HOLTEC BRITAIN

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PSR Part B Chapter 9 Conduct of Operations

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9.1 INTRODUCTION

This chapter of the Preliminary Safety Report (PSR) outlines the approach to plant operations and the undertaking of operations. These are not expected to be outlined in full at the Generic Design Assessment (GDA) stage, as they are developed by the licensee in the site licensing phase as the detailed information about the reactor's operation evolves. However, general approaches, methodologies and principles are outlined for operational management of the Generic Small Modular Reactor (SMR)-300 so that the eventual conduct of operations is integrated into the plant design.

Part B Chapter 9 of the Preliminary Safety Report (PSR) presents the Claims, Arguments and intended Evidence (CAE) for the description of operational aspects and conduct of operations Structures, Systems and Components (SSCs) that underpin the design of the generic SMR-300.

9.1.1 Purpose and Scope

The Overarching Safety, Security and Environmental Case (SSEC) Claims are presented in PSR Part A Chapter 3 Claims, Arguments and Evidence [1] of this PSR.

This chapter (Part B Chapter 9) links to the overarching claim through Claim 2.3:

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

As set out in Part A Chapter 3 [1], Claim 2.3 is further decomposed across several disciplines which support the development of through-life management arrangements. This chapter presents the conduct of operations aspects for the generic SMR-300 and therefore directly supports a claim focused on the appropriate arrangements to safely manage people and plant, Claim 2.3.1.

Claim 2.3.1: Appropriate arrangements to safely manage people and plant during the construction, commissioning and operation of the Generic Holtec SMR-300 are suitably mature for a generic design.

Further discussion on how the Level 3 claim is broken down into Level 4 claims and how the Level 4 claims are met is provided in Subchapter 9.2.

The generic SMR-300 basis of design has been produced in line with United States (US) regulations and takes due cognisance of good practice such as International Atomic Energy Agency (IAEA) guidance and standards. The basis of the design is described further in PSR Part A Chapter 2 General Design Aspects and Site Characteristics [2]. Any potential gaps between the United Kingdom (UK) Office for Nuclear Regulation (ONR) GDA requirements and US regulations will be identified and suitable methodology resolutions identified across the PSR. The methodologies for resolution related to the conduct of operations are outlined within the Forward Actions in sub-chapter 9.10.2.3 and are collated and managed via the process described in Holtec SMR GDA PSR Part A Chapter 4 Lifecycle Management of Safety and Quality Assurance [3].

9.1.2 Exclusions

The following are excluded from the identification of normal operations functions:

- SSCs outside the GDA scope have been excluded (defined in PSR Part A Chapter 2 General Design Aspects and Site Characteristics [2]).

9.1.3 Interfaces with other PSR Chapters

Operational aspects and conducts of operations interface with numerous Chapters of the PSR to varying degrees, notably within maintenance considerations and Human Factors sections, a number of the key chapters are listed below, this is a non-exhaustive list:

PSR Part A Chapter 4 Lifecycle Management of Safety and Quality Assurance [3]. The Reference Design and Reference Plant are stated in Chapter A2 [2]. Chapter A4 [3] will address the lifecycle management of this reference design and its interactions with the reference plant. PSR Part A Chapter 5 Summary of ALARP [4]. Chapter A5 [4] presents the ALARP methodology and ALARP justifications for the generic SMR-300, this provides the basis for the concluding subchapter, which provides an ALARP summary, including a description of the FAPs.

The requirements for Examination, Inspection, Maintenance and Testing (EIMT) will be captured in the following chapters:

- PSR Part B Chapter 1 Reactor Coolant System and Engineered Safety Features [5], Reactor Coolant and Connected SSCs must allow reliable operation and control to maintain safety of the plant.
- PSR Part B Chapter 2 Reactor Fuel and Core [6], Fuel and Core SSCs must allow reliable operation and control to maintain safety of the plant.
- Part B Chapter 4 Control and Instrumentation Systems [7], Instrumentation & Control (I&C) SSCs must provide reliable means for monitoring and control to maintain safety of the plant.
- PSR Part B Chapter 5 Reactor Supporting Facilities [8], Reactor Supporting SSCs must allow reliable operation and control to maintain safety of the plant.
- PSR Part B Chapter 6 Electrical Engineering [9], Electrical SSCs must allow reliable operation and provide power to maintain the safety of the plant.
- PSR Part B Chapter 10 Radiological Protection [10], Radiological Protection SSCs must allow reliable operation and control to maintain normal operational safety of the plant.
- PSR Part B Chapter 11 Environmental Protection [11], Environmental Protection SSCs must allow reliable operation and control to maintain environmental safety and minimise discharges and wastes generated by the plant.
- PSR Part B Chapter 16 Probabilistic Safety Assessment (PSA) [12], PSA can be used to inform the EIMT proof testing intervals of SSC.
- PSR Part B Chapter 18 Structural Integrity [13], the requirements for EIMT of Structural Integrity SSCs will be captured here.
- PSR Part B Chapter 19 Mechanical Engineering [14], the requirements for EIMT of Mechanical Engineering SSCs will be captured here.

- PSR Part B Chapter 20 Civil Engineering [15], the requirements for EIMT of Civil Engineering SSCs will be captured here.

PSR Part B Chapter 12 Control of Non-Radiological Hazards [16], Operations will ensure safety and risk reduction of non-radiological hazards. The safety of operators during maintenance activities will be assessed using principles developed in this chapter. PSR Part B Chapter 13 Radioactive Waste Management [17], Operations for managing the solid, liquid, and gaseous wastes generated by the generic SMR-300 are summarised in this chapter and deemed to be essential to normal operations. PSR Part B Chapter 15 Beyond Design Basis and Severe Accident Analysis, and Emergency Preparedness [18], Emergency Operating Procedure and Severe Accident Management Guidelines are addressed in this chapter and will require input from Chapter B15 [18]. PSR Part B Chapter 17 Human Factors [19], Human Factors will be required to define and input into the operating procedures, staffing and qualification, optimising the design equipment, workspaces, and tasks/working arrangements for human performance, through the application of relevant good practice. It is important to align the concept of operations with the Human Factors (HF) Integration process. PSR Part B Chapter 23 Reactor Chemistry [20], Chemistry associated with Chemical Specifications (e.g. control of the reactor chemistry should allow operations such as determining the concentration of soluble boron in the Reactor Coolant System). PSR Part B Chapter 24 Fuel Transport and Storage [21], the fuel route will define the refuelling operations, dry storage system operational procedures and arrangements. EIMT aspects will also need to be considered at a high level for the fuel route and dry storage systems.

9.1.4 Assumptions

No assumptions are identified in this revision.

9.2 NORMAL OPERATING STATES

Normal operations are defined in the IAEA Nuclear Safety and Security Glossary [22] as 'operations within specified operational limits and conditions'. Normal operation includes all the operating modes permitted at the facility, e.g., start-up and shutdown states and temporary situations arising due to maintenance and testing. They also include minor deviations from desired operating conditions provided these are appropriately justified in the safety case (i.e., they include what the IAEA terms Anticipated Operational Occurrences (AOOs) [22]).

This subchapter outlines the normal operating modes that are relevant to the generic SMR-300, including start-up, normal operations to produce power, shut-down, abnormal operations and refuelling. It also identifies the fuel route, transport, and storage as operations, and lifting operations that will ultimately be subject to operating limits and conditions. These will remain subject to development beyond this GDA stage, and will inform the scope of the SSEC, which will in turn, derive the limits and that require incorporation into the operating documentation, to operate the plant safely. The production of operating documentation is discussed further in sub-chapter 9.7.1, which will be undertaken in the licensing stage.

As the generic SMR-300 design is still under development, the final operating regime for normal operations is yet to be fully defined. Future revisions of this chapter will reflect the ongoing development of the normal operating regime.

Normal operation of the generic 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 [23] 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.

The alignment with UK plant condition classes is shown in PSR Part B Chapter 14 Design Basis Accident Analysis [24]. In the Plant Overview document, Holtec have defined the operating state Abnormal Operations to align with those AOOs occurring once per rector lifetime. This will not change during the period of the GDA.

The following operations are defined for SMR-300, covered by the SMR-300 Plant Overview [25]:

- Start-up.
- Normal operations to produce power.
- Shut down.
- Abnormal operations.
- Refuelling.

9.2.1 Startup Operations

Reactor Coolant Pump (RCPs) are used to establish Reactor Coolant System – General (RCS) flow and provide heat for plant warmup. Once the RCS is at normal operating pressure and temperature (Normal Operating Pressure (NOP)/ Normal Operating Temperature (NOT), control rods are withdrawn to criticality. Power increases are performed with further withdrawal of the control rods. Steam is bypassed around the main steam isolation valve to warm up the steam lines and the steam turbine. Steam is admitted to the turbine for turbine warmup and the turbine generator is synchronized to the grid at a minimum power level.

9.2.2 Power Operations

During normal operations, the primary coolant is driven by RCPs. High pressure, superheated steam is produced in the once-through Steam Generator (SGE) and directed to the turbine generator. Feedwater is returned to the SGE with the use of condensate and feedwater pumps. Feedwater heating is used to assist in secondary side efficiencies and SGE thermal limits.

The plant control scheme is designed for operation with the SGE and is based on a "feed forward" strategy. A change in the electrical grid load is automatically compensated by the turbine control valves opening or closing to admit steam to the turbine to keep the active power variations within the limits mandated by the national grid code. The generic SMR-300 is designed to accommodate the following operational occurrences without causing a reactor trip or Engineered Safety Functions (ESF) actuation signal:

- (REDACTED)

9.2.3 Shutdown Operations

Control rods are fully inserted to shut down the core. The SGE is initially used to remove the decay heat. The steam produced by the SGE after reactor shutdown is sent directly to the condenser through the turbine bypass valves. The condensate from the condenser is returned to the SGE using the main feedwater system. The RCPs are used to ensure thermal equilibrium of the RCS. When RCS pressure is reduced to below the Residual Heat Removal System (RHR) operating pressure, RHR is placed in service to continue decay and sensible heat removal from the core. The RHR heat exchangers are sized to cool the RCS to (REDACTED).

9.2.4 Abnormal Operations

If the Main Generator is unavailable the Unit Auxiliary Transformer (UAT) is supplied from the switchyard via back feed of the Main Step-Up Transformer (MSU). If the UAT is unavailable, a bus transfer will be initiated to transfer the loads to the Station Service Transformer (SST), which is fed from the switchyard. In the event of a loss of transmission grid, the main generator can continue to provide power plant loads independent from the grid in Island Mode operation. The generic SMR-300 design does not require ‘immediate’ operator action and has the ability to mitigate design basis accidents with no operator intervention.

The ability to operate the generic SMR-300 in island mode adds an additional layer of reliability during a response to a LOOP. This allows the generic SMR-300 to be sited in a wider variety of locations, even where the grid is unreliable, without loss of safety.

For a list of the AOOs relevant to the generic SMR-300, see PSR Part B Chapter 14 Design Basis Accident Analysis [24].

9.2.5 Refuelling Operations

The generic SMR-300 is designed to operate on a nominal 18-month fuel cycle with approximately one-third of the fuel assemblies in the core discharged every refuelling outage. The refuelling process for the generic SMR-300 is similar to a conventional Boiling Water Reactor (BWR) given the orientation and close proximity of the Spent Fuel Pool (SFP) to the reactor. Placing the SFP adjacent to the Reactor Pressure Vessel (RPV) eliminates the complexity and maintenance associated with conventional Pressurised Water Reactor (PWR) fuel up-ending and transferring equipment to facilitate transfer of fuel between the containment and the fuel handling building.

A high level summary of the SMR-300 refueling operations are as follows:

- New fuel is loaded into a Multi-Purpose Canister (MPC) (inside HI-TRAC), in the Fuel Handling Area (FHA) of the Reactor Auxiliary Building (RAB).
- The HI-TRAC with its MPC is moved from the FHA into containment via the equipment hatch and then into the SFP cask area.
- All fuel assemblies from the reactor core are offloaded to the SFP storage rack.
- New fuel is moved from the HI-TRAC to the SFP storage rack.
- Fuel insert hardware is re-arranged to the required fuel assemblies.
- New, once, and twice-burned fuel assemblies are moved to the reactor core.
- Spent fuel remains in the SFP until the required cooling time for dry storage is achieved. Spent fuel from prior cycles that has reached its required cooling time is moved into the same MPC used to bring new fuel into the SFP and is removed from SFP using the same HI-TRAC.
- The HI-TRAC with its MPC is moved to the FHA.
- The MPC is processed for dry storage inside the HI-TRAC.
- The HI-TRAC is brought to the onsite dry storage facility and transferred into the UMAX storage module.

