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PSR Part B Chapter 1 Reactor Coolant System and Engineered Safety Features

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1.1 INTRODUCTION

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

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

Part B Chapter 1 of the PSR presents the Claims, Arguments and intended Evidence (CAE) for the Reactor Coolant System (RCS) and Engineered Safety Features (ESFs).

1.1.1 Purpose and Scope

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

This chapter of the PSR, Part B Chapter 1, links to the Overarching Claim through Claim 2.2:

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

As set out in PSR Part A Chapter 3, Claim 2.2 is further decomposed across several engineering disciplines that are responsible for the design and development of relevant Structures, Systems, and Components (SSCs).

This chapter presents the RCS and ESFs, directly supporting the claims made on the overall design and architecture of RCS and ESF SSCs (Claim 2.2.4 and Claim 2.2.5 respectively). Furthermore, these SSCs function to support the three critical safety functions presented through Claim 2.2.1, Claim 2.2.2, and Claim 2.2.3.

Claim 2.2.1: Adequate provision for the control of reactivity is incorporated into the design of the reactor systems, engineered safety features, and fuel and core design.

Claim 2.2.2: Adequate provision for the removal of heat from the reactor core and spent fuel pool is incorporated into the design of the reactor systems, engineered safety features, and fuel and core design.

Claim 2.2.3: Adequate provision for the control of radiation exposure and control of release of radioactive material is incorporated into the design of the reactor systems, supporting facilities, engineered safety features, and fuel and core design.

Claim 2.2.4: The Reactor Coolant System and Connected Systems are designed to ensure they deliver relevant safety features, supported by substantiation which is suitably mature.

Claim 2.2.5: The Engineered Safety Features are designed to ensure they deliver relevant safety features, supported by substantiation which is suitably mature.

Further discussion, on how the Level 3 claims are broken down into Level 4 claims, and how the Level 4 claims are met, is provided in sub-chapter 1.3.

Sub-chapter 1.4 provides a description of the classification and categorisation of SSCs, and the main codes and standards used in the design of these SSCs.

The scope of this chapter covers the RCS and the ESFs, as presented in sub-chapter 1.5 and sub-chapter 1.6 respectively.

Finally, a Technical Summary and a summary of the considerations against the ALARP principle is provided in sub-chapter 1.7, alongside any commitments that have arisen.

A master list of definitions and abbreviations relevant to all PSR Chapters can be found in Part A Chapter 2 [3].

1.1.2 Assumptions

Any assumptions that relate to this topic have been formally captured in the Commitments, Assumptions and Requirements process [4]. Further details of this process are provided in Part A Chapter 4 [5].

1.1.3 Interfaces with Other SSEC Chapters

The RCS and ESFs chapter interfaces with other chapters within the PSR. The following highlights the key interfaces.

Part B Chapter 2 Reactor [6], presents the Reactor Fuel and Core design topic area, interfacing directly with the RCS. The Reactor Pressure Vessel (RPV) is described in this chapter, while the Reactor Internal Structure (RIS) is detailed in Part B Chapter 2.

Part B Chapter 2 provides a description of the Control Rod Drive System (CDS) and Rod Cluster Control Assembly (RCCA), which interface directly with the RPV. The RCCAs support the claim to control reactivity in the core, as a means of control that is diverse and independent from the ESF, Passive Core Cooling System (PCC). Part B Chapter 2 presents a description of the External Core (Ex-Core) Instrumentation (EIS) and In-Core Instrumentation (IIS), which provide core measurements, interfacing with the RPV.

Control and Instrumentation Systems are presented in Part B Chapter 4 [7]. The RCS instrumentation provides data to the Plant Safety System (PSS) and Diverse Actuation System (DAS), which actuates reactor protection functions and ESFs as required.

Part B Chapter 5 [8] presents an overview of the Reactor Supporting Facilities (including the Auxiliary Systems, Steam and Power Conversion Systems, Mechanical Handling Systems, and Heating, Ventilation, and Air Conditioning (HVAC)). Many of the Auxiliary Systems directly interface with the RCS and ESFs, for example, in providing chemical and volume control to the reactor coolant, via the Chemical and Volume Control System (CVC). Similarly, the Residual Heat Removal System (RHR) interfaces with the RCS to remove decay heat from the reactor core and RCS sensible heat to reduce reactor coolant temperature during normal shutdown and refuelling operations. Power Conversion Systems (Main Feedwater System (MFW) and Main Steam System (MSS)) interface with the Steam Generator (SGE) to transfer heat from the primary coolant.

SSCs of the RCS and ESFs have electrical and mechanical aspects to their design, these are described in greater detail in the Electrical Engineering Chapter (Part B Chapter 6 [9]) and the Mechanical Engineering Chapter (Part B Chapter 19 [10]) respectively. SSCs and ESFs, notably PCH, interface with SSCs presented in Part B Chapter 20 Civil Engineering Chapter [11].

Part B Chapter 14 Safety and Design Basis Accident Analysis [12] presents an overview of the Design Basis Accident Analysis (DBAA). The state of the RCS forms some of the Fault Groups used in the analysis. The ESFs are involved in the mitigation of postulated Design Basis Accidents (DBAs).

Part B Chapter 18 Structural Integrity [13] describes the methodology for assessing any High Reliability and Very High Reliability components, some of which are present in the RCS and ESFs.

The water chemistry regime of the RCS is detailed in Part B Chapter 23 Reactor Chemistry [14]. Material selection for RCS and ESF SSCs is dictated by the chemistry regimes present in the design.

Part A Chapter 5 Summary of ALARP [15] concludes that it can be demonstrated that the generic SMR-300 reduces risks to ALARP and that the Fundamental Purpose of the SSEC has been fulfilled.

The Best Available Technique (BAT) demonstration for the SMR-300 generic design has been developed, in line with BAT Approach [16] and GDA Scope [17], to indicate how radioactive waste will be prevented and minimised to reduce the impact on the members of the public and environment As Low As Reasonably Achievable (ALARA). Part B Chapter 1 will help to support the Approach and Application of BAT Demonstration.

Part B Chapter 11, Environmental Protection [18], services the interface between PSR and Preliminary Environmental Report (PER), and mainly summaries the PER and the approach of BAT for the generic SMR-300 GDA, including an overview of legislation, policy, environmental claims, arguments and evidence relevant to this new nuclear development.

1.2 OVERVIEW OF PLANT SYSTEMS

The SMR-300 is an advanced, passively safe, small pressurised light water reactor capable of producing 1050 MW of thermal energy at steady state operating conditions, yielding a minimum of 300 MW of electricity net of house loads. The SMR-300 is designed and engineered with a minimum service life of 80 years. The plant includes passively actuated (i.e. no operator input) and operated safety features, simplifying the design relative to conventional large light water reactors, and making the plant relatively simpler and safer to operate in comparison. Further information about the philosophy of the SMR-300, including passive safety feature, can be found in [3].

1.2.1 Reactor Coolant System

The RCS, as described in [19], has a single heat transfer circuit, consisting of the following subsystems, as seen in Figure 1:

- RPV.
- SGE.
- Reactor Coolant Pumps (RCPs) with associated hot and cold leg pipes.
- Pressuriser (PZR) (integrated to the SGE).

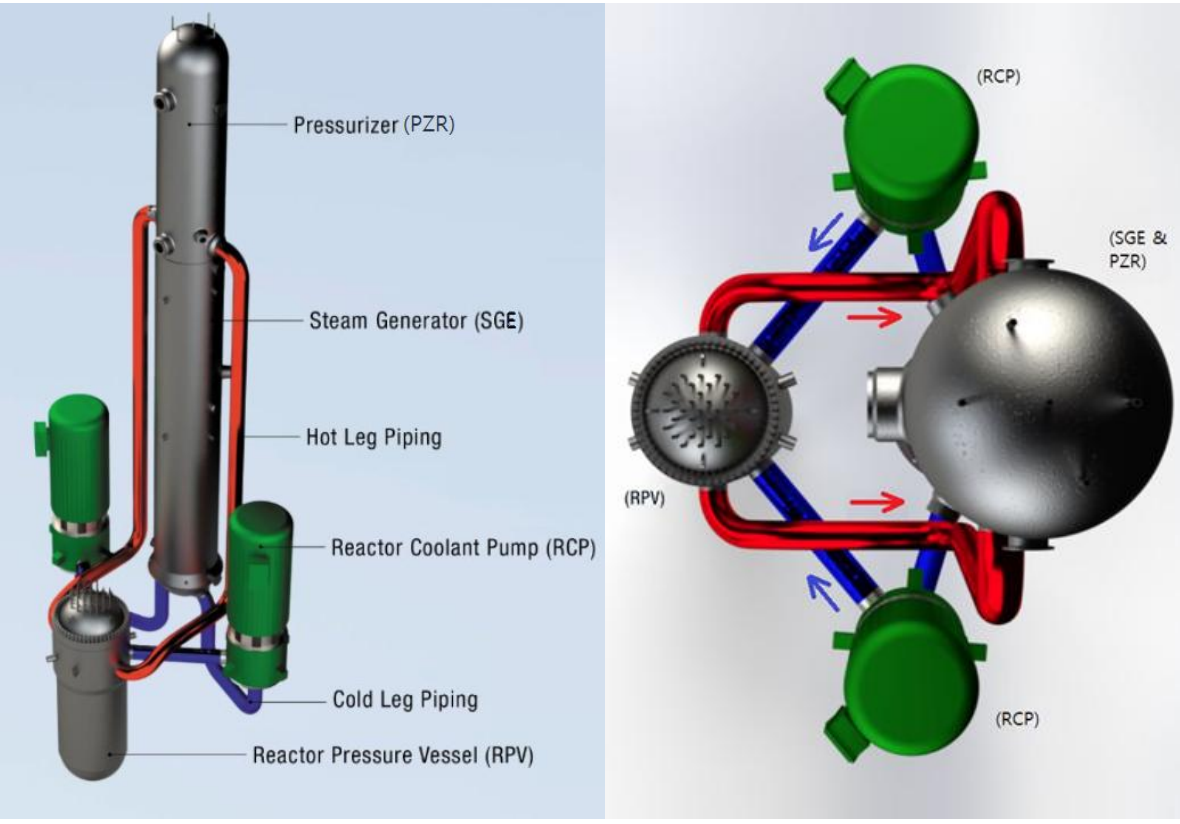

Figure 1: RCS Subsystems

All RCS equipment is located within the reactor Containment Structure (CS). During operation, the RCPs circulate pressurised water through the RPV and SGE. The water serves as coolant, moderator, and solvent for boric acid that provides chemical shim reactivity control. Circulated water is heated as it passes through the core. Upon exiting the core, the heated water is transported to the SGE, where the heat is transferred to the MSS.

