Radiochemical Specification	PDC	Prospective Design Change
CRV	Control Room Normal Ventilation System	PDH	Primary Decay Heat Removal System
CS	Containment Structure	PDS	Plant Damage States
CSD	Control System Description	PE	Product Excellence
CSH	Overhead Heavy Load Handling System	PER	Preliminary Environmental Report
CSNI	Committee on the Safety of Nuclear Installations	PESR	Preliminary Environmental Safety Report
CSR	Component Safety Report	PFA	Probability of Failure per Annum
CSRA	Cyber Security Risk Assessment	PFD	Probability of Failure on Demand
CVC	Chemical & Volume Control System	PFS	Preliminary Fault Schedule
CWS	Chilled Water System	PFY	Failure per Year
DA	Design Authority	PGA	Peak Ground Acceleration
DAC	Design Adaptation Committee	PH	Personnel Hatch
DAS	Diverse Actuation System	PIE	Postulated Initiating Event
DB	Design Basis	PIF	Planar Intervessel Forging
DBA	Design Basis Accident	PIM	Power Interface Modules
DBAA	Design Basis Accident Analysis	PIO	Process Input and Output
DBC	Design Basis Condition	PLSF	Plant Level Safety Functions
DBE	Design Basis Event	PMO	Project Management Office
DBT	Design Basis Threat	PMP	Project Management Plan
DC	Direct Current	POCO	Post Operation Clean-Out

Acronym / Abbreviation	Definition	Acronym / Abbreviation	Definition
DCC	Document Control Team	POL	Problem Oriented Language
DCE	Direct Current Power Distribution System	PORV	Power Operated Relief Valve
DCP	Design Control Process	POS	Plant Operating State
DDF	Depth-Duration-Frequency	PP	Position Paper
DDP	Digital Delivery Plan	PPE	Personal Protective Equipment
DDT	Deflagration to Detonation Transition	PPL	Panelboard
DEC	Design Extension Condition	PQP	Project Quality Plan
DECC	Department of Energy & Climate Change	PR	Power Range
DEDP	Decommissioning Developed Principles	PRA	Probabilistic Risk Assessment
DEFRA	Department for Environment, Food and Rural Affairs	Pry	Per reactor year
DESNZ	Department for Energy Security and Net Zero	PS	Purchase Specification
DFC	Damaged Fuel Container	PSA	Probabilistic Safety Assessment
DfS	Design for Seismic	PSAR	Preliminary Safety Assessment Report
D-HIS	Diverse Actuation System Human System Interface	PSC	Parameter Setting Computer
DHRS	Decay Heat Removal System	PSgR	Preliminary Safeguards Report
DI	Digital Input	PSI	Pre-Service Inspection
DiD	Defence in Depth	PSL	Primary Sampling System
DIM	Digital Input Modules	PSR	Preliminary Safety Report
DL	Document List	PSS	Plant Safety System
DNBR	Departure from Nucleate Boiling Ratio	PSV	Pressuriser Safety Valve
DO	Digital Output	PSyR	Preliminary Security Report
DOE	Department of Energy	PT	Periodic Test
DOM	Digital Output Modules	PUWER	Provision and Use of Work Equipment Regulations
DPM	Decades per Minute	P-VDU	Procedural Visual Display Unit
DPP	Diverse Protection Processor	P-VDU-P	Procedural Visual Display Unit Processor
DPUC	Dose Per Unit Calculation	PWHT	Post Weld Heat Treatment
DR	Design Reference	PWR	Pressurised Water Reactor
DRC	Design Decision and Risk Review Committee	PWSCC	Primary Water Stress Corrosion Cracking
DRP	Design Reference Point	PZR	Pressuriser
DS	Design Specification	QA	Quality Assurance
DSA	Deterministic Safety Analysis	QAM	Quality Assurance Manual
DSEAR	Dangerous Substances and Explosive Atmospheres Regulations	QAP	Quality Assurance Programme
DSM	Defect Size Margin	QAP TR	Topical Report on the Quality Assurance Program