A detailed description of the generic SMR-300 refueling operations is presented within in PSR Part B Chapter 24 [21].

During refuelling, maintenance of equipment inside containment, including inspection of SGE tubes, is performed, as necessary.

9.2.6 Fuel Route, Transport and Storage Operations

9.2.6.1 Transfer of New Fuel to Reactor

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

The refuelling operation is summarised in sub-chapter 9.2.5.

An MPC is placed inside a HI-TRAC transfer cask and brought to the RAB door on a vertical cask transporter. The HI-TRAC is lowered onto a low profile transporter (LPT) which transports the HI-TRAC into the FHA of the RAB.

The NFV is in the FHA of the RAB. New fuel assemblies are loaded into the HI-TRAC one at a time, using the overhead crane in the RAB. The HI-TRAC is then transported on the LPT through the Containment Structure (CS) equipment hatch. Inside the CS, the HI-TRAC is flooded with borated water and the Polar Crane lifts the HI-TRAC off the LPT and lowers it into the SFP.

The new fuel from the HI-TRAC is moved into the reactor core using the fuel handling bridge crane. Fuel that will be used for another cycle is returned to the reactor core from the Spent Fuel Storage Racks (SFSR) using the fuel handling bridge crane.

9.2.6.2 Transfer of Spent Fuel to Dry Storage

The SMR-300 spent fuel dry storage system is part of the integrated fuel management system. SMR-300 utilises on-site interim spent fuel storage within Holtec's HI-STORM UMAX system, containing an MPC.

Consistent with requirements relating to decay heat, burnup, and cooling time, spent fuel can be retrieved from the SFP for dry storage after as little as 3 years of cooling in the SFP. It is expected that dry storage campaigns will commence after the third or fourth refuelling outage (4 ½ – 6 years after start of generation), when the discharged spent fuel assemblies of the first cycle will be removed from the SFP and placed into underground dry storage in the HI-STORM UMAX system, which is located on-site. Dry fuel storage campaigns will be undertaken following every subsequent refuelling outage and until all the spent fuel assemblies are placed into dry storage.

Lifting, handling, processing, and transportation equipment is designed to efficiently move spent fuel from the SFP to the UMAX Independent Spent Fuel Storage Installation (ISFSI).

When a dry storage campaign is performed during the refuelling outage, the same HI-TRAC that was used to bring in the new fuel will be used to move spent fuel assemblies into dry storage. After the new fuel has been removed, and the HI-TRAC is still within the SFP, the spent fuel assemblies are loaded into the HI-TRAC using the Fuel Handling Bridge Crane.

A thick stainless-steel lid is placed on the MPC using the polar crane. The HI-TRAC is removed from the SFP using the polar crane and placed on the LPT. The HI-TRAC provides shielding from the spent fuel assemblies and structural protection for the MPC. The thick MPC lid and the water in the MPC also provide shielding.

The HI-TRAC is moved out of the CS the same way it entered and is placed in a dedicated area in the RAB where the spent fuel assemblies and the MPC are prepared for dry storage.

The MPC is drained, dried, backfilled with inert gas and welded shut to provide a pressure vessel boundary as a high integrity containment/ confinement barrier for radioactivity both whilst on-site for example during interim storage within the ISFSI and during any subsequent off-site transport. When complete, the HI-TRAC is moved outside of the RAB the same way it entered.

A Vertical Cask Transporter (VCT) then lifts the HI-TRAC off the LPT and carries it to the on-site HI-STORM UMAX dry storage facility. At the dry storage facility, the HI-TRAC is positioned over the HI-STORM UMAX cavity. The bottom hatch of the HI-TRAC is removed and the MPC is lowered into place.

The HI-TRAC is removed from the area and the HI-STORM UMAX closure lid is placed over the cavity to complete the dry storage process.

Each HI-STORM UMAX Vertical Ventilated Storage Module (VVM) provides storage of an MPC in the vertical configuration inside a cylindrical cavity located entirely below the top-of-grade of the ISFSI. The VVM, akin to an above ground over-pack, is comprised of a cavity enclosure container and closure lid, as well as interfacing structures.

On-site interim dry storage within the HI-STORM UMAX system is expected to be the storage means for spent nuclear fuel for at least the design life of the SMR-300 reactor.

Based on previous Requesting Parties' GDA submissions for standard PWR fuel with similar characteristics, an interim storage period for spent fuel of 100 years or more is likely to be necessary prior to disposal in the UK's planned Geological Disposal Facility (GDF).

A detailed description of the generic SMR-300 fuel route, including transport and storage is discussed in Chapter B24 [21].

9.2.7 Conduct of Lifting Operations

Heavy loads are not typically lifted within the Containment Structure of the SMR-300 during power operations since most heavy loads are associated with refuelling operations (described above in subchapter 9.2.5), and a heavy load drop would not result in a plant trip during normal power operations. There is the potential for lifts to be undertaken within the RAB during power operations which, should they fall, could result in a reactor trip.

The conduct of lifting operations will be managed in accordance with the requirements of the SSEC to ensure that risks posed by lifting operations will be ALARP. Lifting operations will be assessed within the safety assessment as outlined within PSR Part B Chapter 22 Internal Hazards [26], and any limits and conditions that are required to ensure that risk is managed ALARP (eg plant configurations to ensure appropriate Defence in Depth (DiD), maximum lift masses in accordance with the withstand capability of SSCs). Further information will be presented on this as the SSEC matures.

Within the UK all lifts required within the generic SMR-300 facility and site will comply with the Approved Codes of Practice (ACoP) for Lifting Operations and Lifting Equipment Regulations (LOLER) [27] and Provision and Use of Work Equipment Regulations (PUWER) [28] guidance. The conduct of lifting operations will be defined during the nuclear site licensing phase to ensure that lifting operations are compliant with the limits and conditions of the SSEC.

9.3 SSCS ESSENTIAL TO NORMAL OPERATIONS

This subchapter outlines the relevant SSCs which provide a function within the normal operations of the generic SMR-300 plant. This includes normal startup, shutdown, and refuelling operations.

The SSCs supporting normal operational modes of the SMR-300 are identified relevant to the operational modes identified in subchapter 9.2.

PSR Part A Chapter 2 [2] identifies the following GDA gap: A UK categorisation and classification scheme for the generic SMR-300 safety functions is required to be presented for normal operations and fault conditions for all aspects of the Generic SMR-300 within the GDA scope. This exercise will identify all SSCs with a normal operation safety function and will inform the full scope of normal operation SSCs to be included within this subchapter at PSR Revision 1.

For this PSR chapter revision, for each of the within GDA scope SSCs (defined in PSR Part A Chapter 2 [2]), the following documentation types have been reviewed to identify an indicative list of normal operation SSCs with requirements applicable to Normal Operations. Table 1 captures the SSCs which fall into the category of being both within the GDA scope and essential to normal operations. This table will be further developed at PSR Revision 1 and will continue to mature beyond GDA, upon completion of all detailed safety analyses.

Table 1: SSCs Essential to Normal Operations

System Acronym	SSC Description	Sub-Systems and Components
MSS	Main Steam System (MSS)	Turbine Bypass System Extraction Steam System Piping and Fittings Main Steam Safety Valves Atmospheric Dump Isolation Valve Atmospheric Dump Valve Main Steam Drains Main Steam Isolation Valve Main Steam Isolation Bypass Valve Turbine Bypass Valves Turbine Bypass Spray Valves Desuperheaters Moisture Separator Reheater (MSR) Valves MSR Drain Tank Extraction Steam Components (Valves and Drains)
MFS	Main Feedwater System (MFS)	Condensate Pumps Feedwater Pumps Condensate Storage Tank Structure Condensate Polisher Package Feedwater Heaters Feedwater Control Valves Feedwater Startup Control Valve Feedwater Isolation Valve Feedwater Check Valve Piping

System Acronym	SSC Description	Sub-Systems and Components
SFC	Spent Fuel Pool Cooling System (SFC)	Spent Fuel Pool Cooling Pumps Spent Fuel Pool Cooling Heat Exchangers Spent Fuel Pool Cooling Demineralizers with Filters Refueling Water Storage Tank Refueling Water Storage Tank Purification Pump Containment Isolation Valves Demineralizer Flow Control Valve Piping
CBV	Containment Ventilation System (CBV)	Air Handling Units Air Filtration Units Containment Cooling and Reactor Cavity Cooling Fans Containment Cooling Return Fans Ducting/Piping Valves/Dampers
LRW	Liquid Radioactive waste System (LRW)	Equipment Drain Subsystem Floor Drain Subsystem Chemical Waste Subsystem Containment Isolation Valves Reactor Coolant Drain Tank Pumps Containment Sump Pumps Effluent Holdup Tank Pump Waste Holdup Tank Pump Effluent Monitoring Tank Pump Waste Monitoring Tank Pump Reactor Coolant Drain Tank Containment Sump Effluent Holdup Tanks Waste Holdup Tanks Effluent Monitoring Tanks Waste Monitoring Tanks Chemical Waste Tank Ion Exchangers Filters Degasifier
GRW	Gaseous Radioactive waste System (GRW)	Gas Cooler Moisture Separator Guard Bed Delay Beds Filters
SRW	Solid Radioactive waste System (SRW)	Spent Resin Tank Resin Mixing Pump Resin Transfer Pump Filter Transfer Cask Resin Fines Filter
LVE	Low Voltage AC Distribution System (LVE)	Load Center Transformers Load Voltage Switchgear Motor Control Center Low Voltage Breakers Motor-Generator Set Potential Transformers and Current Transformers Protective Relays

System Acronym	SSC Description	Sub-Systems and Components
ICE	I&C Power Distribution System (ICE)	Uninterruptible Power Supply (UPS) Inverter Static Switch Static Bypass Switch Maintenance Bypass Switch Regulating Transformer Panelboard 120 VAC Breakers
DCE	DC Power Distribution System (DCE)	Battery Charger Panelboard Battery Bank HVAC DC Breakers Instruments
CS	Containment Structure (CS)	Containment Structure Support Equipment Hatches Mechanical Penetrations Electrical & Instrument Line Penetrations
RAB	Reactor Auxiliary Building (RAB)	Containment equipment hatch entrance and containment personnel access Class 1E and non-Class 1E electrical equipment and rooms Instrumentation and control equipment and rooms Mechanical equipment rooms Containment mechanical piping penetration areas Containment electrical penetration areas New fuel vault
CES	Containment Enclosure Structure (CES)	N/A
STE	Steam Tunnel	N/A

System Acronym	SSC Description	Sub-Systems and Components
CVC	Chemical and Volume Control System (CVC)	Mixed Bed Demineralizers Cation Bed Demineralizer Deborating Bed Demineralizer Reactor Coolant Filters Makeup Filter Letdown Orifice Boric Acid Mixing Tee Regenerative Heat Exchanger Letdown Heat Exchanger Recirculation Pumps Charging Pumps Boric Acid Recirculation Pump Effluent Holdup Tank Pump Boric Acid Mixing Tank Boric Acid Storage Tank Chemical Mixing Tank Effluent Holdup Tanks Reactor Coolant Pressure Boundary Isolation Valves Pressurizer Spray Pressure Control Valve Pressurizer Spray Bypass Valve Letdown Orifice Bypass Valve Demineralizer Bypass Valve Makeup Filter Bypass Valve Demineralizer Inlet Valves Charging Line Isolation Valve Refuelling Water Storage Tank (RWST) Supply Valve Effluent Holdup Tank Supply Valves Effluent Holdup Tank Discharge Valves Boric Acid Recirculation Valves Effluent Holdup Tank to Charging Pump Isolation Valve Batch Flow Control Valve Boric Acid Flow Control Valve Demineralized Water Flow Control Valve Demineralized Water System Isolation Valves Containment Isolation Valves Piping Degasifier
RCS	Reactor Coolant System (RCS)	Reactor Pressure Vessel Steam Generator Pressurizer Jet Pump Reactor Coolant Startup Pump Reactor Coolant Startup Heat Exchanger Pressurizer Safety Valves RCS Piping