1.2.2 Engineered Safety Features

The SMR-300 incorporates multiple levels of defence-in-depth to remove heat from the reactor and assure safety. Most of the safety systems are located inside the robust Containment Enclosure Structure (CES), with the Main Control Room Habitability (MCH) located in the Reactor Auxiliary Building (RAB), rendering them secure and protected from external threats, both natural and man-made.

The ESFs are present in the following systems:

- Passive Containment Heat Removal System (PCH) [20].
- Containment Isolation System (CIS) [21].
- Main Control Room Habitability System (MCH) [22].
    - Breathing Air and Pressurisation System (BAP).
    - Control Room Normal Ventilation System (CRV).
- PCC [23].
    - Automatic Depressurisation System (ADS).
    - Passive Core Makeup Water System (PCM).
    - Primary Decay Heat Removal System (PDH).
    - Secondary Decay Heat Removal System (SDH).

The safety systems are simpler than those included in currently operating large Light Water Reactors, eliminating active pumps, thus making them more reliable. In the case of an accident, no operator actions are required to place and maintain the reactor in a safe shutdown condition. All makeup water needed for a postulated Loss Of Cooling Accident (LOCA) is held inside the CS, allowing the RCS to be fully isolable during such an event, minimising dose to the public and effects on the environment.

1.3 RCS AND ESF CLAIMS, ARGUMENTS AND EVIDENCE

This chapter presents the RCS and ESFs aspects for the generic SMR-300 and therefore supports Claims 2.2.1, 2.2.2, 2.2.3, 2.2.4, and 2.2.5.

Claim 2.2.1: Adequate provision for the control of reactivity is incorporated into the design of the reactor systems, engineered safety features, and fuel and core design.

Claim 2.2.1 has been further decomposed within this chapter (Part B Chapter 1) and Part B Chapter 2 Reactor (Fuel and Core).

Claim 2.2.1 is associated with reactivity control via independent means, and is decomposed into Claim 2.2.1.1, relating to reactivity control via boronated water, and Claim 2.2.1.2, relating to reactivity control via control rods.

Table 1 shows the breakdown of Claim 2.2.1 and identifies in which chapter of this PSR these claims are demonstrated to be met to a maturity appropriate for PSR v1.

Table 1: Claim 2.2.1 breakdown

Claim No.	Claim	Chapter Section
2.2.1.1	There is provision in the design to ensure that the reactor can be shutdown, via boron control in the RCS, in all relevant plant modes.	1.6.3 Passive Core Cooling System
2.2.1.2	Control rods provide appropriate reactivity control during normal reactor operation and the means for reactor shutdown during a trip.	Part B Chapter 2 Reactor

Claim 2.2.2: Adequate provision for the removal of heat from the reactor core and spent fuel pool is incorporated into the design of the reactor systems, engineered safety features, and fuel and core design.

Claim 2.2.2 has been further decomposed within this chapter (Part B Chapter 1) and Part B Chapter 2 Reactor (Fuel and Core).

Claim 2.2.2 asserts the claim of heat removal from the reactor core and spent fuel pool and is decomposed into four sub-claims. Claims 2.2.2.1 – 2.2.2.3 relate to the RCS and ESFs which function to remove reactor core and spent fuel pool heat. Claim 2.2.2.4 proposes that the core design ensures that it is always maintained in a coolable geometry.

Table 2 shows the breakdown of Claim 2.2.2 and identifies in which chapter of this PSR these claims are demonstrated to be met to a maturity appropriate for PSR v1.

Table 2: Claim 2.2.2 breakdown

Claim No.	Claim	Chapter Section
2.2.2.1	Core decay heat removal is ensured following credible initiating events in all plant states.	1.5.2 Reactor Pressure Vessel 1.5.3 Steam Generator 1.5.5 Reactor Coolant Pump 1.6.2 Passive Containment Heat Removal System 1.6.3 Passive Core Cooling System
2.2.2.2	Sufficient coolant inventory is maintained following credible initiating events in all plant states.	1.5.2 Reactor Pressure Vessel 1.6.3 Passive Core Cooling System
2.2.2.3	Spent fuel heat removal is ensured following credible initiating events in all plant states.	1.6.3 Passive Core Cooling System
2.2.2.4	The core is always maintained in a coolable geometry.	Part B Chapter 2 Reactor

Claim 2.2.3: Adequate provision for the control of radiation exposure and control of release of radioactive material is incorporated into the design of the reactor systems, supporting facilities, engineered safety features, and fuel and core design.

Claim 2.2.3 has been further decomposed within this chapter, Part B Chapter 2 Reactor (fuel and core), and Part B Chapter 5 Reactor Supporting Facilities.

Claim 2.2.3 asserts that there is provision to control radiation exposure and the release of radioactive material and is decomposed into six sub-claims. Claims 2.2.3.1 and 2.2.3.4 declare the Reactor Coolant Pressure Boundary (RCPB), and components that comprise the RCPB, as a barrier to radiation release, described within this chapter and Part B Chapter 5. Claims 2.2.3.2 and 2.2.3.5 declare the CS, and components that comprise the CS, as a barrier to radiation release, described within this chapter and Part B Chapter 5. Claim 2.2.3.3 proposes the ability of the MCH to ensure safe habitability of the main control room following credible initiating events. Claim 2.2.3.6 declares the fuel rod cladding as a barrier to radiation release and is described in Part B Chapter 2.

Table 3 shows the breakdown of Claim 2.2.3 and identifies in which chapter of this PSR these claims are demonstrated to be met to a maturity appropriate for PSR v1.

Table 3: Claim 2.2.3 breakdown

Claim No.	Claim	Chapter Section
2.2.3.1[1]	The RCS and ESF SSCs ensure the integrity of the Reactor Coolant Pressure Boundary following credible initiating events in all plant states.	1.5.2 Reactor Pressure Vessel 1.5.3 Steam Generator 1.5.4 Pressuriser 1.5.5 Reactor Coolant Pump 1.5.6 RCS Piping 1.6.3 Passive Core Cooling System
2.2.3.2	The RCS and ESF SSCs ensure the integrity of the Containment Structure following credible initiating events in all plant states.	1.6.2 Passive Containment Heat Removal System 1.6.3 Passive Core Cooling System 1.6.5 Containment Isolation System
2.2.3.3	Habitability of the main control room is ensured following credible initiating events in all plant states.	1.6.4 Main Control Room Habitability System
2.2.3.4[1]	The Reactor Supporting Facilities SSCs ensure the integrity of the Reactor Coolant Pressure Boundary following credible initiating events in all plant states.	Part B Chapter 5 Reactor Supporting Facilities
2.2.3.5	The Reactor Supporting Facilities SSCs ensure the integrity of the Containment Structure following credible initiating events in all plant states.	Part B Chapter 5 Reactor Supporting Facilities
2.2.3.6	The fuel rod clad integrity is maintained during normal operation and Anticipated Operational Occurrences (AOOs).	Part B Chapter 2 Reactor

Claim 2.2.4: The Reactor Coolant System and Connected Systems are designed to ensure they deliver relevant safety features, supported by substantiation which is suitably mature.

Claim 2.2.4 has been further decomposed within this chapter (Part B Chapter 1).

Claim 2.2.4 asserts that the RCS has been designed to ensure it delivers its safety functions and is decomposed into six sub-claims. The six sub-claims relate to the key aspects of the RCS design. Table 4 shows the breakdown of Claim 2.2.4 and identifies in which chapter of this PSR these claims are demonstrated to be met to a maturity appropriate for PSR v1.

[1] It is recognised that LOCAs are a credible design basis fault that challenges integrity of the RCPB. Part B Chapter 14 considers these specific faults to ensure there are appropriate lines of protection when RCPB integrity is challenged.

Table 4: Claim 2.2.4 breakdown

Claim No.	Claim	Chapter Section
2.2.4.1	RCS SSCs are designed using appropriate codes and standards, taking cognisance of Relevant Good Practice (RGP) and Operational Experience (OPEX).	1.4 RCS and ESFs Codes and Standards
2.2.4.2	Where the integrity of RCS SSCs is critical in preventing the release of radioactive material, methodology for High Reliability and Very High Reliability components is followed.	1.5.1.4 RCS Materials
2.2.4.3	The RCS design incorporates instrumentation which ensures that relevant safety functions are delivered.	1.5.1.6 RCS Instrumentation
2.2.4.4	The RCS design incorporates electrical systems which ensure that relevant safety functions are delivered.	1.5.1.7 RCS Electrical Systems
2.2.4.5	The safety components of the RCS are designed to withstand expected environmental conditions and a single failure with redundancy and independence.	1.5.1.8 RCS System Reliability
2.2.4.6	The RCS is designed with appropriate Inspection and Testing procedures to ensure that its design intent is met.	1.5.1.9 RCS Inspection and Testing

Claim 2.2.5: The Engineered Safety Features are designed to ensure they deliver relevant safety features, supported by substantiation which is suitably mature.