Acronym / Abbreviation	Definition	Acronym / Abbreviation	Definition
DSMP	Design Safety Management Plan	QC	Quality Control
DSRL	Dounreay Site Restoration Limited	QEDL	Quantification of Effluent Discharges and Limits
DSS	Desktop Scale Simulator	QEDS	Qualified Examination Defect Size
DSW	Dry Solid Wastes	QMS	Quality Management System
DTA	Defect Tolerance Assessment	QNM	Qualifying Nuclear Material
D-VDU	Diverse Actuation System Visual Display Unit	R&D	Research & Development
D-VDU-P	Diverse Actuation System Visual Display Unit Processor	Ra	Roughness Average
DVI	Direct Vessel Injection	RAB	Reactor Auxiliary Building
DWMP	Decommissioning Waste Management Plan	RADD	Radiation Discharges Database
DWS	Demineralised Water System (or Demineralised Water Transfer System)	RADTRAD	Radionuclide Transport, Remove and Dose
DWT	Demineralised Water Treatment System	RAW	Risk Achievement Worth
E / E / PE	Electrical, Electronic and Programmable Electronic	RBH	Reactor Auxiliary Building Truck Bay Crane
EA	Environmental Agency	RBV	Radioactive Waste Building Heating, Ventilation and Air Conditioning System
EAB	Exclusion Area Boundary	RC	Reinforced Concrete
EAD	Aging and degradation	RCA	Radiologically Controlled Area
EBA	Enriched Boric Acid	RCCA	Rod Cluster Control Assembly
EC&I	Electrical Control & Instrumentation	RCDT	Reactor Coolant Drain Tank
ECCS	Emergency Core Cooling System	RCMG	Rod Control Motor Generator
ECE	Civil engineering	RCO	Reactor Core
ECH	Chemistry	RCP	Reactor Coolant Pump
ECM	Commissioning	RCPB	Reactor Coolant Pressure Boundary
ECR	Criticality safety	RCS	Reactor Coolant System
ECS	Safety Classifications and Standards	RCT	Risk Contingency Tool
ECV	Containment and Ventilation	RCV	Radiologically Controlled Area Heating, Ventilation and Air Conditioning System
EDP	Engineering Design Principle	RD	Responsible Designer
EDR	Design for Reliability	RDS	Radioactive Drain System
EE	Expected Events	REA	Register of Environmental Aspects
EES	Essential services	REGDOC	Regulatory Document
EF	Enhanced Fujita	REPPIR19	The Radiation (Emergency Preparedness and Public Information) Regulations 2019

Acronym / Abbreviation	Definition	Acronym / Abbreviation	Definition
EGDP	Engineering Generic Derived Principle	RFI	Radio Frequency Interference
EGR	Graphite Reactor Core	RG	Regulatory Guide
EH	Equipment Hatch	RGP	Relevant Good Practice
EH&S	Environmental, Health and Safety	RHR	Residual Heat Removal System
EHA	Engineering Principles: External and Internal Hazards	RHWG	Reactor Harmonisation Working Group
EHT	Heat Transport Systems	RI	Regulatory Issue
EIADR	The Nuclear Reactors (Environmental Impact Assessment for Decommissioning) Regulations	RIA	Radiological Impact Assessment
EIDR	Environmental Impact Assessment for Decommissioning Regulations	RIO	Regulatory Interface Office
EIMT	Examination, Inspection, Maintenance and Testing	RIS	Reactor Internal Structure
EIS	Ex-core (nuclear) Instrumentation System	RL	Strategies for radioactively contaminated land
EKP	Engineering Key Principle	RM	Radioactive Material
ELLDS	End-of-life Limiting Defect Size	RMI	Reflective Metal Insulation
ELO	Engineering Layout and Configuration Principles	RMS	Radiation Monitoring System
ELOG	Extended Loss of Grid	RO	Regulatory Observation
ELOOP	Extended Loss of Offsite Power	ROM	Read Only Memory
EM	Evaluation Model	RP	Requesting Party
EMC	Electromagnetic Compatibility	RP	Radiation Protection
EMDAP	Evaluation Model Development and Assessment Process	RPDP	Radiological Protection Developed Principles
EMI	Electromagnetic Interference	RPDS	Radiation Protection Design Standard
EMP	Environment Management Plan	RPE	Respiratory Protective Equipment
EMPR	Electromagnetic Pulse Resistance	RPI	Rod Position Indication
EMS	Environmental Management System	RPP	Reactor Protection Processor
EMT	Maintenance, Inspection, and Testing	RPV	Reactor Pressure Vessel
ENC	Integrity of non-metal Components and Structures	RQ	Regulatory Query
ENDP	Engineering Design Principle	RRM	Risk Reduction Measure
ENIQ	European Network for Inspection and Qualification	RSC	Robust Shielded Container
ENM	Control of Nuclear Matter	RSF	Remote Shutdown Facility
ENSREG	European Nuclear Safety Regulators Group	RSG	Recirculating Steam Generators