System Acronym	SSC Description	Sub-Systems and Components
RHR	Residual Heat Removal System (RHR)	Residual Heat Removal Pumps Residual Heat Removal Heat Exchangers Low Temperature Overpressure Protection Relief Valve Reactor Coolant Pressure Boundary Isolation Valves Residual Heat Removal Heat Exchanger Flow Control Valves Residual Heat Removal Pump Recirculation Valves Residual Heat Removal Heat Exchanger Bypass Flow Control Valves Reactor Coolant System Fill from RWST Valve Residual Heat Removal to RWST Control Valve Piping
PCS	Plant Control System (PCS)	Large Display Panel Operator Visual Display Unit (VDU) Alarm VDU Procedure VDU
EIS	Excore Instrumentation System (EIS)	N/A
DAS	Diverse Actuation System (DAS)	N/A
IIS	Incore Instrumentation System (IIS)	N/A
CRD	Control Rod Drive Mechanism (CRD)	N/A
SGE	Steam Generator	N/A
RPV	Reactor Pressure Vessel (RPV)	RPV Upper Shell RPV Lower Shell RPV Bottom Head RPV Reverse Flange RPV DVI Nozzles RPV-SG Nozzle RPV Closure Head CRDM Nozzles In-Core Instrumentation Nozzles Head Vent Nozzle RPV Top Flange Flange Stud RPV Supports
PZR	Pressuriser (PZR)	Pressurizer Flange Flange Gasket Leak Detection Lateral Restraints Nozzles Pressurizer Spray Nozzle Studs and Bolts
CDS	Control Rod Drive System (CDS)	Control Rod Drive Mechanism Drive Shaft Assembly Remote Disconnect Mechanism Control Rod Assembly Control Rod Control System Control Rod Position Indication System

System Acronym	SSC Description	Sub-Systems and Components
MCH	Main Control Room Habitability System (MCH)	Control Room Normal Ventilation System Main Control Room Air Handling Units Main Control Room Supplementary Air Filtration Unit Main Control Room Smoke Purge Fan Toilet/Kitchen Exhaust Fan Air Operated Dampers Ductwork

9.4 NORMAL OPERATIONS, 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 PSR Part A Chapter 1 Introduction [29]) is achieved.

The fundamental purpose follows a golden thread throughout the SSEC to CAE via objectives of the PSR, Preliminary Environmental Report (PER) and Generic Security Report (GSR). The overarching SSEC claims are presented in PSR Part A Chapter 3 [1], and this chapter links to the overarching claims through Claim 2.3.

This chapter contributes directly to Claim 2.3, which is focused on the demonstration that the design and safety assessment are developed to consider all reactor lifecycle stages, that normal operations are adequately assessed and informed by the safety case.

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

As set out in Part A Chapter 3 [1], Claim 2.3 is further decomposed across several disciplines which support the development of through-life management arrangements. This chapter presents the conduct of operations aspects for the generic SMR-300 and therefore directly supports a claim focused on the appropriate arrangements to safely manage people and plant, Claim 2.3.1.

Claim 2.3.1: Appropriate arrangements to safely manage people and plant during the construction, commissioning and operation of the Generic Holtec SMR-300 are suitably mature for a generic design.

Claim 2.3.1 has been further decomposed within Part B Chapter 9 to provide confidence that the relevant requirements for operational aspects and conduct of operations will be met. To achieve this, claim 2.3.1 is broken down into 5 Level 4 sub-claims.

Claim 2.3.1.3 contributes to the demonstration of appropriate management arrangements during operation by identifying that the conduct of operations is based upon international good practice.

Claim 2.3.1.4 shows that there is a robust process to ensure all limits and conditions will be clearly identified.

Claim 2.3.1.5 ensures that these limits and conditions will be incorporated into operational documentation.

Claim 2.3.1.6 identifies that arrangements will then be in place to ensure the plant is operated in accordance with these limits and conditions.

Claim 2.3.1.7 is identified to ensure that EIMT arrangements are in place to ensure that SSCs can continue to deliver their relevant functions throughout their operational life.

It should, however, be noted that operational aspects and conduct of operations constitutes safety case implementation and as such, largely undertaken by the Licensee following the Pre-Construction Safety Report stage. It will therefore be fully developed within the nuclear site licensing phase of the generic SMR-300 and not within the GDA process.

Table 2 shows in which chapter of this PSR these claims are demonstrated to be met.

Table 2: CAE Chapters

Claim No	Claim	Chapter Section
2.3.1.3	The SMR-300 conduct of operations is derived based upon identified international good practice.	9.5 – Codes, Standards and Regulations
2.3.1.4	The limits and conditions of the safety case are clearly identified, covering all operational modes and plant configurations.	9.6 – Operating Limits and Conditions
2.3.1.5	The limits and conditions of the safety case will be incorporated into operational documentation.	9.7 – Operating Procedures
2.3.1.6	The plant will be operated safely in accordance with the limits and conditions of the safety case.	9.8 – Main Control Room and Remote Shutdown Facility
2.3.1.7	The Examination, Inspection, Maintenance and Testing (EIMT) activities to ensure the SSCs continue to achieve their safety functional requirements throughout operational life are identified.	9.9 – Examination, Inspection, Maintenance and Testing.

A summary of the current CAE route map for Part B Chapter 9 is provided in Appendix A and a further update on claim decomposition, argument development and evidence maturity will be provided in the subsequent update of the Chapter.

9.5 CODES, STANDARDS AND REGULATIONS

Claim 2.3.1.3: The SMR-300 conduct of operations is derived based upon identified international good practice.

This subchapter outlines the codes and standards that are relevant for the operational aspects and conduct of operations of the generic SMR-300. The full definition of the conduct of operations to enable compliance with the UK Licence Conditions will be developed within the nuclear site licensing phase of the generic SMR-300, and not within the GDA process.

9.5.1 SMR-300 Concept of Operation Codes, Standards and Regulations

The SMR-300 conduct of operations is based on the requirements set out in:

- NUREG 0800 Standard Review Plan [30].
- NUREG 0711 Human Factors Engineering Program Review Model [31].
- NUREG/CR-7216 Human Performance issues related to the design and operation of SMRs [32].

In addition to the above, the following relevant good practice guidance is available from the IAEA and displayed in Table 3.

Table 3: List of Regulations and International Guidance used for Generic SMR-300 Design Standard

Document/Label	Title
IAEA Safety Guide No. NS-G-70 [33]	Operational Limits and Conditions and Operating Procedures for Nuclear Power Plants
IAEA Safety Guide No. NS-G-72 [34]	The Operating Organization for Nuclear Power Plants
IAEA Safety Guide No. NS-G-74 [35]	Testing, Surveillance, and Inspection in Nuclear Power Plants
IAEA Safety Guide No. NS-G-76 [36]	Conduct of Operations at Nuclear Power Plants

The expectations of NUREG 0800 regarding the conduct of operations are that the following topics are addressed:

- Management of the technical support organisation and the operating organisation, including the training and qualification of operators and staff, operational programmes, and administrative procedures - more on the development of the future operating organisation can be found within Chapter A4 [3] of this SSEC.
- Emergency planning - more on this aspect is described within Chapter B15 [18].
- Operational procedures - this aspect is discussed in subchapter 9.7.
- Maintenance – this is discussed within subchapter 9.8.
- Security and cyber-security – are described within the Generic Security Report [37].

PSR Part B Chapter 17 [19] presents how Human Factors consideration will be applied within the GDA stage to demonstrate that the process for Human Factors Engineering (HFE) applied to the generic SMR-300 is aligned with ONR expectations for HF integration in the UK. This includes the Operational Philosophy and the feasibility of proposed staffing levels.

The compliance with NUREG-0711 [31] and NUREG/CR-7216 [32] and the involvement of human factors engineering in the conduct of operations is described in subchapter 9.7 where operating procedures and staffing and qualification review processes are discussed.

9.5.2 UK Context and GDA

The generic SMR-300 must be capable of meeting all 36 licence conditions set out by the ONR within their Licence Condition (LC) handbook [38] when deployed to a site in the UK. Multiple LCs relate to this PSR Chapter, and a few of the key ones are listed below, this is a non-exhaustive list:

- LC 23 – Operating Rules [39].
- LC 24 – Operating Instructions [40].
- LC 25 – Operational Records [41].
- LC 26 – Control and Supervision of Operations [42].
- LC 27 – Safety Mechanisms, Devices and Circuits [43].
- LC 28 – Examination, Inspection, Maintenance and Testing [44].

The ONR Safety Assessment Principles (SAPs) from Technical Assessment Guide (TAG)-35 [45]. ONR TAG-35 provides useful guidance relating to LACs and Operating Rules.

The Limits and Conditions for Nuclear Safety [45] include specific principles for EIMT, including:

- Identification of requirements EMT.1.
- Frequency EMT.2.
- Type-testing EMT.3.
- Validity of equipment qualification EMT.4.
- Procedures EMT.5.
- Reliability claims EMT.6.
- Functional testing EMT.7.
- Continuing reliability following events EMT.8.

SAPs relevant to ageing and degradation include:

- Safe working life EAD.1.
- Lifetime margins EAD.2.
- Periodic measurement of material properties EAD.3.
- Periodic measurement of parameters EAD.4.
- Obsolescence EAD.5.

SAPs relevant to layout include:

- Access ELO.1.
- Unauthorised access ELO.2.
- Movement of nuclear matter ELO.3.
- Minimisation of the effects of incidents ELO.4.

TAG-09 Examination, Inspection, Maintenance and Testing of items important to Safety [46] provides useful guidance relating to EIMT of items important to safety. The following useful IAEA guidance is contained within TAG-09:

- Items important to safety shall be designed to be calibrated, tested, maintained, repaired, or replaced, inspected, and monitored as required to ensure their capability of performing their functions and to maintain their integrity in all conditions specified in their design basis.
- The design life of items important to safety shall be determined. Appropriate margins shall be provided in the design to take due account of relevant mechanisms of ageing, neutron embrittlement and wear out and of the potential for age related degradation, to ensure the capability of items important to safety to perform their necessary safety functions throughout their design life.
- The operating organisation shall ensure that a systematic assessment is carried out to provide reliable confirmation that safety related items are capable of the required performance for all operational states and for accident conditions.
- The operating organization shall ensure that effective programmes for maintenance, testing, surveillance, and inspection are established and implemented.
- From construction to commissioning and finally to operation, the plant should be adequately monitored and maintained. The plant should be subject to the required inspection and periodic testing to protect equipment, to support the testing stage and to continue to comply with the safety analysis report and operational limits and conditions.
- The maintenance programme for a nuclear power plant should include all the administrative and technical measures that are necessary to detect and mitigate the degradation of a functioning SSC or to restore to an acceptable level the performance of design functions of a failed SSC.

In addition to the LCs and TAG set out by the ONR, operators, members of the public and the environment must be protected as required by the legislation below:

- Health and Safety at Work (etc.) act 1974 [47].
- Occupiers Liability Act 1957 [48].
- Factories Act 1961 [49].
- Offices, Shops and Railway Premises Act 1963 [50].
- Fire Safety Act 2021 [51].
- Occupiers Liability Act 1984 [52].
- Environmental Protection Act 1990 [53].
- Environment Act 2021 [54].
- Marine and Coastal Access Act 2009 [55].
- Pollution Prevention and Control Act 1999 [56].
- Building Act 1984 [57].

In addition to the suite of general legislation that applies over-arching health and safety requirements on the company (both as an employer and as nuclear operator); there are specific legislations, regulations and standards that apply to the management, design, ownership, and maintenance of the civil infrastructure on the nuclear sites.