Claim 2.2.5 has been further decomposed within this chapter (Part B Chapter 1).

Claim 2.2.5 asserts that the ESFs have been designed to ensure that they deliver their safety functions and is decomposed into six sub-claims. The six sub-claims relate to the key aspects of the ESFs design.

Table 5 shows the breakdown of Claim 2.2.5 and identifies in which chapter of this PSR these claims are demonstrated to be met to a maturity appropriate for PSR v1.

Table 5: Claim 2.2.5 breakdown

Claim No.	Claim	Chapter Section
2.2.5.1	ESFs SSCs are designed using appropriate codes and standards, taking cognisance RGP and OPEX.	1.4 RCS and ESFs Codes and Standards
2.2.5.2	Where the integrity of ESFs SSCs is critical in preventing the release of radioactive material, methodology for High Reliability and Very High Reliability components is followed.	1.6.3 Passive Core Cooling System
2.2.5.3	The ESFs design incorporates instrumentation which ensures that relevant safety functions are delivered.	1.6.2 Passive Containment Heat Removal System 1.6.3 Passive Core Cooling System
2.2.5.4	The ESFs design incorporates electrical systems which ensure that relevant safety functions are delivered.	1.6.2 Passive Containment Heat Removal System 1.6.3 Passive Core Cooling System

Claim No.	Claim	Chapter Section
2.2.5.5	The safety components of the ESFs are designed to withstand expected environmental conditions and a single failure with redundancy and independence.	1.6.2 Passive Containment Heat Removal System 1.6.3 Passive Core Cooling System 1.6.4 Main Control Room Habitability System 1.6.5 Containment Isolation System
2.2.5.6	The ESFs are designed with appropriate Inspection and Testing procedures to ensure that their design intent is met.	1.6.2 Passive Containment Heat Removal System 1.6.3 Passive Core Cooling System 1.6.4 Main Control Room Habitability System 1.6.5 Containment Isolation System

Appendix B provides a full Claims, Arguments and Evidence mapping for Part B Chapter 1, which includes any summarises arguments and evidence needed to support the claims in the table above. This includes identification of evidence available at PSR v1 and aspects for future development of evidence to support these claims beyond PSR v1.

1.4 RCS AND ESF CODES AND STANDARDS

Claim 2.2.4.1: RCS SSCs are designed using appropriate codes and standards, taking cognisance of RGP and OPEX.

Evidence for Claim 2.2.4.1 is:

- **HI-2240338, Part B Chapter 4, Control and Instrumentation Systems** [7]**:** Part B Chapter 4 presents the Claims, Arguments and Intended Evidence for Control and Instrumentation that underpins the design of the generic SMR-300.
- **HI-2240339, Part B Chapter 6, Electrical Engineering** [9]: Part B Chapter 6 presents the Claims, Arguments and Intended Evidence for Electrical Engineering that underpins the design of the generic SMR-300.
- **HI-2240349, Part B Chapter 18, Structural Integrity** [13]: Part B Chapter 18 presents the Claims, Arguments and Intended Evidence for Structural Integrity that underpins the design of the generic SMR-300.
- **HI-2240356, Part B Chapter 19, Mechanical Engineering** [10]: Part B Chapter 19 presents the Claims, Arguments and Intended Evidence for Mechanical Engineering that underpins the design of the generic SMR-300.
- **HI-2240165, System Design Description for Reactor Coolant System** [19]: This document provides a high-level description of the functional, performance and safety requirements of the RCS. It provides the classification, safety related and non-safety related functions of the RCS.

Claim 2.2.5.1: ESFs SSCs are designed using appropriate codes and standards, taking cognisance of RGP and OPEX.

Evidence for Claim 2.2.5.1 is:

- **HI-2240338, Part B Chapter 4, Control and Instrumentation Systems** [7]**:** Part B Chapter 4 presents the Claims, Arguments and Intended Evidence for Control and Instrumentation that underpins the design of the generic SMR-300.
- **HI-2240339, Part B Chapter 6, Electrical Engineering** [9]: Part B Chapter 6 presents the Claims, Arguments and Intended Evidence for Electrical Engineering that underpins the design of the generic SMR-300.
- **HI-2240349, Part B Chapter 18, Structural Integrity** [13]: Part B Chapter 18 presents the Claims, Arguments and Intended Evidence for Structural Integrity that underpins the design of the generic SMR-300.
- **HI-2240356, Part B Chapter 19, Mechanical Engineering** [10]: Part B Chapter 19 presents the Claims, Arguments and Intended Evidence for Mechanical Engineering that underpins the design of the generic SMR-300.
- **HI-2240146, System Design Description for Passive Core Cooling System** [23]: This document provides a high-level description of the functional, performance and safety requirements of the PCC. It provides the classification, safety related and non-safety related functions of the PCC.
- **HI-2240170, System Design Description for Passive Containment Heat Removal System** [20]: This document provides a high-level description of the functional, performance and safety requirements of the PCH. It provides the classification, safety related and non-safety related functions of the PCH.

- **HI-2146060, System Design Description for the Main Control Room Habitability System** [22]: This document provides a high-level description of the functional, performance and safety requirements of the MCH. It provides the classification, safety related and non-safety related functions of the MCH.
- **HI-2146278, System Design Description for the Containment Isolation System** [21]: This document provides a high-level description of the functional, performance and safety requirements of the CIS. It provides the classification, safety related and non-safety related functions of the CIS. This is a SMR-160 System Design Description (SDD) that remains valid for the SMR-300 design for the purposes of GDA. An SMR-300 version is currently in the process of production.

1.4.1 Codes and Standards

Relevant codes and standards are selected based on the US SMR-300 categorisation and classification of the SSCs used within the RCS and the ESFs [24]. Justification and appropriateness of codes and standards selection is described in Part B Chapter 18 Structural Integrity [13] and Part B Chapter 19 Mechanical Engineering [10]. The SSC classification methodology, currently applied to the SMR-300, [24], utilises the approach described in U.S. Nuclear Regulatory Commission (U.S.NRC) Regulatory Guide (RG) 1.26, Revision 6 [25] and related guidance documents. This methodology is further summarised in Part A Chapter 2, sub-chapter 3.1 [3], which describes RG 1.26 Quality Groups A through D, and corresponding SSC Classification Standard “SMR Class”.

The SMR-300 SSC Classification Standard [24] designates the Quality Classification of SSCs as a function of SSC safety function. An overview of the classification of the RCS and ESFs SCCs is shown in Table 14 of Appendix A.

Table 6 displays a summary of the construction codes and standards for SMR-300 components, based on their Quality Group.

Table 6: Summary of Codes and Standards for SMR-300 Components [24]

Component	Quality Group A	Quality Group B	Quality Group C	Quality Group D
Pressure Vessels	ASME BPV Code, Section III, Division 1, Subsection NB: **Class 1**, Nuclear Power Plant Components	ASME BPV Code, Section III, Division 1, Subsection NC: **Class 2**, Nuclear Power Plant Components	ASME BPV Code, Section III, Division 1, Subsection ND: **Class 3**, Nuclear Power Plant Components	ASME Boiler and Pressure Vessel Code, Section VIII, Division 1
Piping	Class 1 (NB)	Class 2 (NC)	Class 3 (ND)	ANSI B31.1 Power Piping
Pumps	Class 1 (NB)	Class 2 (NC)	Class 3 (ND)	Manufacturer’s standards
Valves	Class 1 (NB)	Class 2 (NC)	Class 3 (ND)	ANSI B31.1 Power Piping and ANSI B16.34
Atmospheric Storage Tanks	N/A	Class 2 (NC)	Class 3 (ND)	API-650, AWWA D100, or ANSI B96.1
0-15 psig Storage Tanks	N/A	Class 2 (NC)	Class 3 (ND)	API-620

American National Standards Institute (ANSI)/Institute of Electrical and Electronics Engineers (IEEE)-603-1991 [26] gives the basic criteria for safety-related electrical and Instrumentation and Control (I&C) systems and equipment. Electrical and I&C system equipment and components are classified as Class 1E or non-Class 1E in accordance with definitions stated in IEEE 603 Standard. In general, the equipment and components that perform safety-related functions are designated as Class 1E and the equipment and components that do not perform any safety-related function are designated as Non-Class 1E. IEEE-603 Standard is endorsed by Regulatory Guide 1.153 [27] as a method acceptable to the U.S.NRC for complying with 10 CFR 50 Appendix A General Design Criteria [28],10 CFR 50.49 [29], and 10 CFR 50.55a [30], with respect to the design, reliability, qualification, and testability of the power, instrumentation, and control portions of safety systems for nuclear power plants. Further details of Electrical and I&C systems codes and standards is provided in Part B Chapter 6 Electrical Engineering [9] and Part B Chapter 4 Control and Instrumentation Systems [7].

Holtec acknowledge the existence of differences in the approach to safety categorisation and classification between the U.S.NRC regulatory guides and the UK expectations. SSCs will be provided a UK classification following the complete UK DBAA being conducted which will inform the detailed design at Pre-Construction Safety SSEC maturity. Discussion on how SSCs within the scope of this chapter are impacted can be found in sub chapter [REDACTED]. Further detail on the strategy for the categorisation and classification process as well as initial likely UK SSC classifications is described in Part A Chapter 2 [3]. Further work regarding the classification of SSCs and any relevant GDA commitments is managed via Part B Chapter 14 [12].

The approach adopted for UK deployment is centred around demonstrating equivalency between the SMR-300 design, and UK categorisation and classification expectations. This is achieved through the application of formal safety assessment techniques, which are consistent with UK context expectations. These safety assessment techniques are developed to identify a comprehensive set of UK aligned safety functions and associated safety measures, and to demonstrate that radiological risks are tolerable and ALARP. This formal UK aligned safety assessment commenced during GDA Step 2, through the development of a Preliminary Fault Schedule and a limited set of UK DBAA.