Acronym / Abbreviation	Definition	Acronym / Abbreviation	Definition
EO	Equipment Operator	RSMDP	Radioactive Substances Management Developed Principle
EOC	End of Cycle	RSMGDP	Radioactive Substances Management Generic Derived Principle
EOF	Emergency Operations Facility	RSR	Radioactive Substances Regulation
EOP	Emergency Operating Procedure	RSR:GDP	Radioactive Substances Regulation: Generic Developed Principles
EP	Emergency Preparedness	RT	Reactor Trip
EPE	Chemical (Process) Engineering	RTB	Reactor Trip Breaker
EPF	Environment Protection Functions	RTD	Resistance Temperature Detectors
EPM	Environment Protection Measures	RWB	Radioactive Waste Building
EPR	European Pressurised Reactor	RWM	Radioactive Waste Management
EPR16	Environmental Permitting (England and Wales) Regulations 2016	RWMA	Radioactive Waste Management Arrangements
EPRI	Electric Power Research Institute	RWMC	Radioactive Waste Management Case
EPS	Pressure systems	RWMD	Radioactive Waste Management Directorate
EPZ	Emergency Planning Zone	RWST	Refuelling Water Storage Tank
EQU	Equipment Qualifications	RX	Reactor System
ER	Environmental Report	S&Q	Staffing and Qualification
ERA	Environmental Risk Assessment	SA	Severe Accident
ERC	Reactor core	SAA	Severe Accident Analysis
ERCA	External Radiation Controlled Area	SAFDLs	Specified Acceptable Fuel Design Limits
ERIC	Eliminate, Reduce, Isolate, Control	SAH	Safety Assessment Handbook
ERICA	Environmental Risk from Ionising Contaminants: Assessment and Management	SAM	Severe Accident Management
ERICPD	Eliminate, Reduce, Isolate, Control, Personal Protective Equipment, Discipline	SAMA	Severe Accident Mitigation Alternatives
ERL	Reliability Claims	SAMG	Severe Accident Management Guideline
ERP	Emergency Response Facilities	SAP	Safety Assessment Principles
ESF	Engineered Safety Feature	SAR	Safety Analysis Report
ESFA	Engineered Safety Features Actuation System	SB	Small Break
ESR	Control and Instrumentation of Safety-related Systems	SBD	Standby Diesel Generator
ESS	Extraction Steam System	SbD	Secure by Design

Acronym / Abbreviation	Definition	Acronym / Abbreviation	Definition
ETA	Ethanolamine	SBLOCA	Small Break Loss of Coolant Accident
ETI	Energy Technologies Institute	SBO	Station Black Out
ETS	Environmental Technical Specification	SC	Steel-Concrete
EU	European Union	SC1	Safety Class 1
EUR	European Utility Requirements	SCC	Stress Corrosion Cracking
FA	Function Allocation	SCV	Steel Containment Vessel
FAC	Flow Accelerated Corrosion	SDCV	Spatially Dedicated Continuously Visible
FACP	Fire Alarm Control Panel	SDD	System Design Description
FAD	Failure Assessment Diagram	SDG	Standby Diesel Generator System
FAP	Forward Action Plan	SDH	Secondary Heat Removal System
FArP	Funding Arrangements Plan	SDRS	Seismic Design Response Spectra
FCM	Fuel Centreline Melt	SEI	Structural Engineering Institute
FDD	Flow Distribution Device	SEL	Seismic Equipment List
FDP	Funded Decommissioning Programme	SEP	Solar Energetic Particles
FE	Finite Element	SEPA	Scottish Environmental Protection Agency
FEA	Finite Element Analysis	SER	Safety Evaluation Report
FEH	Flood Estimation Handbook	SEWS	Screening Evaluation Work Sheet
FF	Failure Frequencies	SF	Safety Function
F-gases	Fluorinated Greenhouse Gases	SFA	Spent Fuel Assemblies
FHA	Fuel Handling Area	SFAIRP	So Far As is Reasonably Practicable
FHBC	Fuel Handling Bridge Crane	SFC	Spent Fuel Cooling System
FHD	Forced Helium Dehydration System	SFIS	Spent Fuel Interim Storage
FID	Final Investment Decision	SFP	Spent Fuel Pool
FLEX	Flexible Coping Strategies	SFR	Safety Functional Requirement
FMEA	Failure Modes and Effects Analysis	SFSR	Spent Fuel Storage Rack
FNEF	Future Nuclear Enabling Fund	SFV	Security Facilities Heating, Ventilation and Air Conditioning System
FOAK	First-of-a-Kind	SGB	Steam Generator Blowdown
FP	Fundamental Principle	SgbD	Safeguards by Design
FPGA	Fuel Programmable Gate Array	SgC	Safeguards Claim
FPS	Fire Protection System	SGE	Steam Generator
FRA	Functional Requirements Analysis	SGTR	Steam Generator Tube Rupture