Relevant legislation to civil assets and infrastructure include:

- The Ionising Radiation Regulations 2017 (IRR 2017) [58].
- The Management of Health & Safety At Work Regulations 1992 [59].
- The Construction (Design & Management) Regulations 2015 [60].
- The Construction (Health, Safety and Welfare) Regulations 1996 [61].
- The Workplace (Health, Safety and Welfare) Regulations 1992 [62].
- The Control of Substances Hazardous to Health (Amendment) Regulations 1988 [63].
- The Construction (Head Protection) Regulations 1989 [64].
- The Manual Handling Operations Regulations 1992 [65].
- The Personal Protective Equipment at Work Regulations 1992 [66].
- The Reporting of Injuries Diseases and Dangerous Occurrences Regulations 1995 [67].
- The Confined Spaces Regulations 1997 [68].
- The Lifting Operations and Lifting Equipment Regulations 1998 [27].
- The Control of Noise at Work Regulations 2005 [69].
- The Safety Signs Regulations 1980 [70].

Although the SMR-300 has been designed in line with US NRC Regulatory Requirements, the operational constraints, maintenance, and procedures are likely to remain in line with one another. Where the conduct of operations has not been considered within the generic SMR-300 design due to the differing approaches between UK and US regulatory requirements, gaps have been identified, and where appropriate, Forward Actions (FAs) proposed across the PSR engineering chapters.

The work carried out in support of the generic SMR-300 has been informed by activities described in the Human Factors Engineering Program Management Plan [71]. It is recognised that this is based on NUREG 0711, which is based on NRC expectations rather than those of the ONR. While both represent relevant good practice there are differences between US and UK conventions that must be accounted for. Chapter B17 [19] will demonstrate the applicability and propose forward actions to address gaps in the processes.

The involvement of Human Factors team based in the UK and the HFE team based in the US in the conduct of operations is fundamental to the successful involvement of the operators in the conduct of operations.

Concept of Operations (ConOps) chapter of the human factors engineering programme refers to the way in which the operating crew is organised, and how it monitors and controls the plant under normal and abnormal conditions. The ConOps includes topics such as:

- Plant Mission.
- Agents' Roles and Responsibilities
- Staffing, Qualifications, and Training.
- Management of:
    - Normal Operations.
    - Off-normal Conditions and Emergencies.
    - Maintenance and Modifications.

The ConOps is not yet complete and will be developed during regulatory engagement.

The SMR-300 HFE program [71] includes a staffing and qualification analysis, whose purpose to determine the required number and necessary qualifications of personnel to complete tasks allocated to people, while meeting all regulations. The staffing analysis includes the full range of plant operating modes such as startup, normal operations, shutdown, abnormal conditions, and emergency conditions. The scope of personnel included in the evaluation is limited to licenced control room operators - other personnel will be analysed by the licensee and this aspect is covered in PSR Chapter A4 [3].

The conduct of operations, including the development of operating documentation will be undertaken during the licensing phase of the generic SMR-300 and not within the GDA process.

This subchapter identifies many of the UK Licensing conditions (LC 23 [39], LC 24 [40], LC 25 [41], LC 26 [42], LC 27 [43]), codes, and standards relevant to the conduct of operations that require compliance to be demonstrated against by the Licensee during the licensing phase.

Operating documentation will be developed for the generic SMR-300, based upon US NRC requirements and with consideration of IAEA guidance (see subchapter 9.5.1) which can inform the development of operating documentation by the Licensee.

9.5.3 CAE Summary

The SMR-300 conduct of operations will be derived based upon identified good practice, including US NRC NUREG and IAEA guidance for the definition of the conduct of operations.

PSR Part B Chapter 17 [19] presents how Human Factors consideration will be applied within the GDA stage to demonstrate that the process for HFE applied to the generic SMR-300 is aligned with ONR expectations for HF integration in the UK, including the operational philosophy and the feasibility of proposed staffing levels. This also supports the argument that the operating organisation will be feasible.

PSR Part A Chapter 4 [3] presents the approach to the management of the technical support organisation and the operating organisation, including the training and qualification of operators and staff, operational programmes, and administrative procedures. This chapter supports the claim that the conduct of operations will be derived based upon identified good practice and that the operating and technical support organisation will be defined and adequately managed to enable and ensure safe operation of the plant.

PSR Part B Chapter 15 [18] presents the approach to emergency planning.

The definition of the conduct of operations and ensuring that safety case requirements are captured by operating documentation will be fully developed within the nuclear site licensing phase of the generic SMR-300, and not within the GDA process. This subchapter identifies many of the UK Licensing conditions, codes, and standards relevant to the conduct of operations that require compliance to be demonstrated against by the Licensee.

Operating documentation will be developed for the generic SMR-300, based upon US NRC requirements and with consideration of IAEA guidance which can inform the development of operating documentation by the Licensee.

This subchapter shows that international good practice is being followed in the definition of the conduct of operations for the generic SMR-300, and that work is planned to enable the development of arrangements and documentation to be compliant with the UK Licence Conditions.

9.6 OPERATING LIMITS AND CONDITIONS

Claim 2.3.1.4: The limits and conditions of the safety case are clearly identified, covering all operational modes and plant configurations.

This subchapter outlines the operating limits and conditions that are relevant for the operational aspects and conduct of operations of the generic SMR-300.

The safety management during the operation of the plant will be the responsibility of the licensee within the operating organisation. Operating limits and conditions are required to ensure the plant is operated safely at all times. Implementation of these limits and conditions on a Nuclear Licenced Site is captured by procedural arrangements responding to the 36 standard LCs that are attached to every Nuclear Site Licence (NSL).

The operational limits and conditions defining normal operations should be derived from the safety case and are Operating Rules (OR) for the purposes of compliance with Licence Condition 23. There are many other detailed requirements applying to nuclear sites, but these are not directly relevant to the GDA process. During GDA, the safety case ownership role rests with the Requesting Party (RP).

The UK ONR regulatory framework is principles driven and based on the over-riding legal requirement that licensees operating nuclear plant reduce the risks arising from their nuclear activities to ALARP. The manner in which ALARP is demonstrated during the generic SMR-300 GDA is via the SSEC and its supporting documentation.

The SSEC will be expected to be developed beyond this GDA to determine the limits and conditions required for safe operation that form the operating envelope for the generic SMR-300 plant. In particular, the design basis analysis and fault studies should provide an input into the safety classification and the engineering requirements for systems, structures and components performing a safety function; the limits and conditions for safe operation; and the identification of requirements for operator actions.

ORs link the safety case analysis and assumptions with actual operational limits and conditions in force at the facility. Many ORs are derived from the facility's deterministic safety analysis. Use of DBA will ensure that there will be sufficient margins of conservatism between the limits applied and the point where there is a threat to safety. The DBA and Fault Studies:

- Determine the limits of normal operation, safety settings, design basis limits and (in conjunction with the engineering analysis) safety limits.
- Inform the required availability of safety measures; and
- Provide significant insight against the expectations of the DiD framework.

The set of ORs will include limits and conditions on the minimum availability of safety measures. PSR Part B Chapter 14 [24] presents the approach to DBA in the GDA that will ultimately be used to inform the development of ORs during the nuclear site licensing phase.

ORs will be derived from the SSEC engineering substantiation to: provide limits on operation to prevent fault initiation or escalation; ensure design assumptions and intent are met; set conditions on appropriate plant and equipment configurations; specify the timing of maintenance and testing activities; cater for plant ageing and corrosion effects; and set ORs relating to equipment qualification.

The PSA can also be used to inform EIMT related ORs and proof test intervals. PSR Part B Chapter 16 [12] presents the approach to PSA in the GDA that will ultimately be used to inform the development of ORs during the nuclear site licensing phase.

PSA will be particularly important in determining ORs governing the availability of safety measures for deriving time-based ORs, those relating to allowed substitution periods and the unavailability of safety measures, so that any periods of elevated risk may be suitably justified by the Licensee and be a key input to holistic reviews of ORs.

Operating limits and conditions will also be determined from:

1. **Periodic Test (PT) Requirements:** This aims to ensure that the safety functions and environment protection functions of SSC continuously meet the design intended reliability during the plant operation. A comprehensive list of the PT that is to be performed on a given system should be established.
2. **Loading Condition (Design Transient) Requirements:** The structural integrity of SSC is ensured based on the analysis of the overall and local study of the primary and secondary components against damage mechanisms such as excessive deformation and plastic instability, elastic or elastoplastic instability, progressive deformation, fatigue, and fast fracture risk. The operating conditions for the main primary and secondary circuits are defined to include transients anticipated in normal operation and emergency and accident conditions.
3. **Core Design Requirements:** This aims to identify the operating limits and conditions from the fuel system design, nuclear design, and thermal and hydraulic design.

Assumptions and other parts of the safety case, such as Severe Accident Analysis, are also sources that will inform the definition of ORs. The development of the SSEC beyond the GDA stage will inform the development of these ORs.

The SSEC will enable the derivation of ORs in accordance with ONR TAG-35 [45] during the nuclear site licensing phase, resulting in:

1. **Operating Technical Specification (OTS) Requirements:** This aims to set out the limits and conditions that must be followed to ensure the plant operation is within the assumptions and limits justified by the safety case. This should include safety limits, safety system settings, limits and conditions for normal operation and the required action for deviation from the limits and conditions.
2. **Environmental Technical Specification (ETS) Requirements:** This aims to ensure compliance with any granted environmental permit, any letter of compliance granted by waste service providers and application of Best Available Techniques (BAT). Environmental requirements are to be established and the SSC assigned with environment protection functions and significant operating limits and conditions are identified.
3. **Chemical and Radiochemical Specification (CRS) Requirements:** This defines the technical regulations of chemistry and radiochemistry that shall be respected during

normal operation of the plant. It aims to reduce chemistry-related risks SFAIRP, examples of this would be minimising the source term and ensuring the integrity of the nuclear safety barriers. Further information is available in PSR Part B Chapter B23 [20].

4. **In-Service Inspection (ISI) Requirements:** ISI is a preventative maintenance process involving the use of visual inspections and Non-destructive Testing (NDT) for all items important to safety at scheduled intervals during operation. It can detect anticipated degradation before it compromises structural integrity and confirms the absence of unanticipated degradation that could lead to failure.

The operational phase of the plant requires the following key arrangements:

- Operating Instructions – for operations that may affect safety during all phases of the plant life (discussed further in subchapter 9.7).
- Staffing levels – which must be maintained with technical competences to ensure adequate protection of health, safety, and the environment (discussed further in subchapter 9.7).
- Training – for personnel who have responsibility for an action that may affect safety (discussed further in subchapter 9.7).
- Emergency procedures and services – an appropriate emergency plan should be in place to control and mitigate unforeseen circumstances.
- Radiological protection – to keep worker and public doses from radiation exposure to ALARP.
- Nuclear material arrangements – to ensure that the introduction and storage of any nuclear materials is controlled.

Operating limits and conditions are subject to plant-specific development and on-going maintenance throughout the whole plant lifetime. The whole set of the operating documentation relies on the design requirements and the licensee's experience and practice and will be finalised in the nuclear site licensing phase.

9.6.1 CAE Summary

This subchapter outlines that production of the SSEC will ensure that operating limits and conditions of the generic SMR-300 will be identified.

The SSEC will enable the derivation of ORs (limits and conditions) in accordance with ONR TAG-35 [45] during the nuclear site licensing phase, resulting in operating limits and conditions, to inform the production of the Technical Specifications for the purposes of compliance with Licence Condition 23. The SSEC will be expected to be developed beyond this GDA to determine the ORs required for safe operation that form the operating envelope for the generic SMR-300.

ORs link the safety case analysis and assumptions with actual operational limits and conditions in force at the facility. The SSEC:

- DBA and Fault Studies will determine the limits of normal operation, safety settings, design basis limits and (in conjunction with the engineering analysis) safety limits; Inform the required availability of safety measures; and provide significant insight against the expectations of the DiD framework.

- Engineering substantiation will provide limits on operation to prevent fault initiation or escalation; ensure design assumptions and intent are met; set conditions on appropriate plant and equipment configurations; specify the timing of maintenance and testing activities; cater for plant ageing and corrosion effects; and set ORs relating to equipment qualification.
- PSA will inform EIMT related ORs and proof testing intervals, and also determining ORs governing the availability of safety measures for deriving time-based ORs, those relating to allowed substitution periods and the unavailability of safety measures, so that any periods of elevated risk may be suitably justified by the Licensee and be a key input to holistic reviews of ORs.