The categorisation and classification expectations which are derived from this UK aligned assessment, can then be compared with the existing SMR-300 design and its corresponding US classification. An initial judgement has therefore been formed on whether the codes and standards being applied as a result of the US classification, satisfy UK classification expectations. Where equivalency is at risk of not being demonstrable, it may lead to a UK design challenge and potential modification to the design or additional safety justification to demonstrate the chosen codes in the current design reduce risks to ALARP.

For RCS related SSCs, there is judged to be good alignment between UK equivalent expectations and the existing US design, from the output of the DBAA work to date. The only noted significant risk relates to the SG Shell and the expectation this would be equivalent to UK Class 1. This issue is discussed further in Part B Chapter 19 and via an associated Design Challenge paper. Appendix C provides further detail on a system by system basis.

For ESF related SSCs, there is judged to be good alignment between UK equivalent expectations and the existing US design, from the output of the DBAA work to date. The only noted significant risk relates to the PCH and expectation this would be equivalent to UK Class 1. This issue is discussed further in Part B Chapter 20 and the Containment Structure System Based view, which provides a preliminary justification of the appropriateness of the codes chosen for the containment structure. A design challenge has also been raised on this issue.

1.4.2 RGP and OPEX

The SMR-300 design is informed by OPEX from existing Pressurised Water Reactors (PWRs). A summary of RGP and OPEX identified to support this chapter can be found in sub-chapter 1.7.2.1.

Where UK design risks have been identified, Design Challenge Papers, managed by the Design Acceptance Committee (DAC), have been raised in accordance with the GDA Design Management Procedure [31]. Design Challenge Papers supporting topics within this chapter are discussed in sub-chapter 1.7.2.3. Further detail on the Design Challenge process can be found in PSR Part A Chapter 4 [5].

1.4.3 CAE Summary

The codes and standards used are described in the preceding sub-chapter which supports the following claims:

Claim 2.2.4.1: RCS SSCs are designed using appropriate codes and standards, taking cognisance of RGP and OPEX.

The codes and standards used in the design of RCS SSCs in the SMR-300 are appropriate for developing Nuclear Power Plants (NPP) and their SSCs in the UK. The RCS SSCs are designed based on significant OPEX, as collated in the Electric Power Research Institute (EPRI) Utilities Requirements Document (URD) and U.S.NRC Regulatory Guides. Holtec Britain are incorporating UK RGP and OPEX into the design of the RCS SSCs through established project processes using risk reviews, gap analysis, Design Challenges and the DAC. Therefore, demonstrating the above claim to a maturity expected for PSR.

Claim 2.2.5.1: ESFs SSCs are designed using appropriate codes and standards, taking cognisance of RGP and OPEX.

Similar to RCS SSCs, the codes and standards used in the design of ESFs SSCs in the SMR-300 are appropriate for developing nuclear power plants and their SSCs in the UK. The ESFs SSCs are designed based on significant OPEX, as collated in the EPRI URD and U.S.NRC Regulatory Guides. Holtec Britain are incorporating UK RGP and OPEX into the design of the ESFs SSCs through established project processes using risk reviews, gap analysis, Design Challenges and the DAC. Therefore, demonstrating the above claim to a maturity expected for PSR.

1.5 REACTOR COOLANT SYSTEM

1.5.1 Reactor Coolant System Overview

The RCS, as described in [19], is a dual leg primary heat transfer circuit, consisting of the following subsystems:

- RPV.
- SGE.
- RCPs (with associated hot and cold leg pipes).
- PZR.

All RCS equipment is located within the CS. During operation, the RCPs circulate pressurised water through the RPV and SGE. The water serves as coolant, moderator, as well as a solvent for boric acid which provides chemical shim reactivity control.

The reactor coolant is heated by the reactor core (heat source) and flows up to the top of the SGE, via two hot legs. The SGE (heat sink) cools the reactor coolant as it transfers the energy to the shell-side fluid (i.e., converts feedwater to steam). The cooler reactor coolant flows downward from the SGE and splits between two cold legs, each with an RCP, which provides the motive force for the reactor coolant during Normal Operations.

The RCS components are designed to enhance natural circulation flow during DBAs and AOOs, when RCPs are not available.

The predominant part of feedwater heating is performed by multi-stage feedwater heaters, of the Main Feedwater System (MFS), as in traditional power plants. Once in the SGE, feedwater is sprayed down, through an annulus, where it mixes with aspirating steam to heat it to near saturation conditions before reaching the lower tube-sheet and entering the tube bundle.

The RCS may be depressurised in the event of an accident (such as a LOCA), via its interface with the ADS, to allow for passive safety injection. Additional information on this depressurisation function is available in sub-chapter 1.6.3.2.8.

1.5.1.1 System Design Parameters

Table 7 presents key RCS design parameters.

Table 7: RCS Design Parameters

Parameter	Value	Reference
Plant Design Life, years	80	[32]
Nuclear Steam Supply System (NSSS) Thermal Power, MWt	1050	[33]
Nominal Operating Pressure, psig	[REDACTED]	[33]
Design Pressure, psig	[REDACTED]	[33]
Design Temperature, °F	[REDACTED]	[33]
PZR Design Temperature, °F	[REDACTED]	[33]
Number of heat transfer legs	2	[34]

1.5.1.2 RCS Functions

1.5.1.2.1 Safety Functions

The safety functions [19] of the RCS are to:

- **Remove heat from the fuel**

  The RCS provides a flow path for reactor coolant to remove heat from the reactor core, which supports maintaining the fuel condition within the operating bounds.

  Reactor coolant is transported through the RCS to transfer heat from the fuel to the balance of plant systems via the SGE during Power Operation as well as in the Hot Standby and Safe Shutdown modes (when the reactor is subcritical).

  The RCS transfers the core decay heat to the RHR during the subsequent phases of Shutdown, Cold Shutdown, and Refueling modes. (The RHR is described in more detail in Part B Chapter 5 Reactor Supporting Facilities [8]).

  During a DBA where the RCPB remains intact, the RCS transfers core decay heat from the fuel to the PDH, and to the SGE, where it is removed via the SDH.

  During LOCA DBAs where the RCPB is breached, connections at the RPV provide for direct safety injection of coolant from the PCM to meet emergency core cooling requirements. Decay heat from the core and spent fuel pool is passively transferred to the Annular Reservoir (AR) Ultimate Heat Sink (UHS) via the PCH. RCP flywheels provide continued flow for a short period, ensuring coast-down requirements are met.

- **Confine radioactive material**

  The RCS equipment piping, and components contain radioactive material and assure the integrity of the RCPB as a fission product barrier. The primary coolant is isolated by the RCPB from the CS atmosphere during Normal Operation and DBAs other than a LOCA.

  The SGE tubes, and tube sheet boundary prevent the transfer of radioactivity, generated within the reactor core, to the secondary cooling systems.

- **Overpressure protection**

  The PZR safety valves provide overpressure protection and prevent the RCS pressure exceeding design limits during operational transients to protect the integrity of the RCPB and vacuum breakers on the outlet piping.

1.5.1.2.2 Non-Safety Functions

The non-safety functions of the RCS are to:

- **Control RCS pressure and accommodate volume changes**

  The PZR design controls pressure and accommodates coolant volume changes to ensure operational limits are not exceeded. The PZR maintains the system pressure during operation, and limits pressure transients by establishing a saturated vapor-liquid interface (i.e., steam bubble). The PZR is sized to accommodate volume changes in the reactor coolant during a reduction or increase of plant load using the PZR volume during power operation.

  The RHR provides Low Temperature Overpressure Protection (LTOP) for the RCS during normal shutdown and startup operations. (The RHR is described in more detail in Part B Chapter 5 Reactor Supporting Facilities [8]).

- **Supply forced flow during startup and shutdown**

  The RCPs drive flow through the RCS to cool the core and ensure effective heat transfer to the SGE during power operations. During Startup and Shutdown, the flow removes decay heat from the core and ensures uniform heat-up or cooldown of RCS components.

- **Provide venting capability**

  The RCS provides a means to vent the RPV head and RCPs. The PZR steam space is vented via connections to other systems.

  The RPV head, RCP vent, and PZR head vent are used to fill the RCS during plant Startup. The ADS valves provide the capability to vent non-condensable gases from the RCS following an accident, see sub-chapter 1.6.3.2.8.

  RPV head vent valves and the ADS valves can be operated from the Main Control Room. SMR-300 does not require use of a RPV head vent to provide safety-related core cooling following a postulated accident.

- **Conduct non-critical heat up**

  The RCS provides a means to perform a non-critical heat up of the plant during startup operations using heat from the RCPs.

1.5.1.3 RCS Description

The RCS equipment is arranged and designed to transfer heat from the reactor core to the SGE, during Normal Operations, using forced circulation. In the event of an AOO or DBA, decay heat can be transferred to the SGE or other interfacing systems, such as PDH or RHR. The RCS is designed to use forced circulation at power and natural circulation for emergency cooling.

The RPV is designed to have two hot and two cold leg piping connections to a single SGE. The SGE, located adjacent to the RPV, is a Once-Through Steam Generator (OTSG) designed to produce superheated steam at full power conditions.

The PZR is attached to the SGE, above the upper tube sheet. The PZR is integral to the SGE, this eliminates the need for a separate pressure vessel, and associated surge piping, keeping the RCS footprint compact. An internal divider plate separates the PZR volume from the flowing RCS coolant volume. The PZR heaters are located above the divider plate. The PZR safety valves are located at the top of the PZR.

The LTOP for the RCS is provided by the relief valve in the RHR. Overpressure protection of the SGE secondary side is provided by the main steam safety valves in the MSS (see Part B Chapter 5 Reactor Supporting Facilities [8]).