Acronym / Abbreviation	Definition	Acronym / Abbreviation	Definition
FS	Fault Studies	SHARP	Systematic Human Action Reliability Procedure
FSAR	Fire Safety Analysis Report	SI	Structural Integrity
FSE	Fundamental Safeguards Expectation	SKI	Swedish Nuclear Inspectorate
FSF	Fundamental Safety Function	SL	Submission List
FSyP	Fundamental Security Principle	SLB	Steam Line Break
FUE	Fuel	SLD	Single Line Diagrams
FV	Fussell-Vesely	SLOCA	Small Break Loss of Coolant Accident
FW	Feedwater	SMA	Seismic Margin Assessment
FWH	Feedwater Heater	SMACNA	Sheet Metal and Air Conditioning Contractors National Association Inc.
FWLB	Feedwater Line Break	SME	Subject Matter Expert
GA	General Arrangement	SMEq	Seismic Margin Earthquake
GALL-SLR	Generic Aging Lessons Learned for Subsequent Licence Renewal	SMR	Small Modular Reactor
GB	Great Britain	SMS	Safety Management System
GCC	Grid Code Compliance	SoDA	Statement of Design Acceptability
GDA	Generic Design Assessment	SOE	Safe Operating Envelope
GDC	Generic Design Criteria	SONGS	San Onofre Nuclear Generating Site
GDF	Geological Disposal Facility	SOP	Standard Operating Procedure
GIC	Geomagnetically Induced Current	SOV	Solenoid Operated Valve
GLE	Ground Level Events	SPAR-H	Standardised Plant Analysis Risk-Human
GLP	Grounding and Lightning Protection	SPC	Steel Plate Composite
GLSEA	Great Lake Surface Environmental Analysis	SPDS	Safety Parameters Display System
GNSL	General Nuclear System Ltd	SPM	Software Program Manual
GRW	Gaseous Radioactive Waste System	SPND	Self-Powered Neutron Detector
GSE	Generic Site Envelope	SQ	Significant Quantity
GSER	Generic Site Envelope Report	SQEP	Suitably Qualified and Experienced Personnel
GSG	General Safety Guidance	SR	Source Range
GSR	Generic Security Report	SRES	Special Report on Emission Scenarios
GSS	Gland Seal System	SRF	Small Release Frequency
GTCC	Greater Than Class C	SRO	Senior Reactor Operator
GUI	Graphical User Interface	SRP	Standard Review Plan
GW	Giga Watt	SRS	System Requirements Specifications
H&S	Health & Safety	SRSS	Square Route of the Sum of the Squares