Operating limits and conditions are subject to plant-specific development and on-going maintenance throughout the whole plant lifetime. The whole set of the operating documentation relies on the design requirements and the licensee’s experience and practice and will be finalised in the nuclear site licensing phase.

9.7 OPERATING PROCEDURES

Claim 2.3.1.5: The limits and conditions of the safety case will be incorporated into operational documentation.

This subchapter outlines operating procedures that are important to plant safety. It includes how they support and guide personnel to interact with plant systems and respond to plant-related events under different conditions. These procedures form an essential part of the administrative safety measures.

Modern power stations adopt a Technical Specification (Tech Spec) approach to ORs, whereby limits and conditions include specific time periods so that non-compliance is only deemed to have occurred when the limit has been exceeded for longer than a prescribed time, or on a specific number of times within a given time period.

This approach, where appropriately justified in the safety case, provides a graded approach of OR compliance and can seek to avoid situations whereby returning to normal operations may induce a greater risk than a slower, more measured return to normal operations. The Tech Spec approach has also been applied to environmentally significant plant and activities and are designated as Environmental Specifications (ESpecs).

As outlined within the SMR-300 Plant Overview Document [25], Technical Specifications for the Construction Permit Application (CPA) will be developed in accordance with 10 CFR 50.34(a)(5) [23]. Technical Specifications do not need to be fully developed in accordance with 10 CFR 50.36 until the Operating License Application (OLA).

The generic SMR-300 Tech Specs will inform the development of operating documentation in accordance with LCs 23 [39], 24 [40] and 25 [41] during the nuclear site licensing phase. The Tech Specs will be based upon the limits and conditions derived by the SSEC, as described in subchapter 9.6.

9.7.1 Operating Procedure Development

The following operating procedures will be detailed in the nuclear site licensing phase:

1. NOP.
2. Emergency Operating Procedure (EOP).
3. Severe Accident Management Guideline.

Good practice for procedure structure and layout will be used to develop these operating procedures. Subchapter 9.5.2 shows that operating documentation will be developed for the generic SMR-300, based upon US NRC requirements and with consideration of IAEA guidance which can inform the development of operating documentation by the Licensee.

Procedures are essential to plant safety because they support and guide personnel interactions with plant systems and personnel responses to plant-related events. In the nuclear industry, procedure development is the responsibility of the licensee however, the information comes from the requesting party in the first instance and is then developed by the licensee, using the site-specific requirements and information gained from the construction and commissioning programme.

Appropriate and sound procedures will be supported by the analyses used to develop the human-system interfaces and training. All three elements will be subject to a common evaluation process that verifies the three elements work together to maximise operator performance. To ensure complete integration and consistency, relevant good practice, including human factors principles will be applied to procedures and other human-system interfaces provided to personnel.

PSR Part B Chapter 17 [19] articulates how the HFE programme includes the design and evaluation for the SMR-300 the design. This includes investigating the adequacy of HSIs including indications, controls, alarms, and procedures for credited manual actions. The procedure element is integral to the human factors engineering programme and will be developed by Holtec using accepted human factors engineering principles.

9.7.1.1 Normal Operating Procedure

Normal operation takes place when the plant operates within specified operating limits and conditions, including during start-up, shutdown, power operation and other transient operations within operating limits and conditions. Operation within the specified limits and conditions (i.e., within the 'normal operating domain') ensures that the plant can be run safely. The normal operating domain of the plant is divided into different partitions called operating modes which are used in operating procedures for operational management of the plant.

The NOP provides instructions for safe conduct of all operating modes, implemented during normal operation of the plant, including refuelling. The NOP generally includes periodic test procedures, preventative maintenance procedures, in-service inspection procedures, unit operating procedures, system operating procedure, system alarm sheet and abnormal operating procedure.

The NOP is presented as a set of documents which focus on the main operations and risks for operators during start-up, shutdown, and power operation of the plant. The normal shutdown process consists of plant shutdown, primary circuit draining and opening and core unloading. The normal start-up process consists of core reloading, primary circuit closing and filling and plant start-up.

9.7.1.2 Emergency Operating Procedure

The EOP is a crucial part of the DiD concept. The EOP is to be used in the context of Licence Condition 11 [72] to reestablish safe and steady conditions following transients, incidents, and accidents, by outlining the operator actions and plant operation required in this scenario.

There are Symptom based and Event based procedures. The most advanced approach is Symptom based Operating Procedures. The concepts of symptoms/states have been introduced to respond to the need for reliable and continuous plant diagnosis. Symptom based EOP generally contain both scenario independent and scenario dependent procedures. The operators are guided through an assessment of the overall status of the plant by focusing on a predetermined set of safety functions.

In closed states (where the primary circuit is closed), the EOP is divided into incident conditions (i.e., no degraded state functions or barriers) and accident conditions (i.e., at least one degraded state function or barrier, with the exception of core melt). In non-closed states, no distinction is made between incident and accident conditions.

The EOP addresses the implementation of additional alternative measures wherever possible, but normal operational systems are implemented with priority to avoid demands on engineered safety features.

9.7.1.3 Severe Accident Management Guideline

While the EOP focuses on protecting core integrity, the Severe Accident Management Guideline (SAMG) aims to ensure containment integrity and limiting the release of fission products to the environment. The objective of SAMG is to reach a controlled and stable state. The transition from EOP to SAMG is based on core outlet temperature and/or dose rate in containment.

9.7.1.4 Operating Procedures of the Computer-Based Procedure (CBP) System

The Computer-Based Procedure (CBP) system supports the execution of plant operation by displaying the operating procedures and plant information to the operators of the Main Control Room and Remote Shutdown Facility. The procedure data of CBP system is created based on the paper-based procedures.

The CBP system covers all procedures, including:

- EOP.
- Alarm Response Procedure (ARP).
- Standard Operating Procedure (SOP).

An operating procedure generally consists of an operating objective, prerequisites for applying the operation, and several individual steps (e.g., checking plant parameters and equipment status, determining plant status, operating equipment, and switching to another procedure). The CBP system displays this kind of information in a compact form with the intention of efficiently supporting the tasks performed by the operator and reducing the workload, as well as detecting human errors, improving situational awareness. This will be done by utilizing touch screen displays to provide control room operators with an operating procedure system that is interactive and continuously updated with live plant data. Operators will be able to quickly navigate to appropriate procedure steps or other related procedures using hyperlink navigation.

As the plant detailed design and development progresses, the general technical guideline and site-specific technical guideline for procedures will be prepared, and the generic SMR-300 operating procedures will be finalised. The detailed functional specifications of this document will be revised to match the contents of the final procedures.

9.7.2 CAE Summary

PSR Part B Chapter 17 [19] articulates how the HFE programme includes the design and evaluation for the generic SMR-300. The procedural element is integral to the HFE programme and will be developed by Holtec using accepted human factors engineering principles. The procedures developed by the requesting party will follow accepted HF engineering principles and will ensure that the limits and conditions of the safety case will be incorporated into operational documentation.

Technical Specifications for the generic SMR-300 Construction Permit Application (CPA) will be developed in accordance with 10 CFR 50.34(a)(5) [23]. Technical Specifications do not need to be fully developed in accordance with 10 CFR 50.36 until the Operating License Application (OLA).

The generic SMR-300 Tech Specs will inform the development of operating documentation in accordance with LCs 23 [39], 24 [40] and 25 [41] during the nuclear site licensing phase. The Tech Specs will be based upon the limits and conditions derived by the SSEC.

9.8 MAIN CONTROL ROOM AND REMOTE SHUTDOWN FACILITY

Claim 2.3.1.6: The plant will be operated safely in accordance with the limits and conditions of the safety case.

The Main Control Room (MCR) is designed to provide a habitable area from which to safely operate the reactor. The Remote Shutdown Facility (RSF) is a control station which serves as a backup to the MCR, should it be inoperable or require evacuation.

The SMR-300 will be operated from the MCR in accordance with Technical Specifications to ensure that the plant is operated within the limits and conditions of the safety case (described further within subchapter 9.7).

This subchapter presents a description of the overall design of the MCR and RSF to show that will provide an appropriate means of managing operational activities and delivery of safety functions. This subchapter includes a highly cross-cutting set of arguments, which have multiple interfaces and are supported by a significant number of engineering and safety analysis disciplines. This may be further developed into specific standalone claims and sub-claims that during development of this GDA.

The intent of this subchapter at PSR v0 is to provide arguments, with relevant signposting to the other parts of the PSR, to provide confidence that MCR and RSF related aspects and associated safety justification, is under development for the SMR-300.

9.8.1 Description of the Main Control Room

The MCR is the central location from which all plant operations are coordinated and monitored. It is the principal control station, utilised during both normal and accident conditions to operate the plant safely and maintain it safe. The MCR houses an array of consoles and visual displays to oversee plant systems, monitor plant instrumentation, and control plant components. The MCR is designed to be the controlling station for both SMR-300 units and is in the Reactor Auxiliary Building.

Consoles in the MCR provide the human-system interface (HSI) necessary for plant operation and serve as operator workstations, providing process systems indications as well as the controls required to position, run, or apply power to plant components. There is a console for each on-duty MCR operator, specialized based on their individual role and responsibilities.

Separate, dedicated consoles are provided for interfacing with the Plant Safety System or the Diverse Actuation System for each SMR-300 unit. Auxiliary workstations, such as the HSI Parameter Setting Computer (HPSC), are also present in the MCR, which are not used for plant operation, but which provide access to supporting functions such as screen customization and configuration control.

Plant monitoring and control is done via specialized LCD displays. These Visual Display Units (VDUs) host graphical interfaces which allow operators to view plant indications and trending data, operate plant components, navigate plant procedures, and access plant alarm information.

Each VDU is specialized according to its function and the console at which it is located. A Large Display Panel (LDP) for each SMR-300 unit is mounted on the wall of the MCR and provides readily obtained, real-time data, including key parameters and a plant diagram. This enables operators to make decisions regarding control.

The MCR provides the necessary tools by which to rapidly assess plant safety status. Consoles and screens are arranged so that the Reactor Operator (RO) can, without additional navigation, determine if a protective action has occurred, its completion status, and if a safety channel is inoperable or bypassed. In the event of an accident, the MCR is designed to be utilised for post-accident monitoring (PAM). There is a supervisor's (Senior Reactor Operator (SRO) console as well as the operator console. The supervisor console is monitoring only with commands undertaken from the operator console. It is noted that finalisation of the layout of the MCR is still under development.

REDACTED

Figure 1: (REDACTED)

REDACTED

Figure 2: (REDACTED)

9.8.2 Description of the Remote Shutdown Facility

The RSF is fully functional monitoring and control capability for all SMR-300 systems, including those required to safely shut down the plant. The RSF is designed to be the secondary control station for both SMR-300 units. The MCR and RSF shall not be enabled at the same time to prevent unexpected operation from the location which is not used. The layout of the RSF is the same as the layout in the MCR.

The RSF is located in the Reactor Auxiliary Building. The RSF contains consoles and displays necessary for plant operation. One fully equipped console is provided for the Reactor Operator, and a separate console provides direct interfacing with the PSS. VDUs at both consoles are configured in the same manner as their counterparts in the MCR. The operator in the RSF has access to all plant indications which are normally accessible from the MCR, including those required to assess plant safety status and for post- accident monitoring.

9.8.3 Main Control Room and Remote Shutdown Facility Safety Justification

The following section sets out a number of arguments to support Claim 2.3.1.6, with signposting to interfacing Chapters of the safety report where the evidence to support these arguments is being developed (noting that at PSR stage the maturity of this evidence is limited). The arguments are based upon demonstration that each of the systems physically contained within the MCR and RSF and those interfacing systems that directly support them, are appropriate designed and operated to enable delivery of their relevant safety functions. This includes the need to ensure that hazards and faults with the potential to impact the delivery of these safety functions have been considered and that the MCR and RSF remain habitable to operators.

The design approach for the SSCs associated with the MCR and RSF is appropriate and supports delivery of relevant safety functions.

This argument aims to demonstrate that the design process will ensure that SSCs within the MCR and RSF are appropriately designed and substantiated, in order to deliver their relevant safety functions. The design approach for the MCR and RSF related SSCs is not considered to be different to any other plant area and Part A Chapter A4 [3] provides an overarching description of the SMR-300 design process. Signposting is also provided below for the interfacing Chapters, where further detail on the design approach for each engineering discipline is provided.