The Control Rod Drive Mechanisms (CRDMs) are mounted on the RPV upper hemispherical head. Details of the CRDMs is presented in Part B Chapter 2 Reactor [6]. The RPV head vent discharges to the Passive Core Makeup Water Tank (PCMWT), part of the PCM.

The supply line to the PDH branches from the hot leg closest to the South PCMWT and returns to the SGE lower head.

The RCS design has provisions for venting non-condensable gases from the RPV head, RCPs, and from the top of the PZR. The RPV head vent line and the PZR vent line have orifices located downstream of the vent valves to limit the flow.

Reflective metal insulation will be used to the extent practical for all RCS equipment, components, and SSC inside containment. Any fibrous insulation used will be encapsulated to prevent fibrous debris from clogging the strainers located in the PCMWTs.

RCPB leakage detection will be accomplished using instrumentation from several systems. Diverse means of measurement, such as level, flow, pressure, temperature, and radioactivity are used. Design features include monitoring, and to the extent practical, collecting and quantifying RCPB leakage.

Sampling connections to the Primary Sampling System (PSL) are provided to allow for sampling from each hot leg and both the liquid and gaseous spaces of the PZR. The PSL is described in Part B Chapter 5 Reactor Supporting Facilities [8].

The RCS, or primary circuit boundary, provides the second barrier against the release of radioactivity generated within the fuel, and is designed to ensure a high degree of mechanical integrity throughout the lifetime of the power plant. SCCs containing reactor coolant have a safety function to maintain the RPCB during Normal Operation and DBAs other than a LOCA.

The RCS Process Flow Diagram [32] is shown in Figure 2.

[REDACTED]

Figure 2: RCS Process Flow Diagram

1.5.1.4 Higher Reliability Components

Claim 2.2.4.2: Where the integrity of RCS SSCs is critical in preventing the release of radioactive material, methodology for High Reliability and Very High Reliability components is followed.

Argument 2.2.4.2-A1: Components within the RCS have been selected as Higher Reliability candidates when appropriate.

Evidence for Argument 2.2.4.2-A1 is:

- **HI- 2240349, Part B Chapter 18, Structural Integrity** [13]: Part B Chapter 18 presents the Claims, Arguments and Intended Evidence for Structural Integrity that underpins the design of the generic SMR-300.

The following RCS components perform a critical role in preventing the release of radioactive material and are listed as Candidate Higher Reliability Components, as seen in Part B Chapter 18 [13]:

- RPV.
- SGE.
- PZR.
- RCP bowl.
- RCP flywheel and shaft.
- RCS Hot Leg and Cold Leg pipework.

1.5.1.5 RCS Materials

RCPB materials in contact with reactor coolant are selected to minimise corrosion. A water chemistry program, as described in Part B Chapter 23 Reactor Chemistry [14], is proposed to control corrosion and maintain compatibility with RCS materials.

Suitable materials for RCS SSCs are selected based on the US SMR-300 category and classification of the components, and applicable codes and standards. Descriptions of material selection and assessment is presented in Part B Chapters 18 Structural Integrity [13], Part B Chapter 19 Mechanical Engineering [10], and Part B Chapter 23 Reactor Chemistry [14].

1.5.1.6 RCS Instrumentation

Claim 2.2.4.3: The RCS design incorporates instrumentation which ensures that relevant safety functions are delivered.

Argument 2.2.4.3-A1: RCS instrumentation provides data to the PSS to actuate reactor protection functions and ESFs as required.

Evidence for Argument 2.2.4.3 – A1 is:

- **HI-2240338, Part B Chapter 4, Control and Instrumentation Systems** [7]**:** Part B Chapter 4 presents the Claims, Arguments and Intended Evidence for Control and Instrumentation that underpins the design of the generic SMR-300.
- **HI-2240165, System Design Description for Reactor Coolant System** [19]: This document provides a high-level description of the functional, performance and safety

requirements of the RCS. It provides the classification, safety related and non-safety related functions of the RCS.

RCS instrumentation is used for:

- Measuring conditions of the RCS and providing measurement data to the PSS, which actuates reactor protection functions and ESFs as required [19].
- Measuring conditions of the RCS and providing measurement data to the Plant Control System (PCS) for normal control and monitoring.

For the information regarding the Instrumentation and Control systems and architecture, see the Part B Chapter 4 Control and Instrumentation Systems [7].

1.5.1.6.1 Instrumentation for PSS

The RCS includes instrumentation to measure parameters which provide a signal to the PSS, for reactor protection and ESF actuation, during DBAs and some AOOs. The RCS uses redundant instrumentation (using four independent channels) for protection functions (i.e., for the Reactor Trip and Engineered Safety Features Actuation System (ESFA) logic).

The RCS hot leg temperature (T_{hot}) and cold leg temperature (T_{cold}) signals are processed by the PSS and used for Control Room indication, inputs to various control systems, and inputs to the PSS logic. T_{hot} and T_{cold} are both measured with two instrument sensors, per leg, that are averaged out to provide the channel readings. The averaged signals for T_{hot} and T_{cold} are used to generate RCS average temperature (T_{avg}) and delta temperature signals. The T_{avg} setpoint is used by PSS within the ESFA logic.

PZR pressure instrumentation is provided for Reactor Trip and ESFA logic in the PSS and is part of the PZR pressure permissive used in the ESFA logic.

PZR level instrumentation is provided for reactor trip and ADS actuation in the ESFA logic of the PSS.

RCS level instrumentation is provided to measure the RCS inventory, spanning from the top of the PZR to the RPV Direct Vessel Injection (DVI) lines. RCS inventory is crucial to all stages of accident response.

The cold legs are provided with instrumentation for monitoring the RCP flow. This instrumentation is provided for Reactor Trip logic in the PSS, for loss of flow Departure from Nucleate Boiling (DNB) protection.

The PSS shares measurement information (with proper isolation for independence) with the PCS for non-safety control of the RCS. The PSS transmits signals via a unit safety bus to the PCS for control purposes and normal monitoring.

Ex-core instrumentation provides permissive setpoints and reactor trip signals in the PSS logic.

1.5.1.6.2 Instrumentation for PCS

The RCS provides instrumentation to measure parameters which provide a signal to the PCS for plant control and monitoring in Normal Operation and AOOs.

The PZR heaters are controlled by the PCS, based on the PZR pressure instrumentation. The PZR level instrumentation provides input to the PCS to calculate and control the mass balance in the PZR. The purpose of the PZR level control system is to control charging and letdown flows, as well as heater operation, to maintain a programmed level.

Pressure instrumentation is also provided in the PZR to measure RCS pressure for indication during Normal Operation. PZR level is measured by narrow range level instruments. RCS inventory is measured by wide range level instruments, spanning from the PZR to the DVI lines. Both of these level instruments have a common reference leg. These instruments allow operators to monitor the level in the PZR during Normal Operation and in the RPV while the RCS is at reduced inventory, such as during refuelling operations.

Instrumentation is provided for operators to monitor the reactor coolant flow by four elbow-tap differential-pressure flow meters in each cold leg, tapping into the 90° elbow located upstream of each RCP. Pressure indicators are located immediately upstream and downstream of each RCP to monitor differential pressure, indicating the status of the pump.

The Pressuriser Safety Relief Valves (PSVs) are provided with full-open and full-closed position switches. The PSV discharge piping includes a temperature element to detect leakage or relief through the valve. [REDACTED]. Safety valve leakage detection is provided to ensure that operating personnel are aware of the status of SSCs important to safety, and able to perform timely corrective actions.

Leg temperatures are monitored by thermocouples and provide input signals for steam control and rod control. In addition to rod control and steam control the leg temperatures also provide inputs to PZR level control because the program level in the PZR depends on T_{avg}. RCS temperature sensors utilise thermowells, welded into the piping, to provide isolation between the temperature sensor and the process fluid, as opposed to direct immersion temperature sensors. The temperature probe and thermowell design allows easy removal and insertion for maintenance. Use of thermowells facilitates replacement of temperature sensors without requiring the RCS to be drained.

EIS provide input signals to control feedwater demand, rod control, reactor coolant letdown and charging.

1.5.1.7 RCS Electrical Systems

Claim 2.2.4.4: The RCS design incorporates electrical systems which ensure that relevant safety functions are delivered.

Argument 2.2.4.4-A1: Components and instrumentation required to conduct safety functions are supplied with safety-related electrical power.

Evidence for Argument 2.2.4.4 – A1 is:

- **HI-2240165, System Design Description for Reactor Coolant System** [19]: This document provides a high-level description of the functional, performance and safety requirements of the RCS. It provides the classification, safety related and non-safety related functions of the RCS.

- **HI-2240339, Part B Chapter 6, Electrical Engineering** [9]: Part B Chapter 6 presents the Claims, Arguments and Intended Evidence for Electrical Engineering that underpins the design of the generic SMR-300.

Safety-related valves and instrumentation are supplied with safety-related Class 1E power, direct from battery packs to meet requirements for redundancy and reliability [19].

Non-safety-related components and instrumentation are powered with non-Class 1E power. For example, the RCPs are supplied with non-Class 1E power since the pumps do not have a safety function, other than maintaining the RCPB which does not require electrical power.

The Stand-by Diesel Generator System can provide back-up to non-Class 1E AC power for the following equipment:

- PZR heaters.

The description and justification of the electrical engineering systems is presented in the Part B Chapter 6 Electrical Engineering [9].

1.5.1.8 RCS System Reliability

Claim 2.2.4.5: The safety components of the RCS are designed to withstand expected environmental conditions and a single failure with redundancy and independence.

Argument 2.2.4.5-A1: RCS is designed to ensure components required to conduct safety functions meet redundancy and reliability requirements.

Evidence for Argument 2.2.4.5 – A1 is:

- **HI-2240165, System Design Description for Reactor Coolant System** [19]: This document provides a high-level description of the functional, performance and safety requirements of the RCS. It provides the classification, safety related and non-safety related functions of the RCS.