Acronym / Abbreviation	Definition	Acronym / Abbreviation	Definition
HA	Human Action	SRV	Steam Relief Valve
HARW	Higher Activity Radioactive Waste	SRW	Solid Radwaste System
HASAWA	Health and Safety at Work Act	SSA	Secondary Source Assembly
HAW	Higher Activity Waste	SSC	Structures, Systems, and Components
HAZID	Hazard Identification	SSE	Safe Shutdown Earthquake
HAZOP	Hazard and Operability	SSEC	Safety, Security and Environmental Case
HB	Holtec Britain	SSER	Safety, Security, and Environment Report
HBSC	Human-Based Safety Claim	SSG	Specific Safety Guide
HCLPF	High Confidence of Low Probability Failure	SSI	Soil Structure Interaction
HCMC	Human System Interface Configuration Management Computers	SSR	Specific Safety Requirements
HDEC	Hyundai Engineering and Construction	SSS	Secondary Sampling System
HDL	Hardware Description Language	SST	Station Service Transformer
HDS	Heater Drain System	ST	Siting
HEAF	High Energy Arcing Fault	STA	Shift Technical Adviser
HECA	Hazard Evaluation and Consequence Assessment	STE	Steam Tunnel
HEP	Human Error Probability	STPA	Systems Theoretic Process Analysis
HEPA	High Efficiency Particulate Air	STS	Standard Technical Specifications
HEQ	Human Error Quantification	SUR	Start-Up Rate
HF	Human Factors	S-VDU	Safety Visual Display Unit
HFE	Human Factors Engineering	SVDU-P	Safety Visual Display Unit Processor
HFI	Human Factors Integration	SWESC	Site-Wide Environmental Safety Case
HFIAR	Human Factors Issues and Assumptions Register	SWGR	Switchgear
HFIP	Human Factors Integration Plan	SWS	Service Water System
HFLC	High-Frequency, Low-Consequence	SyAP	Security Assessment Principles
HFT	Hot Functional Testing	SyC	Security Claim
HGV	Heavy Goods Vehicle	T&M	Testing and Maintenance
HHGW	High Heat Generating Waste	TA	Task Analysis
HI	Holtec International	TAG	Technical Assessment Guide
HI-C	High-Integrity Container	TAGSI	Technical Advisory Group on the Structural Integrity
HITS	Human Factors Engineering Issue Tracking System	Tavg	Average Temperature
HLSF	High-Level Safety Function	TB	Turbine Building
HLW	High Level Waste	TBC	To Be Confirmed

Acronym / Abbreviation	Definition	Acronym / Abbreviation	Definition
HMI	Human Machine Interface	TBD	To Be Determined
HMS	Hydrogen Monitoring System	TBS	Turbine Bypass System
HOC	Hierarchy of Controls	TBV	Turbine Bypass Valve
HOCM	Hierarchy of Controls Measure	Tcold	Cold Leg Temperature
HP	High Pressure	TEDE	Total Effective Dose Equivalent
HPA	Health Protection Agency	TEMA	Tubular Exchanger Manufacturers Association
HPLV	Human Performance Limiting Value	TEP	Topic Engagement Plan
HPME	High Pressure Melt Injection	TETRA	Terrestrial Trunked Radio
HPP	Holtec Project Procedure	TGV	Turbine Governor Valve
HPR1000	Hua-Long Pressurised Reactor 1000	THERP	Technique for Human Error Rate Prediction
HPSC	Human System Interface Parameter Setting Computer	Thot	Hot Leg Temperature
HQP	Holtec Quality Procedures	TI	Turbine Island
HR	High Reliability	TIG	Technical Inspection Guide
HRA	Human Reliability Assessment	TIHA	Treatment of Important Human Action
HRAP	Human Reliability Analysis Procedure	TJ	Technical Justification
HSE	Health & Safety Executive	ToR	Tolerability of Risk
HSG	Health and Safety Guidance	TR	Topical Report
HSI	Human System Interface	TRL	Technology Readiness Level
HSP	Holtec Standard Procedures	TSC	Technical Support Centre
HUT	Holdup Tank	TSV	Turbine Stop Valve
HVAC	Heating, Ventilation, and Air Conditioning	TT	Turbine Trip
HX	Heat Exchanger	TTLOOP	Turbine Trip coincident with a Loss of Offsite Power
I&C	Instrumentation and Control	UAT	Unit Auxiliary Transformer
I/O	Input and Output	UCP	Upper Core Plate
IA	Instrument Assembly	UHS	Ultimate Heat Sink
IAEA	International Atomic Energy Agency	UK	United Kingdom
IASCC	Irradiation-Assisted Stress Corrosion Cracking	UKCA	United Kingdom Conformity Assessed
IB	Intermediate Building	UKCP	United Kingdom Climate Projections
IC	Intelligent Customer	UKCP09	United Kingdom Climate Projections 2009
ICBM	Independent Confidence Building Measures	UKCP18	United Kingdom Climate Projections 2018
ICE	Instrumentation and Control Power Distribution System	UMAX	Underground Maximum Capacity System
		UN	United Nations
ICIA	In-Core Instrument Assemblies	UPS	Uninterruptible Power Supply
		UR	Utilisation Ratio