- See PSR Part B Chapter 1 Reactor Coolant System and Engineered Safety Features [5] for the design approach to the habitability related systems within and interfacing with the MCR and RSF.
- See Part B Chapter 4 [7] for the design approach to the I&C systems within and interfacing with the MCR and RSF.
- See Part B Chapter 5 [8] for the design approach to the HVAC within and interfacing with the MCR and RSF.
- See Part B Chapter 6 [9] for the design approach to the electrical systems within and interfacing with the MCR and RSF.
- See Part B Chapter 10 [10] for the design approach to the fire systems within and interfacing with the MCR and RSF.
- See Part B Chapter 17 [19] for the design approach to human factors for those systems within and interfacing with the MCR and RSF.
- See Part B Chapter 20 [15] for the design approach to the civil structures which house and protect the MCR and RSF.

Hazards and faults with the potential to impact the SSCs located in the MCR and RSF are identified and safety functions are appropriately incorporated in the design

This argument aims to demonstrate that the hazards and faults with the potential to impact MCR and RSF are identified and that the effect of hazards and faults is minimised. The safety analysis approach for the MCR and RSF is not considered to be different to any other plant area and follows the safety principles set out in Part A Chapter 2. Signposting is also provided below for the interfacing Chapters, where further detail on the approach for each safety analysis discipline is provided (noting that at PSR stage the maturity of any evidence is limited).

- See Part B Chapter 14 [24] for the approach and analysis to demonstrate that SSCs in the MCR and RSF remain available, to provide adequate lines of defence against faults.

- See Part B Chapter 15 [18] for the approach and analysis to demonstrate that SSCs in the MCR and RSF remain available, to provide adequate DiD against severe accidents.
- See PSR Part B Chapter 21 External Hazards [73] for the approach and analysis to demonstrate that SSCs in the MCR and RSF remain available, to mitigate the consequences of external hazards (including combinations of hazards and extreme external hazards).
- See PSR Part B Chapter 22 Internal Hazards [26] for the approach and analysis to demonstrate that SSCs in the MCR and RSF remain available, to prevent or mitigate the consequences of internal hazards.
- See Part B Chapter 17 [19] for the approach and analysis to demonstrate that where measures to protect SSCs located in the Main Control MCR and RSF are human based, any human actions can be appropriately carried out.

Impacts to the habitability of the MCR and RSF are identified, and safety measures are in place to ensure habitability is maintained

This argument aims to demonstrate that the habitability of the MCR is ensured for all required faults and hazards and transfer to the RSF is available should the MCR become uninhabitable (e.g. in the event of fire within the MCR). Signposting is also provided below for the interfacing Chapters, where further detail is provided (noting that at PSR stage the maturity of any evidence is limited).

- See Part B Chapter 14 [24] for the approach and analysis to demonstrate that the MCR and RSF will remain available in the event of relevant faults.
- See Part B Chapter 15 [18] for the approach and analysis to demonstrate that the MCR and RSF will remain available in the event of severe accidents and discussion of where transfer to the RSF will occur, should the MCR become uninhabitable.
- See Part B Chapter 21 [73] for the approach and analysis to demonstrate that the MCR and RSF will remain available in the event of relevant external hazards (including combinations of hazards and extreme external hazards).
- See Part B Chapter 22 [26] for the approach and analysis to demonstrate that the MCR and RSF will remain available in the event of relevant internal hazards.
- See Part B Chapter 10 [10] for the approach and analysis to demonstrate that the MCR and RSF is not rendered uninhabitable in the event of radiological releases as a consequence of accidents.
- See Part B Chapter 1 [5] for the Main Control Room Habitability System (MCH) which provides an environment that is safe and comfortable for human occupancy in the MCR by maintaining temperature, pressure, and humidity in the MCR.

Operational procedures, staffing and training will ensure that the MCR and RSF is operated in accordance with the assumptions and requirements of the safety assessment.

This argument aims to demonstrate that an appropriate approach is being used to develop operational arrangements, to ensure that the activities conducted from the MCR and RSF can be safety carried out. The approach to development of operational arrangements for the MCR and RSF is not considered to be different to any other plant area and follows the approach outlined in Part B Chapter 9. Signposting is also provided below for the interfacing Chapters, where further detail is provided (noting that at PSR stage the maturity of any evidence is limited).

- Staffing and qualification is discussed further in sub-chapter 9.8.4. See Part B Chapter 17 [19] for the full consideration of Human Factor Integration (HFI) within the design of procedures and operating documentation, staffing decisions and training.

Key design decisions for the MCR and RSF will be suitably justified to reduce risks to ALARP

In future safety reports, further information will be included under this argument to provide details of the overall MCR and RSF design and how an appropriate balance has been struck between any competing requirements, to ensure that overall design development of the MCR and RSF reduces risks to ALARP. This is likely to include further details on aspects such as:

- Decisions to share MCR and RSF facilities between two Units.
- Decisions around staffing arrangements.
- Decisions concerning the chosen metrication approach and impact on operational activities in the MCR and RSF.

It is also noted that Part A Chapter 5 [4] of this PSR considers the holistic risk-reduction process for the Generic SMR-300. The process for the assessment of further risk reduction options is presented in Holtec SMR-300 Generic Design Assessment Reference Design Process and GDA Prospective Design Change Register [74].

9.8.4 Staffing and Qualification

The generic SMR-300 HFE program [71] includes a staffing and qualification analysis, whose purpose to determine the required number and necessary qualifications of personnel to complete tasks allocated to people, while meeting all regulations. The staffing analysis includes the full range of plant operating modes such as startup, normal operations, shutdown, abnormal conditions, and emergency conditions. The scope of personnel included in the evaluation is limited to licenced control room operators.

Other personnel will be analysed by the licensee and this aspect is covered in PSR Chapter A4 [3].

Staffing and qualification will be demonstrated fully in the nuclear site licensing phase in accordance with Licence Conditions 10 [75] , 12 [76] and 36 [77].

9.8.5 CAE Summary

The design approach for those SSCs which comprise the MCR and RSF are set out in relevant engineering PSR Chapters. These individual engineering Chapters provide supporting arguments (e.g. adoption of relevant good practice, DiD) to ensure relevant MCR and RSF safety functions can be delivered. The key design decisions for the MCR and RSF will be suitably justified to reduce risks to ALARP.

Similarly, the safety analysis PSR Chapters set out the approach to ensure hazards and faults with the potential to impact the SSCs located in the MCR and RSF are identified and safety functions to protect and mitigate these are incorporated in the design. Safety analysis PSR Chapters of PSR will also demonstrate that habitability of the MCR or RSF can be achieved in accident conditions and under which scenarios transfer to the RSF is required.

Human Factors has a key interface with both the design process and associated safety analysis for the MCR and RSF. In addition, Human Factor Integration (HFI) is necessary to support the design of procedures and operating documentation, staffing decisions and training to ensure the MCR and RSF provides an appropriate means of managing operational activities.

The SMR-300 will be operated from the MCR in accordance with Technical Specifications to ensure that the plant is operated within the limits and conditions of the safety case.

The staffing levels and qualification requirements of the licenced control room operators will be defined for the full range of plant operating modes, including emergency conditions.

The generic SMR-300 HFE program [71] includes a staffing and qualification analysis, whose purpose to determine the required number and necessary qualifications of personnel to complete tasks allocated to people, while meeting all regulations. The staffing analysis will include the full range of plant operating modes such as startup, normal operations, shutdown, abnormal conditions, and emergency conditions. The scope of personnel included in the evaluation is limited to licenced control room operators.

The staffing and qualification requirements of other plant personnel will be defined by the licensee during the nuclear site licensing phase. This aspect is addressed within Chapter A4 [3].

9.9 EXAMINATION, INSPECTION, MAINTENANCE AND TESTING

Claim 2.3.1.7: The Examination, Inspection, Maintenance and Testing (EIMT) activities to ensure the SSCs continue to achieve their safety functional requirements throughout operational life are identified.

EIMT is required to maintain the availability of SSC during the service life by controlling wear and tear with the aim of minimising or preventing failures. If failures do occur, maintenance activities shall be conducted to restore the capability of failed SSC to their required functionality.

Procedures that include all EIMT activities shall be established for the management of EIMT. The management of EIMT is consistent with the deterministic safety analysis, probabilistic safety assessment, SSC design and other safety case requirements. In addition, the management of EIMT considers environmental protection in EIMT activities.

This subchapter outlines the EIMT activities that are relevant for the generic SMR-300 at the GDA stage. It is important to ensure that the approach to EIMT is embedded within the design process to ensure that it can be conducted, as required, to ensure that the SSCs continue to achieve their safety functional requirements throughout the operational life.

The definition of EIMT is conducted at different stages following completion of the design, such that it is conducted:

- By the correct personnel.
- Using the correct equipment and methodology.
- At the optimum frequency to satisfy the requirements of the SSEC, reactive inspections and how guidance on the implementation of the inspector's recommendations shall be developed.
- In accordance with a defect categorization scheme that ensures nuclear safety risk is managed ALARP. For example, a defect that has a nuclear safety implication shall be recorded as a Category A or Category 1 defect and will be actioned immediately.
- In accordance with QA arrangements, including in a manner that will demonstrate that the maintenance regime is appropriately risk informed (eg to manage ageing and degradation through life).

These aspects are undertaken by the Licensee during the nuclear site licensing phase.

Human Factors is crucial to ensuring that EIMT can be conducted, further information on the implementation of HF across the design is provided in PSR Part B Chapter 17 [19]. The purpose of this guide is to assist designers in ensuring that human capabilities and limitations are considered during the design and layout of equipment, including all aspects of EIMT. The aim is to maximise efficiency and safety and minimise errors during maintenance, inspection, and testing activities once the equipment is installed. Designing for field personnel safety and performance requires considering HF in the design of plant equipment layout, equipment components and tools used by personnel, and measures for the protection of personnel, tools, and equipment.

Each of the four areas covered by EIMT is discussed in the subchapters 9.9.1, 9.9.2, 9.9.3 and 9.9.4, including guidance to designers where EIMT considerations should be addressed within the design development. Subchapters 9.9.5 to 9.9.9 identifies EIMT activities that are conducted beyond the GDA stage.

9.9.1 Examination

- SSCs are to receive regular and systematic examination as determined by their safety cases and classification.
- Examination should take place at the manufacture stage to ensure a high standard of parts as determined by their safety cases and classification.
- Both pre-service and in-service examination is carried out to detect and characterise defects at a stage before they could develop to cause gross failure as determined by their safety cases and classification.
- The need for examination of SSCs affects the plant layout and means access arrangements are required.
- Welds and other features that will require examination are not placed within civil structures or so close to other features that inspection is prevented.
- Provision should be made for examination that is capable of demonstrating with suitable reliability that the component or structure has been manufactured to an appropriate standard and will be fit for purpose at all times during future operations.
- Methods of examination of components and structures should be sufficiently redundant and diverse.
- The design of non-metallic components or structures should include the ability to examine the item through life for signs of degradation.

9.9.2 Inspection

The generic SMR-300 is designed to simplify inspections by relying on operating experience and lessons learnt from six decades of operating nuclear power plants. This results in design decisions such as minimizing the number of vessel welds to make them readily inspectable. The plant layout is also affected by the consideration of inspection tasks, as adequate space must be provided for access of personnel and machinery.

Large plant structures (such as the CES) are equipped with ladders and platforms to allow for easy and safe inspection. In order to allow inspection and maintenance within the Annular Reservoir, a monorail supported by the CES inner wall is used for equipment movement.

The inspection of SSCs comply with quality standards and criteria that are appropriate based on the importance of their intended functions (including components with high reliability claims). Some of the codes and standards identified through the SDDs are as follows:

- American Society of Mechanical Engineers (ASME) BPV Section XI [78].
- ASME BPVC Section III [79].
- American National Standards Institute (ANSI)/ American Nuclear Society (ANS)-55.4 [80].
- US NRC Regulatory Guide 1.143 [81].
- ANSI/ANS-55. [82].
- Institute of Electrical and Electronic Engineers (IEEE) 338 [83].