The requirements of plant safety measures are developed through fault studies. These requirements drive the design of safety measures to ensure that they are suitably reliable to perform their functions. A robust methodology for identification and assessment of fault conditions for the SMR-300 has been performed based on the approach used in the US. This will be expanded to be in-line with UK RGP, as presented in Part B Chapter 14 Safety and Design Basis Accident Analysis [12].

For system reliability, the safety components of the RCS are designed against a single failure with redundancy and independence, and in accord with the Design Standard for Application of Single Failure Criterion [35]. This ensures that safety functions required for DBAs can be accomplished with acceptable reliability. System reliability is demonstrated through protection against:

- Single active failure.
- Single passive failure.
- Spurious valve actuation.
- Damage from fire, flood, dynamic effects.
- Environmental effects.

Differences between the US and UK approaches to single failure criterion and discussed further in sub-chapter 1.7.2.3.3.

To prevent a common failure and assure protection for pipe breaks, missiles, and fires, safety-related valves and components will meet the grouping and separation requirements described in the SMR-300 Design Standard for Grouping and Separation [36].

The PZR capacity increases operational margins by being sufficiently sized to account for changes in PZR level and RCS pressure, which results in fewer reactor trips that would occur due to high PZR water level or high PZR pressure.

Redundant valves in series assure isolation of the RCS to maintain the RCPB and ensure that an inadvertent actuation of a valve will not result in a LOCA. The RPV head vent forms the high point for the RPV and is designed with redundant valves in series to assure isolation in case an open vent valve cannot be closed.

The following are system design features for reliability [19]:

- Redundancy in RCS components with a safety function:
  a) Redundant PZR safety valves for overpressure protection.
  b) Redundant valves in series to isolate the RCS and maintain the RCPB.
- Duplicate RCS components with non-safety function, other than maintaining the RCPB:
  a) Two RCPs.
  b) PZR spray valves.

1.5.1.9 RCS Inspection and Testing

Claim 2.2.4.6: The RCS is designed with appropriate Inspection and Testing procedures to ensure that its design intent is met.

Evidence for Claim 2.2.4.6 is:

- **HI-2240165, System Design Description for Reactor Coolant System** [19]: This document provides a high-level description of the functional, performance and safety requirements of the RCS. It provides the classification, safety related and non-safety related functions of the RCS.
- **HPP-160-3016, SMR-300 Design Standard for Human Factors: Maintenance, Inspection and Testing** [37]: This document is a Design Guide, for the SMR-300, to assist designers in ensuring that human capabilities and limitations are considered during the design and layout of equipment.

The RCS is designed with provisions for access and clearances around equipment, piping, and components that are part of the RCPB to meet American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (BPVC) Section XI. The equipment arrangement inside the CS is designed to permit periodic inspection and testing. Sufficient space is provided for personnel, inspection equipment, remote inspection equipment, removal space and temporary storage, handling machinery, and repairs/replacement [19].

The system is designed in accordance with the Design Standard for Human Factors: Maintenance, Inspection and Testing [37] to ensure consideration of human factors for inspection and tests.

Schedules will be developed for in-service inspection and testing activities during detailed design in accordance with ASME Boiler and Pressure Vessel Code Section XI. Further discussion regarding EMIT philosophy and management can be found in Part B Chapter 9 [38].

1.5.1.9.1 Inspection

The RCPB SSCs are examined to assess the structural leak-tight integrity to meet the examination requirements of ASME Boiler and Pressure Vessel Code Section III Division 1 Subsection NB for Class 1 components.

Steam generator manways are located near both tube sheets for access and inspection of tubes and tube sheets. Steam generator tubes can be inspected over the entire length for the detection of tube defects.

The Leak Detection System will be developed for detecting and monitoring RCS leakage. It will be determined if the Leak-Before-Break methodology and associated in-service inspection criterion leakage detection will be applied to portions of the RCS piping to minimise the probability of a pipe rupture.

1.5.1.9.2 Testing

System pressure tests (leakage and hydrostatic) will be performed in accordance with ASME Boiler and Pressure Vessel Code Section XI. The charging pump in the CVC can be used for the system leakage test at the RCS pressure corresponding to 100% rated reactor power, in accordance with ASME Boiler and Pressure Vessel Code Section XI Article IWB-5000 for system pressure tests. The CVC is described in the Part B Chapter 5 Reactor Supporting Facilities [8].

The PZR safety valves will be tested and qualified (10 CFR 50.34(f)(2)(x) for TMI Action Plan Item II.D.1 [39]) by the EPRI Test Program [40]. The LTOP relief valve that is part of the RHR has a set pressure less than the full RCS design pressure, therefore, this relief valve is excluded from the EPRI Test Program and will be covered by the test program that will be developed later.

1.5.1.10 System Boundaries

1.5.1.10.1 Mechanical

For lines flowing into the RCS, the mechanical system boundary is at the outlet of the last isolation valve leaving the interfacing system, except for the interfaces listed below where the boundary is at the equipment nozzle (or otherwise noted).

- Discharge line from the RHR. The boundary is at Cold Leg B.
- Return line from CVC from the charging pump. The boundary is at the Cold leg upstream of each RCP.
- PZR spray supply line from CVC. The boundary is at the common spray supply line.
- DVI lines at the RPV from PCM. The boundary is at the RPV.

- Seal injection from CVC. The boundary is at the RCP seals.
- Cooling water supply from the Component Cooling Water System (CCW) to the RCP. The boundary is the RCP.
- Discharge line from the PDH. The boundary is at the SGE lower head.

For lines flowing out of the RCS, the mechanical system boundary is at the outlet of the last isolation valve leading to the interfacing system, except for the interfaces listed below where the boundary is at the RCS equipment nozzle (or otherwise noted).

- Supply line to the RHR. The boundary is at Hot Leg B.
- Letdown line to the CVC. The boundary is at the Cold Leg downstream of each RCP.
- Supply line to the PDH. The boundary is at Hot Leg B.
- RPV head vent line to PCMWT in the PCM. The boundary is at the PCMWT.
- Supply lines to the PSL. The boundary is at the inlet of the first isolation valve in the PSL.

1.5.1.10.2 Electrical

The boundary between the electrical system and the RCS is at the connection of the electrical power to the component (i.e., motor).

1.5.1.10.3 Instrumentation and Control

The instrumentation and control boundary are at the connection of the RCS instruments to the PCS or PSS.

1.5.2 Reactor Pressure Vessel (RPV)

Claim 2.2.2.1: Core decay heat removal is ensured following credible initiating events in all plant states.

Argument 2.2.2.1-A1: The RPV geometry is designed to ensure core heat removal is achieved following credible initiating events in all plant states.

Evidence for Argument 2.2.2.1 – A1 is:

- **HI-2240165, System Design Description for Reactor Coolant System** [19]: This document provides a high-level description of the functional, performance and safety requirements of the RCS. It provides the classification, safety related and non-safety related functions of the RCS.
- **Holtec Britain, Design Basis Report – Structural Integrity Analysis and Design** [41]: This document presents the current design basis for the RPV which eventually be formalised in a design specification.

Claim 2.2.2.2: Sufficient coolant inventory is maintained following credible initiating events in all plant states.

Argument 2.2.2.2-A1: The RPV allows for direct injection of makeup water, during LOCA.

Evidence for Argument 2.2.2.2 – A1 is:

- **DWG-14859 R0, Reactor Pressure Vessel** [42]: This drawing provides details of the geometry of the RPV, displaying the locations of direct injection points.
- **HI-2240165, System Design Description for Reactor Coolant System** [19]: This document provides a high-level description of the functional, performance and safety requirements of the RCS. It provides the classification, safety related and non-safety related functions of the RCS.

Claim 2.2.3.1: The RCS and ESF SSCs ensure the integrity of the Reactor Coolant Pressure Boundary following credible initiating events in all plant states.

Argument 2.2.3.1-A1: RCS SSCs which form the RCPB (RPV, SGE, PZR, RCP, and associated valves and piping) are designed to ensure its integrity following credible initiating events, isolating the reactor coolant from the Containment atmosphere and secondary cooling systems.

Evidence for Argument 2.2.3.1 – A1 is:

- **HI-2240165, System Design Description for Reactor Coolant System** [19]: This document provides a high-level description of the functional, performance and safety requirements of the RCS. It provides the classification, safety related and non-safety related functions of the RCS.

1.5.2.1 RPV Description

The RPV is a thick-walled cylindrical pressure vessel with an integrally welded bottom head and a removable flanged top head [42]. The upper shell of the RPV is thicker than the lower shell to incorporate the flange and provide the necessary strength to weld the RCS leg piping directly to the RPV. By incorporating the flange into the shell, the distance between the RCS leg nozzles and the core can be maximised.

The RPV top head is connected to the shell via high-strength alloy steel fasteners that are pre-tensioned to ensure a high integrity, double-gasket seal during all operating modes. Concentric, self-energising gaskets, compressed between the two mating flanges ensure leak tightness of the RPV. Seal drains, that discharge to the CS sump in the Liquid Radwaste System (LRW), are provided at the RPV flange gaskets to monitor leakage between the gaskets.

The RPV is connected to the SGE via two hot and cold leg pipe runs. The RPV uses a traditional RPV head package including external CRDMs. This enables the use of typical refuelling equipment and operations.

There are no penetrations in the lower two-thirds of the RPV, eliminating the potential for pipe breaks that could lead to uncovering the core. The upper hemispherical head has penetrations for the control rods. Each penetration is welded to a control rod pressure housing. CRDMs are externally attached to the control rod pressure housings. The upper hemispherical head also contains penetrations for the IIS tubes and a high-point vent line. RPV level is measured with shared instrumentation for RCS inventory level (upper RPV volume down to DVI lines) and in-core level (DVI lines to bottom of fuel envelope).