Acronym / Abbreviation	Definition	Acronym / Abbreviation	Definition
ICRP	International Commission on Radiological Protection	URD	Utility Requirements Document
IDHEAS	Integrated Human Event Analysis System	US	United States
IE	Initiating Event	USNRC	United States Nuclear Regulatory Commission
IEC	International Electrotechnical Commission	USP	Upper Support Plate
IED	Industrial Emissions Directive	V&V	Verification and Validation
IEEE	Institute of Electrical and Electronic Engineers	VAC	Volts Alternating Current
IEF	Initiating Event Frequency	VAI&C	Vital Area Identification and Categorisation
IGSCC	Intergranular Stress Corrosion Cracking	VCT	Vertical Cask Transporter
IHA	Important Human Actions	VDR	Vendor Design Review
IIG	Inter-Industry Guidance	VDC	Volts Direct Current
IIS	In-core Instrumentation System	VDU	Visual Display Unit
ILW	Intermediate Level Waste	VDU-P	Visual Display Unit - Processor
INL	Idaho National Laboratory	VFD	Variable Frequency Drive
INPO	Institute of Nuclear Power Operations	VHR	Very High Reliability
IP	Intellectual Property	VLLW	Very Low Level Waste
IPCC	Intergovernmental Panel on Climate Change	VMS	Vibration Monitoring System
IPEEE	Individual Plant Examination of External Events	VOA	Voluntary Offer Agreement
IPS	Information Processing System	VVM	Vertical Ventilated Module
IR	Intermediate Range	WAC	Waste Acceptance Criteria
		WANO	World Association of Nuclear Operators
IRAT2	Initial Radiological Assessment Tool 2	WBS	Work Breakdown Structure
IRR	Ionising Radiation Regulations	WEEE	Waste Electronic and Electrical Equipment
IRR17	Ionising Radiations Regulations 2017	WENRA	Western European Nuclear Regulators' Association
ISA	International Society of Automation	WSW	Wet Solid Wastes
ISF	Interim Storage Facility		
ISFSI	Interim Spent Fuel Storage Installation		
ISFSI	Independent Spent Fuel Storage Installation (UMAX System)		
ISG	Interim Staff Guidance		
ISI	In-Service Inspection		
ISLOCA	Interfacing System Loss of Coolant Accident		
ISO	International Organisation for Standardisation		

Acronym / Abbreviation	Definition	Acronym / Abbreviation	Definition
ISRA	Information Security Risk Assessment		
ISRS	In-Structure Response Spectra		
ISS	Integrated Security Solution		
ISV	Integrated Systems Validation		
IT	Information Technology		
ITA	Independent Technical Advisor		
IVR	In-Vessel Retention		
IWS	Integrated Waste Strategy		
kV	kilo-Volts		
L1	Level 1		
L2	Level 2		
LA	Licence Applicant		
LAC	Limits and Conditions		
LAN	Local Area Networks		
LBB	Leak Before Break		
LBE	Licensing Basis Events		
LBI	Local Business Instruction		
LBLOCA	Large Break Loss of Coolant Accident		
LC	License Condition		
LCO	Limiting Condition for Operation		
LCS	Local Control Station		
LDA	Large Domestic Appliances		
LDHR	Loss of Decay Heat Removal		
LDP	Large Display Panel		
LDS	Leak Detection System		
LED	Light Emitting Diode		
LEL	Lower Explosive Limit		
LER	Licensee Event Report		
LHGW	Low Heat Generating Waste		
LL	Legal Limit		
LLC	Limited Liability Company		
LLH	Light Load Handling System		
LLHM	Light Load Handling Machine		
LLOCA	Large Loss of Coolant Accident		
LLSF	Lower-Level Safety Functions		
LLW	Low Level Waste		
LLWR	Low Level Waste Repository		
LOC	Letter of Compliance		
LOCA	Loss of Coolant Accident		

Acronym / Abbreviation	Definition	Acronym / Abbreviation	Definition
LOE	Limit of Operating Envelope		
LOFW	Loss of Main Feed Water		
LOLA	Loss of Large Area		
LOLER	Lifting Operations and Lifting Equipment Regulations		
LOOP	Loss Of Offsite Power		
LP	Low Pressure		
LPSD	Low Power and Shutdown		

Appendix B SMR-300 Plant Breakdown Structure

[REDACTED]

Figure 14: SMR Plant Breakdown Structure

Appendix C Generic SMR-300 Mapping of SSCs to PSR Chapters

Table 24: SMR-300 GDA Mapping of SCCs to PSR Chapters

[REDACTED]

Appendix D Chapter A2 CAE Route Map

Table 25: Chapter A2 CAE Route Map

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