- IEEE 450 [84].
- ASME Section VIII (UW-46) [85].
- ASME XI [86]

9.9.3 Maintenance

The plant is designed so that the need for maintenance is eliminated where possible. For example, the pressuriser design eliminates the need for power-operated relief valves, reducing the chances of RCS leakage and meaning there is no need for maintenance of these valves. However, it is important that preventative and corrective measures are put in place to detect and mitigate the degradation of functioning SSCs or to restore a failed SSC to an acceptable level of performance. This means appropriate measures must be put in place for accessing SSCs (e.g., hatches, ladders etc.) to allow maintenance activities to be carried out when required.

A Guaranteed Shutdown State (GSS) during shutdown operations is required to support safe maintenance activities. As a prerequisite to any maintenance or refuelling activity inside containment, operators will confirm through various means that the reactor is subcritical. The employment of DiD throughout the plant ensures that subcriticality is maintained given that control rod assemblies cannot be inadvertently removed from the reactor core, therefore supporting safe maintenance activities.

The design specifications of the plant support achieving plant design life objectives and planned maintenance schedules.

9.9.4 Testing

The arrangement of periodic testing ensures that the safety functions and environment protection functions of SSCs continuously and reliably achieve the design intent within the plant.

The goal of testing is to define a comprehensive list of tests applicable to a given system. Each testing procedure must include scope, content, frequency, applicable operating mode, and criteria to be met.

Some of the testing requirements for each system are set out within the System Design Descriptions, however inspection and testing programs will be developed during detailed design.

SSCs will be designed to allow periodic testing to be carried out. Sufficient space will be provided for personnel, (remote) inspection equipment, removal space, temporary storage, and handling machinery to aid this.

The following codes and standards have been outlined for testing to be carried out in accordance with:

- ANSI/ANS-55.6-1993 [87].
- ANSI/ANS-55.4-1993 [80].
- US NRC, “Regulatory Guide 1.143” [81].
- ASME BPVC Section III [79].

- ASME BPVC Section XI [78].
- ANSI/ISA S7.0.01 [88].
- Candu Energy Inc., SMR160-03650-DG-001/HPP-160-3011, Design Standard for Containment Isolation Requirements, Rev. 0, July 2018.
- Sheet Metal and Air Conditioning Contractors National Association Inc. (SMACNA) Standards.
- IEEE C57.12.00 [89].
- Utility Requirements Document (URD) Volume III, Chapter 11 Section 2.9.1 [90].
- IEEE 450 [84].
- IEEE 384 [91].
- National Fire Protection Association (NFPA) 70 [92].
- NFPA 110 [93].
- ASME Section VIII (UW-46) [85].
- ASME Section III/CSA N285.5 [94].
- Regulatory Guide 1.197, "Demonstrating Control Room Envelope Integrity at Nuclear Power Reactors" [95].
- U.S. Code of Federal Regulations (CFR) 10 CFR 50, App. J, Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors [96].

Sub-chapter 9.5.2 outlines that the generic SMR-300 must be capable of meeting all 36 licence conditions set out by the ONR within their Licence Condition (LC) handbook [38]. UK standards will be considered at the licencing stage however confidence in the generic SMR-300 testing regime is provided by these currently identified international codes and standards.

9.9.5 Inspection Personnel

Prior to the inspections taking place and once the Project Team members have been confirmed, confirmation of the names and contact details of the person fulfilling each of the project roles shall be issued to everyone in the Project Team. This ensures that the Project Team are aware of who is responsible for each element of the project and who to contact should the need arise.

The Licensee shall ensure that all inspection work will be supervised and executed by Suitably Qualified Experience Personnel (SQEP) who will ensure that the inspections are adequate and that any defects are resolved appropriately in accordance with and in consideration of the SSEC.

9.9.6 Defect Categorisation

A process for categorisation of defects shall be developed to ensure consistency of reporting and assigning action. Defect categorisation shall be in the form of a letter or numbering system.

Defect examples and their corresponding categorisation shall also be provided to assist the Inspectors and ensure consistency.

9.9.7 Inspection Guidance

A list of guidelines to be followed when conducting inspections will be included to assist the Inspectors.

Items such as protective coatings, sealing materials, etc. shall also be included in the scope of inspections.

9.9.8 Frequency of Inspections

To comply with the requirements detailed in LCs, a series of inspection routines shall be established for the assets and infrastructure. The frequency of inspections will be consistent with the requirements of the safety case as determined by fault studies. These inspection routines shall be completed at defined intervals. The frequency of the intervals may also be informed on a risk basis, taking cognizance of the assets and infrastructure's qualifications for hazards.

It should be noted that routine maintenance, including preventative maintenance, of the assets and infrastructure should be undertaken at regular intervals between inspections. Maintenance comprises any work, including inspection, cleaning, painting, etc., necessary for the civil asset and infrastructure to continue its intended safety function, or to sustain its original or required standard of appearance. Preventative maintenance is any work necessary to prevent the civil asset and infrastructure deteriorating to such a degree that its intended function is impaired. This will also ensure that the deterioration of any defects is not accelerated due to neglect and will also help to completely avoid new defects or reduce the severity or extent of new defects.

9.9.8.1 Reactive Inspections

The site shall develop a guidance document that details when inspections shall be undertaken after defined internal and external hazards.

Trigger levels shall be provided, and inspections undertaken when the trigger levels are met or exceeded.

9.9.9 Ageing and Degradation

This subchapter outlines the ageing and degradation elements that are relevant to the generic SMR-300. As the generic SMR-300 design develops, the EIMT and Ageing and Degradation programme shall be developed, however, this is not a requirement of GDA and will be developed during the nuclear site licensing phase.

The ageing and degradation of components within the generic SMR-300 during operation is unavoidable, however, suitable maintenance regimes shall be established to minimise and control potential impacts of such degradation leading to failure. Should degradation of components result in their failure, procedures are to be in place to ensure disruption and downtime is minimised and normal operations are restored. Detection, analysis and managing of ageing effects on SSCs is essential and enables the related reduction in safety margins to be addressed by corrective actions before loss of integrity or functional capability of SSCs.

The design of the generic SMR-300 is such that degradation and ageing of SSCs is considered. The monitoring and inspection process, to identify and monitor the degradation of SSCs, is outlined within subchapter 9.8. This process helps to ensure that SSC are able to perform their required safety function when required.

Detection, analysis, and management of ageing effects on a given SSC is an important part of maintaining plant safety. This enables addressing the related reduction in safety margins by corrective actions before loss of integrity or functional capability of SSCs. The monitoring and management of SSC degradation is required to ensure that the safety functions and environment protection functions of those SSCs can be ensured as part of plant operational management.

Further design development will stipulate a time period in which material properties and other parameters of SSCs must be tested and measured against the pre-determined design criteria. This will help to identify degradation before failure.

LC15 [97] is central to managing ageing and degradation. LC15 requires the licensee to ensure that throughout its declared lifetime, the plant remains adequately safe and that the safety cases are kept up to date. Towards this end, the condition of the plant and the currency and adequacy of the extant safety cases should be periodically reviewed.

9.9.10 Implementation of Recommendations

Guidance on the implementation of the Inspector's recommendations shall be developed taking account of the Licensee' requirements and procedures.

This may include guidance on how the defects are recorded, such as placing tags at the defect location and completing action or work request forms.

A site-specific reference system should be established and used to track defects from initial identification to action planning, through the implementation phase to completion and close out.

Action plans shall detail the responsibilities of each of the parties involved in the implementation of the recommendations. It may state that a meeting shall take place with all the responsible parties to discuss the defect and agree how the defect shall be resolved.

9.9.11 Quality Assurance

Reference to the Licensee' Quality Assurance (QA) documents shall be provided to ensure that the inspections and the inspection reports are prepared in accordance with the approved QA requirements.

QA requirements shall be stated to ensure that organisations undertaking the inspection work shall operate a Quality Management system to an appropriate standard such as BS EN ISO 9000. Chapter A4 [3] provides more detail and information on the Quality Assurance of the generic SMR-300.

9.9.12 CAE Summary

This subchapter outlines the EIMT activities that are relevant for the generic SMR-300 at the GDA stage. At the GDA stage, it is important to ensure that the approach to EIMT is embedded within the design process to ensure that it can be conducted, as required, to ensure that the SSCs continue to achieve their safety functional requirements throughout the operational life.

The HFE programme is embedded within the generic SMR-300 design process to ensure that EIMT considers human capabilities and limitations, to maximise efficiency and safety and minimise errors during maintenance, inspection, and testing activities once the equipment is installed. This subchapter outlines guidance for the need for examination of SSCs is embedded within the plant design layout, such that examination and inspection can be undertaken, and that SSCs will be designed to allow periodic testing to be carried out.

Guidance is in place to ensure that it will be demonstrated that redundant and diverse examination techniques are available to demonstrate the SSC to a suitable reliability through life.

The generic SMR-300 is designed to simplify inspections by relying on operating experience and lessons learnt. This results in design decisions such as minimizing the number of vessel welds to make them readily inspectable. The plant layout is also affected by the consideration of inspection tasks, as adequate space must be provided for access of personnel and machinery.

The generic SMR-300 is designed so that the need for maintenance is eliminated where possible. For example, the pressuriser design eliminates the need for power-operated relief valves, reducing the chances of RCS leakage and meaning there is no need for maintenance of these valves. However, it is important that preventative and corrective measures are put in place to detect and mitigate the degradation of functioning SSCs or to restore a failed SSC to an acceptable level of performance. This means appropriate measures must be put in place for accessing SSCs (e.g., hatches, ladders etc.) to allow maintenance activities to be carried out when required.

There are EIMT aspects that are conducted at different stages following completion of the design, that are undertaken by the Licensee during the nuclear site licensing phase. A summary of the EIMT principles for these lifecycle stages are provided.

An ageing and degradation programme will be undertaken during the nuclear site licensing phase for the generic SMR-300 and is not a GDA requirement.

At the GDA stage, EIMT is being considered within the generic SMR-300 design process to ensure that the EIMT activities to ensure the SSCs continue to achieve their safety functional requirements throughout operational life are identified, and that the design enables their demonstration.

9.10 CHAPTER SUMMARY AND CONTRIBUTION TO ALARP

This sub-chapter provides an overall summary and conclusion of the Description of Operational Aspects and Conduct of Operations Chapter and how this Chapter contributes to the overall demonstration of ALARP for the generic SMR-300. PSR Part A Chapter 5 [4] sets out the overall approach for demonstration of ALARP and how contributions from individual Chapters are consolidated.

This subchapter therefore consists of the following elements:

- Technical Summary.
- ALARP Summary
  - Review against relevant RGP;
  - Risk Reduction Options;
  - GDA Commitments and Forward Actions.
- Conclusion.

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

9.10.1 Technical Summary

This chapter directly supports Claim 2.3.1.

Claim 2.3.1: Appropriate arrangements to safely manage people and plant during the construction, commissioning and operation of the Generic Holtec SMR-300 are suitably mature for a generic design.

Claim 2.3.1 has been further decomposed within Part B Chapter 9 to provide confidence that the relevant requirements for operational aspects and conduct of operations will be met.

The SMR-300 conduct of operations is derived based upon identified good practice, including US NRC NUREG and IAEA guidance to identify all operational modes and plant configurations. The definition of the conduct of operations and ensuring that safety case requirements are captured by operating documentation will be fully developed within the nuclear site licensing phase of the generic SMR-300, and not within the GDA process.

The SSEC will enable the derivation of ORs (limits and conditions) in accordance with ONR TAG-35 [45] during the nuclear site licensing phase, resulting in operating limits and conditions, to inform the production of the Technical Specifications for the purposes of compliance with Licence Condition 23. The SSEC will be expected to be developed beyond this GDA to determine the ORs required for safe operation that form the operating envelope for the generic SMR-300.

ORs link the safety case analysis and assumptions with actual operational limits and conditions in force at the facility. The SSEC:

- DBA and Fault Studies will determine the limits of normal operation, safety settings, design basis limits and (in conjunction with the engineering analysis) safety limits; Inform the required availability of safety measures; and provide significant insight against the expectations of the DiD framework.
- Engineering substantiation will provide limits on operation to prevent fault initiation or escalation; ensure design assumptions and intent are met; set conditions on appropriate plant and equipment configurations; specify the timing of maintenance and testing activities; cater for plant ageing and corrosion effects; and set ORs relating to equipment qualification.
- PSA will inform EIMT related ORs and proof testing intervals, and also determining ORs governing the availability of safety measures for deriving time-based ORs, those relating to allowed substitution periods and the unavailability of safety measures, so that any periods of elevated risk may be suitably justified by the Licensee and be a key input to holistic reviews of ORs.