During operation, reactor coolant from the cold legs enters the RPV through the inlet nozzles and flows into the downcomer, which is the annulus formed by the space between the core barrel and the RPV inner wall. The flow then enters the lower plenum, which is the area at the bottom of the RPV. From here, the flow enters the Flow Distribution Device (FDD) and is directed to the Lower Support Plate (LSP) which is attached to the FDD. After traversing the core, the heated reactor coolant passes through the Upper Core Plate (UCP) and enters the upper plenum, which is enclosed by the UCP, core barrel and Upper Support Plate (USP). The flow then passes through and around the columns attaching the UCP and USP to reach the RPV outlet nozzles. Further details of the coolant flow within the RPV can be found in the Overview of Core Components document [43] and discussed within Part B Chapter 2 [6].

During accident conditions (LOCA DBAs) where the RCPB is breached, connections at the RPV provide for direct safety injection of coolant from the PCMWS to meet emergency core cooling requirements.

1.5.2.2 RPV External Supports

The external supports for the RPV will be trunnions to the surrounding concrete walls and a lateral restraint at the bottom of the RPV cavity to help withstand seismic events.

1.5.3 Steam Generator (SGE)

Claim 2.2.2.1: Core decay heat removal is ensured following credible initiating events in all plant states.

Argument 2.2.2.1-A2: The SGE geometry is designed to ensure core heat removal is achieved following credible initiating events in all plant states

Evidence for Argument 2.2.2.1 – A2 is:

- **HI-2240165, System Design Description for Reactor Coolant System** [19]: This document provides a high-level description of the functional, performance and safety requirements of the RCS. It provides the classification, safety related and non-safety related functions of the RCS.
- **HI-2240157, Design Specification for the SMR-300 Steam Generator** [44]: This document provides a high-level description of the design, functions, and operation of the SGE for the SMR-300 in compliance with the ASME BPVC and applicable regulatory requirements and guidance.

Claim 2.2.3.1: The RCS and ESF SSCs ensure the integrity of the Reactor Coolant Pressure Boundary following credible initiating events in all plant states.

Argument 2.2.3.1-A1: RCS SSCs which form the RCPB (RPV, SGE, PZR, RCP, and associated valves and piping) are designed to ensure its integrity following credible initiating events, isolating the reactor coolant from the Containment atmosphere and secondary cooling systems.

Evidence for Argument 2.2.3.1-A1 is:

- **HI-2240165, System Design Description for Reactor Coolant System** [19]: This document provides a high-level description of the functional, performance and safety requirements of the RCS. It provides the classification, safety related and non-safety related functions of the RCS.

1.5.3.1 SGE Description

The SGE is a vertical shell and tube OTSG. The SGE is designed to utilise reactor coolant on the tube side and feedwater on the shell side. The SGE tubes and tube sheets are part of the RCPB and serve as a barrier to the release of radioactive materials from the core to the secondary system.

RCS hot legs connect the OTSG upper channel (turn around region) to the RPV outlet nozzles. The hot reactor coolant from the RPV flows down through the OTSG tubes to the lower head and into the cold leg connections back to the RPV. The primary coolant transfers heat to the feedwater flowing up the shell side of the OTSG to produce superheated steam.

The straight OTSG tubes provide easy access for in-service inspection and tube plugging as needed. The OTSG is designed for rated performance with a percentage of the tubes plugged.

The feed water is preheated, prior to entering the OTSG shell. Once in the OTSG, feedwater flows from the feed ring down the outer annulus, where it is preheated with aspirating steam, before reversing direction and flowing up the tube bundle of the OTSG. As feedwater flows up the tube bundle, vapor gradually forms a film on the tube surface. The steam generated is then superheated as it rises to the top of the OTSG. The tubes are supported by broached hole baffles for the entire length of the tube bundle.

The SGE is integral to the performance of the SDH. During an accident condition the RCS transfers decay heat from the core to the SGE where it boils feedwater. The SDH condenses this steam to indirectly remove heat from the reactor coolant (see 1.6.3.2.7 for details regarding SDH function). The condensate is returned to the SGE.

1.5.3.2 SGE External Supports

The external supports for the steam generator will be skirt type at the SGE base with lateral struts supported on an enclosure structure.

1.5.4 Pressuriser (PZR)

Claim 2.2.3.1: The RCS and ESF SSCs ensure the integrity of the Reactor Coolant Pressure Boundary following credible initiating events in all plant states.

Argument 2.2.3.1-A1: RCS SSCs which form the RCPB (RPV, SGE, PZR, RCP, and associated valves and piping) are designed to ensure its integrity following credible initiating events, isolating the reactor coolant from the Containment atmosphere and secondary cooling systems.

Evidence for Argument 2.2.3.1-A1 is:

- **HI-2240165, System Design Description for Reactor Coolant System** [19]: This document provides a high-level description of the functional, performance and safety requirements of the RCS. It provides the classification, safety related and non-safety related functions of the RCS.

1.5.4.1 PZR Description

The PZR features a thick cylindrical shell with a top hemispherical head. The PZR has been sized to account for changes in PZR level and RCS pressure during operational manoeuvres and transients. This eliminates the need for fast-acting Power Operated Relief Valves (PORVs) and therefore the need for a PZR relief tank, which obviates potential PORV malfunctions common to operating PWRs. The PZR is designed to maintain RCPB integrity following credible initiating events.

The internal divider plate is designed to limit thermal communication between the volumes on either side to minimise the size of the PZR heaters while allowing sufficient flow into and out of the PZR for pressure control. [REDACTED]. The PZR controls the pressure of the RCS by maintaining a saturated steam-liquid interface. Pressure changes caused by the expansion and contraction of the reactor coolant are absorbed by condensing the steam using the PZR spray or generating steam using the PZR heaters. RCP discharge pressure drives normal spray flow. Continuous flow through the pressuriser spray line maintains the spray line warm which prevents thermal stratification while also keeping the pressuriser volume at similar boron concentration as the rest of the RCS volume.

1.5.4.2 PZR Supports

The PZR is laterally restrained by support structures tied to an enclosure structure.

1.5.4.3 Pressuriser Safety Relief Valves (PSV)

Claim 2.2.3.1: The RCS and ESF SSCs ensure the integrity of the Reactor Coolant Pressure Boundary following credible initiating events in all plant states.

Argument 2.2.3.1–A3: The PZR safety valves provide overpressure protection and prevent the RCS pressure exceeding design limits to protect the integrity of the RCPB.

Evidence for Argument 2.2.3.1-A3 is:

- **HI-2240660, SMR-300 Pressurizer Safety Valve Sizing Evaluation** [45]: This document establishes a basis for sizing, quality, and setpoint for the PSVs for the SMR-300. A review of system design, U.S.NRC, and ASME requirements was completed to establish the criteria.
- **HI-2240165, System Design Description for Reactor Coolant System** [19]: This document provides a high-level description of the functional, performance and safety requirements of the RCS. It provides the classification, safety related and non-safety related functions of the RCS.

The PSVs prevent the RCPB from exceeding the RCS design pressure during any transients. The PSVs provide overpressure protection in accordance with ASME Boiler Pressure and Vessel Code Section III Division 1, Subsection NB. The discharge line is routed from the outlet of the PSV to the PCMWT.

Preliminary PSV sizing has been completed [45] to ensure the SMR-300 design can adequately cope with an overpressure transient. [REDACTED]

1.5.5 Reactor Coolant Pump (RCP)

Claim 2.2.2.1: Core decay heat removal is ensured following credible initiating events in all plant states.

Argument 2.2.2.1-A3: The RCPs maintain flow to the core via flywheel coast-down.

Evidence for Argument 2.2.2.1 – A3 is:

- **HI-2240165, System Design Description for Reactor Coolant System** [19]: This document provides a high-level description of the functional, performance and safety requirements of the RCS. It provides the classification, safety related and non-safety related functions of the RCS.

Claim 2.2.3.1: The RCS and ESF SSCs ensure the integrity of the Reactor Coolant Pressure Boundary following credible initiating events in all plant states.

Argument 2.2.3.1-A1: RCS SSCs which form the RCPB (RPV, SGE, PZR, RCP, and associated valves and piping) are designed to ensure its integrity following credible initiating events, isolating the reactor coolant from the Containment atmosphere and secondary cooling systems.

Evidence for Argument 2.2.3.1-A1 is:

- **HI-2240165, System Design Description for Reactor Coolant System** [19]: This document provides a high-level description of the functional, performance and safety requirements of the RCS. It provides the classification, safety related and non-safety related functions of the RCS.

1.5.5.1 RCP Description

The primary function of the RCP is to provide forced circulation of reactor coolant for transfer of core heat during reactor startup, shutdown, and power operations. The SMR-300 design does not include provisions to operate with a single pump running during power operations (Mode 1).

Two 50% capacity pumps are provided, one on each cold leg. There are two safety-related characteristics of the RCP:

- In the event of a Reactor Trip or loss of power, the pumps maintain flow to the core during pump coast-down using a flywheel.
- The RCPs serve as a part of the RCPB.

The pumps are not required to provide any active safety functions and will be de-energised during LOCA or operation of the PDH or SDH.

The RCP is a vertical, centrifugal type pump. The pump is designed with traditional seals, allowing continued operation for at least one week, after failure of one of the several seal stages. The pump's design permits an orderly plant shutdown without excessive loss of reactor coolant in the event of the degradation of a second seal stage. In addition, a backup seal stage, capable of holding against full system pressure with the pump idle, is provided. The impellor is mounted to the rotor shaft. The rotor portion of the motor is not exposed to the process fluid. The RCP has provisions to prevent motor damage, due to overheating, using an onboard heat exchanger that is supplied with cooling water via the CCW.

1.5.6 RCS Piping

Claim 2.2.3.1: The RCS and ESF SSCs ensure the integrity of the Reactor Coolant Pressure Boundary following credible initiating events in all plant states.