Operating limits and conditions are subject to plant-specific development and on-going maintenance throughout the whole plant lifetime. The whole set of the operating documentation relies on the design requirements and the licensee's experience and practice and will be finalised in the nuclear site licensing phase.

Operating procedures will be developed during the nuclear site licensing phase utilising the limits and conditions derived by the SSEC. Procedures are essential to plant safety because they support and guide personnel interactions with plant systems and personnel responses to plant-related events. In the nuclear industry, procedure development is the responsibility of the licensee. The HFE programme will ensure that the procedures developed by the requesting party will follow accepted HF engineering principles and will ensure that the limits and conditions of the safety case will be incorporated into operational documentation.

Technical Specifications for the generic SMR-300 CPA will be developed in accordance with 10 CFR 50.34(a)(5) [23]. Technical Specifications do not need to be fully developed in accordance with 10 CFR 50.36 until the Operating License Application OLA.

The generic SMR-300 Technical Specifications will inform the development of operating documentation in accordance with LCs 23 [39], 24 [40] and 25 [41] during the nuclear site licensing phase. The Tech Specs will be based upon the limits and conditions derived by the SSEC.

The MCR is being designed to provide a habitable area from which to safely operate the reactor. The RSF is a control station which serves as a backup to the MCR, should it be inoperable or require evacuation. It is shown that the overall design of the MCR and RSF provides an appropriate means of managing operational activities and delivery of safety functions.

The design approach for the SSCs associated with the MCR and RSF is appropriate and supports delivery of relevant safety functions. The following are demonstrated by providing a route map into the wider PSR:

- Hazards and faults with the potential to impact the SSCs located in the MCR and RSF are identified, and safety functions are appropriately incorporated in the design.
- Impacts to the habitability of the MCR and RSF are identified, and safety measures are in place to ensure habitability is maintained.
- Operational procedures, staffing and training will ensure that the MCR and RSF is operated in accordance with the assumptions and requirements of the safety assessment.
- Key design decisions for the MCR and RSF will be suitably justified to reduce risks to ALARP.

The SMR-300 will be operated from the MCR in accordance with Technical Specifications to ensure that the plant is operated within the limits and conditions of the SSEC.

The operating and technical support organisation will be defined and adequately managed to enable and ensure safe operation of the plant.

- Staffing and qualification analyses will be undertaken for the full range of plant operating modes such as startup, normal operations, shutdown, abnormal conditions, and emergency conditions for licenced control room operators, to enable the demonstration that the plant will be operated safely in accordance with the limits and conditions of the safety case.
- The staffing and qualification requirements of other plant personnel will be defined by the licensee during the nuclear site licensing phase, and adequate arrangements will be in place. This aspect is addressed within Chapter A4 [3]. The requirements of UK Licence conditions will be considered at this point.

EIMT is required to maintain the availability of SSC during the service life by controlling wear and tear with the aim of minimising or preventing failures. If failures do occur, maintenance activities shall be conducted to restore the capability of failed SSC to their required functionality. The EIMT activities that are relevant for the generic SMR-300 at the GDA stage have been identified. It is important to ensure that the approach to EIMT is embedded within the design process to ensure that it can be conducted, as required, to ensure that the SSCs continue to achieve their safety functional requirements throughout the operational life.

The HFE programme is embedded within the generic SMR-300 design process to ensure that EIMT considers human capabilities and limitations, to maximise efficiency and safety and minimise errors during maintenance, inspection, and testing activities once the equipment is installed.

Guidance is provided to provide ensure that examination and inspection can be undertaken, utilizing redundant and diverse examination techniques to a suitable reliability, and that SSCs will be designed to allow periodic testing to be carried out.

The generic SMR-300 is designed to simplify inspections by relying on operating experience and lessons learnt. This results in design decisions such as minimizing the number of vessel welds to make them readily inspectable. The plant layout is also affected by the consideration of inspection tasks, as adequate space must be provided for access of personnel and machinery.

The generic SMR-300 is designed so that the need for maintenance is eliminated where possible. For example, the pressuriser design eliminates the need for power-operated relief valves, reducing the chances of RCS leakage and meaning there is no need for maintenance of these valves. However, it is important that preventative and corrective measures are put in place to detect and mitigate the degradation of functioning SSCs or to restore a failed SSC to an acceptable level of performance. This means appropriate measures must be put in place for accessing SSCs (e.g. hatches, ladders etc.) to allow maintenance activities to be carried out when required.

There are EIMT aspects that are conducted at different stages following completion of the design, that are undertaken by the Licensee during the nuclear site licensing phase. A summary of the EIMT principles for these lifecycle stages are provided.

An ageing and degradation programme will be undertaken during the nuclear site licensing phase for the generic SMR-300 and is not a GDA requirement.

Operational aspects and conduct of operations constitute safety case implementation and as such, is largely undertaken by the Licensee following the Pre-Construction Safety Report stage. It will therefore be fully developed within the nuclear site licensing phase of the generic SMR-300 and not within the GDA process.

9.10.2 ALARP Summary

PSR Part A Chapter 5 [4] presents the totality of the generic SMR-300 design against the ALARP principles. This section presents the ALARP considerations specific to this topic area. The methodology for this is discussed in Chapter A5 [4] and consists of two main elements:

- Design complies with RGP.
- Options to reduce the risk have been identified and implemented where the effort to do so is not grossly disproportionate to the benefit that would be realised.

The following subchapters present the ALARP considerations for the Operational Aspects and Conduct of Operations.

9.10.2.1 Demonstration of RGP

The RGP codes and standards for the conduct of operations has been identified throughout this chapter. By considering this at an early stage, the conduct of operations is likely to result in an operational programme that can be demonstrated to be ALARP.

Where shortfalls exist, this will either be:

1. Because the design does not meet UK RGP.
2. At this time, it cannot be concluded whether the design does or does not meet UK RGP.

In both Instances, FAs will be being raised to ascertain how and when these shortfalls will be addressed. If it is the intent not to demonstrate compliance at some future point, the argument as to why this is the case will be presented. Argument statements should be based around the following items:

- Activities required.
- Effort required in terms of time/cost/design impact etc.
- Respective benefit gained by these efforts.
- Risk carried forward if a particular activity is not undertaken.

These arguments may then be used to consider why activities are acceptable and meet ALARP.

9.10.2.2 Risk Reduction Option Review

Holtec implements a robust configuration control process which also includes a robust consideration of key design decisions, to ensure the impact of such decisions are fully understood. The evaluation of design decisions is undertaken in accordance with the Holtec 'Design Decisions' procedure [98]. Guidance is provided to categorise decisions depending upon their significance and to provide the appropriate level of documentation and approval required for each design decision category.

However, as would be expected for processes developed for a non-UK regulatory regime, there is no specific mention of ALARP or other UK context considerations (e.g., BAT, Construction Design and Management (CDM) etc.) within the existing processes. Therefore, it will be important to develop proportionate arrangements for the GDA process to ensure consideration of Risk Reduction Measure (RRM) options through the process of:

- Identification and evaluation of options (optioneering).
- Risk assessment, as a way of understanding the significance of the issue to the overall demonstration of ALARP.
- Implementation of all reasonably practicable improvements.

The ALARP Design Process [99] provides further detail of how the current US design process interfaces with the requirement to demonstrate compliance with the ALARP principle and highlights options to strengthen this interface. Further details will be set out in the UK GDA project quality arrangements, provided as part of the submission.

The key ALARP consideration at the GDA stage is that the key design decisions for the MCR and RSF will be suitably justified to reduce risks to ALARP.

9.10.2.3 GDA Commitments and Forward Actions

Forward Actions have been collated and are managed via the process described in PSR Chapter A4 [3]. PSR Chapter A5 [4] describes the contribution of the forward actions to the ALARP argument.

A UK categorisation and classification scheme for the generic SMR-300 safety functions is required to be presented for normal operations and fault conditions for all aspects of the Generic SMR-300 within the GDA scope. This exercise will identify all SSCs with a normal operation safety function and will inform the full scope of normal operation SSCs to be included within this chapter at PSR Revision 1 to provide a route map to the where the limits and conditions associated with these SSCs will be captured within the relevant PSR engineering chapters. The list of SSCs which are both in scope and essential to normal operations will be updated at PSR Revision 1 upon completion of GDA.

9.10.3 Conclusion

The conclusion of this Chapter of the PSR is that:

- The Chapter Claims identified have been met to a maturity aligned with a preliminary safety report. Further claims, arguments and evidence will be presented in due course as the design develops.
- This chapter outlines the approach to plant operations and the undertaking of operations. These cannot be outlined in full at the GDA stage, as they are developed by the licensee in the site licensing phase, based on their specific operating requirements, experience and practice.
- The generic SMR-300 conduct of operations is derived based upon identified good practice, including US NRC NUREG and IAEA guidance.
- The SSEC will enable the derivation of ORs (limits and conditions) in accordance with ONR TAG-35 [47] during the nuclear site licensing phase, resulting in operating limits and conditions, to inform the production of the Technical Specifications for the purposes of compliance with Licence Condition 23.
- The HFE programme will ensure that the procedures developed by the requesting party will follow accepted HF engineering principles and will ensure that the limits and conditions of the safety case will be incorporated into operational documentation, and that appropriate staffing and qualification analyses will be undertaken for the full range of plant operating modes.
- The generic SMR-300 Technical Specifications will inform the development of operating documentation in accordance with LCs 23 [41], 24 [42] and 25 [43] during the nuclear site licensing phase. The Tech Specs will be based upon the limits and conditions derived by the SSEC.
- The MCR is designed to provide a habitable area from which to safely operate the reactor. The RSF is a control station which serves as a backup to the MCR, should it be inoperable or require evacuation. It is shown that the overall design of the MCR and RSF provides an appropriate means of managing operational activities and delivery of safety functions.
- The HFE programme is embedded within the generic SMR-300 design process to ensure that EIMT considers human capabilities and limitations, to maximise efficiency and safety and minimise errors during maintenance, inspection, and testing activities once the equipment is installed.
- The generic SMR-300 is designed to simplify inspections by relying on operating experience and lessons learnt.
- Operational aspects and conduct of operations constitute safety case implementation and as such, is largely undertaken by the Licensee following the Pre-Construction Safety Report stage. It will therefore be fully developed within the nuclear site licensing phase of the generic SMR-300 and not within the GDA process.

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Appendix A CAE Route Map

Table 4: CAE Route Map for Conduct of Operations

Corresponding Overarching SSEC Claim	PSR Chapter Claim	PSR Chapter Sub Claim(s)	Chapter Section
Claim 2.3 - Lifecycle The design and safety assessment of the generic SMR-300 considers the entire reactor lifecycle.	**Claim 2.3.1 - Construction, Commissioning and Operation** Appropriate arrangements to safely manage people and plant during the construction, commissioning and operation of the generic SMR-300 are suitably mature for a generic design.	**Sub- Claim 2.3.1.3**: The SMR-300 conduct of operations is derived based upon identified international good practice.	**9.5 CODES, STANDARDS AND REGULATIONS**
		Sub- Claim 2.3.1.4: The limits and conditions of the safety case are clearly identified, covering all operational modes and plant configurations.	**9.6 OPERATING LIMITS AND CONDITIONS**
		Sub- Claim 2.3.1.5: The limits and conditions of the safety case will be incorporated into operational documentation.	**9.7 OPERATING PROCEDURES**
		Sub- Claim 2.3.1.6: The plant will be operated safely in accordance with the limits and conditions of the safety case.	**9.8 MAIN CONTRL ROOM AND REMOTE SHUTDOWN FACILITY**
		Sub- Claim 2.3.1.7: The Examination, Inspection, Maintenance and Testing (EIMT) activities to ensure the SSCs continue to achieve their safety functional requirements throughout operational life are identified.	**9.9 EXAMINATION, INSPECTION, MAINTENANCE AND TESTING**