Argument 2.2.3.1-A1: RCS SSCs which form the RCPB (RPV, SGE, PZR, RCP, and associated valves and piping) are designed to ensure its integrity following credible initiating events, isolating the reactor coolant from the Containment atmosphere and secondary cooling systems.

Evidence for Argument 2.2.3.1-A1 is:

- **HI-2240165, System Design Description for Reactor Coolant System** [19]: This document provides a high-level description of the functional, performance and safety requirements of the RCS. It provides the classification, safety related and non-safety related functions of the RCS.

1.5.6.1 RCS Piping Description

Reactor coolant piping is designed in accordance with ASME Boiler and Pressure Vessel Code Section III, Division 1, Subsection NB for Class 1 Components and Subsection NC for Class 2 Components. The reactor coolant piping maintains the RCPB and is designed to withstand the maximum RCS pressure and temperature under all expected modes of plant operation and AOOs. Seamless piping is used to avoid in-service inspection of longitudinal piping welds. The piping material for the hot legs and cold legs shall be carbon steel with stainless steel cladding. All other pipes shall be stainless steel.

[REDACTED]

1.5.7 System Interfaces

The RCS interfaces with the following systems to ensure that the design requirements are satisfied:

Table 8: System Interfaces with the RCS

System	Description	Function	PSR Chapter
ADS	Automatic Depressurization System	The ADS is a subsystem of the Passive Core Cooling System. The ADS valves are part of the PCC and have safety functions for emergency core cooling.	Sub-chapter 1.6.3.2.8
CVC	Chemical and Volume Control System	Removes fission products, activation products, corrosion products from the reactor coolant with provisions for water chemistry control. The charging pump in the CVC supplies backup spray flow to the pressuriser for pressure control. The CVC controls the RCS water volume via the letdown lines to the LRW and the charging pump through the charging lines as needed. If required, borated water may be injected into the RCS by the charging pump. The charging pump supplies RCP seal water and can be used for the RCS leakage testing.	Part B Chapter 5 Reactor Supporting Facilities

System	Description	Function	PSR Chapter
CCW	Component Cooling Water System	The CCW supplies cooling water to the reactor coolant pumps.	N/A
DAS	Diverse Actuation System	The DAS control safety-related valves upon PSS failure to actuate.	Part B Chapter 4 Control and Instrumentation Systems
DCE	DC Power Distribution System	The DCE supplies power to the safety-related valves and instruments.	Part B Chapter 6 Electrical Engineering
DGS	Stand-By Diesel Generator System	The DGS supplies backup, non-Class 1E AC power to the medium voltage AC bus.	N/A
ICE	I&C Power Distribution System	The ICE supplies power to the instrumentation of the RCS.	Part B Chapter 6 Electrical Engineering
LDS	Leak Detection System	The LDS provides the capability to detect, monitor, and measure reactor coolant pressure boundary leakage from the RCS.	N/A
LVE	Low Voltage AC Distribution System	The LVE is a 480 VAC system which supplies power to the non-safety related valves and pumps in the RCS, including the PZR heaters, as well as distributing power to lower voltage systems.	Part B Chapter 6 Electrical Engineering
MFS	Main Feedwater System	The MFS supplies feedwater and interfaces with the secondary side of the SGE.	Part B Chapter 5 Reactor Supporting Facilities
MHS	Mechanical Handling Systems	The MHS supports refuelling operations for removing the upper head of the RPV, unloading, and refuelling the reactor core during outages. However, there is no permanent physical interface between the RCS and MHS.	Part B Chapter 5 Reactor Supporting Facilities
MSS	Main Steam System	The MSS supplies main steam to the turbine and interfaces with the secondary side of the SGE.	Part B Chapter 5 Reactor Supporting Facilities
MVE	Medium Voltage AC Distribution System	The MVE is a 4kV system which powers large loads and provides distribution to the lower voltage systems. It includes electrical buses in both Tier 1 and Tier 2. The Tier 2 buses are backed by non-safety-related Standby Diesel Generators.	N/A
PCM	Passive Core Makeup Water System	The PCM is a subsystem of the PCC and provides safety injection of water to the RCS from the accumulators and PCMWT. The RPV head vent line, PSVs, and ADS 1 valves discharge to the PCMWT.	Sub-chapter 1.6.3.2.9
PCS	Plant Control System	The PCS monitors and controls the RCS components with non-safety functions. RCS instrumentation provides a signal to PCS.	Part B Chapter 4 Control and Instrumentation Systems
PSS	Plant Safety System	RCS instrumentation provides a signal to PSS. The PSS provides Reactor Trip signal, ESFA actuation signal, component control, and accident automated actuation and manual control of all safety-related plant components.	Part B Chapter 4 Control and Instrumentation Systems
PAM	Post-Accident System	RCS instrumentation supports accident monitoring and post-accident monitoring.	Part B Chapter 4 Control and Instrumentation Systems
PDH	Primary Decay Heat Removal System	A portion of the RCS piping provides a path for the PDH. The PDH is a subsystem of the PCC and provides emergency core cooling for RCS cooling.	Sub-chapter 1.6.3.2.6
PSL	Primary Sampling System	The PSL collects, analyses, and monitors samples of reactor coolant, including sampling of the gaseous and liquid phases of the PZR.	Part B Chapter 5 Reactor Supporting Facilities
RHR	Residual Heat Removal System	The RHR provides RCS (decay and sensible) cooling during normal cooldown. The relief valve in the RHR provides LTOP for the RCS.	Part B Chapter 5 Reactor Supporting Facilities

System	Description	Function	PSR Chapter
SDH	Secondary Decay Heat Removal System	The SDH interface with the secondary side of the SGE. The SDH indirectly provides emergency core cooling for the RCS (independent of PDH).	Sub-chapter 1.6.3.2.7
SFC	Spent Fuel Pool Cooling System	The spent fuel pool water and the reactor cavity interface when the RPV head is removed for refuelling tasks.	Part B Chapter 5 Reactor Supporting Facilities
SGB	Steam Generator Blowdown System	The SGB receives and processes the blowdown from the secondary side of the SGE.	N/A

1.5.8 RCS CAE Summary

The RCS, described in the preceding sub-chapter, supports the following claims by demonstrating the below arguments:

Claim 2.2.2.1: Core decay heat removal is ensured following credible initiating events in all plant states.

Argument 2.2.2.1-A1: The RPV geometry is designed to ensure core heat removal is achieved following credible initiating events in all plant states.

Argument 2.2.2.1-A2: The SGE geometry is designed to ensure core heat removal is achieved following credible initiating events in all plant states

Argument 2.2.2.1-A3: The RCPs maintain flow to the core via flywheel coast-down.

Description of the RPV, SGE and RCPs is detailed in the RCS SDD [19]. The RPV and SGE mechanical design, as per ASME Code Section III, ensures a basis for structural integrity is maintained for a range of transient conditions, as seen in the Design Basis report for Structural Integrity for the RPV [41] and SGE Design Specification [44]. Furthermore, the designation of Higher Reliability to these components ensures manufacturing, inspection and testing of the components is appropriate to the level of safety function that they deliver. Further discussion regarding the designation of Higher Reliability is presented in Part B Chapter 18 [13]. Although further work is still to be conducted with regards assessing initiating events and UK context analysis, this claim has been met to a maturity appropriate for PSR.

Claim 2.2.2.2: Sufficient coolant inventory is maintained following credible initiating events in all plant states.

Argument 2.2.2.2-A1: The RPV allows for direct injection of makeup water, during LOCA.

During accident conditions (LOCA DBAs) where the RCPB is breached, connections at the RPV provide for direct safety injection of coolant from the PCMWS to meet emergency core cooling requirements [19]. Detailed UK safety analysis is still to be conducted, however, US based LOCA fault progression and preliminary analysis conducted by the Preliminary Fault Schedule (PFS) [46] and UK DBAA [47] provide confidence on how the local fault progresses and that the design will meet the safety requirements. Therefore, this has been met to a maturity appropriate for PSR.

Claim 2.2.3.1: The RCS and ESF SSCs ensure the integrity of the Reactor Coolant Pressure Boundary following credible initiating events in all plant states.

Argument 2.2.3.1-A1: RCS SSCs which form the RCPB (RPV, SGE, PZR, RCP, and associated valves and piping) are designed to ensure its integrity following credible initiating events, isolating the reactor coolant from the Containment atmosphere and secondary cooling systems.

Argument 2.2.3.1-A3: The PZR safety valves provide overpressure protection and prevent the RCS pressure exceeding design limits to protect the integrity of the RCPB.

RCS SSCs including RPV, SGE, PZR, RCP, and associated valves and piping are designed to ensure RCPB integrity following credible initiating events, isolating the reactor coolant from the Containment atmosphere and secondary cooling systems [19]. The PZR safety valves also provide overpressure protection and prevent the RCS pressure exceeding design limits to protect the integrity of the RCPB [19]. The SGE shell (which it not part of the RCPB and therefore not covered in this chapter) is likely to need a higher classification in the UK context due to it being a candidate very high integrity component which may introduce differing design code requirements. This is covered in greater detail is PSR Part B Chapter 18.

Following further UK safety analysis, a definitive list of initiating events will be produced. However, utilising preliminary UK analysis [46], initial UK DBAA [47] and US based work, confidence is provided that this claim will be fully met throughout the next phase of safety case development. This claim is currently assessed as met to a maturity appropriate for PSR.

Claim 2.2.4.2: Where the integrity of RCS SSCs is critical in preventing the release of radioactive material, methodology for High Reliability and Very High Reliability components is followed.

Argument 2.2.4.2-A1: Components within the RCS have been selected as Higher Reliability candidates when appropriate.

The RPV, SGE, PZR, RCP bowl, RCP flywheel and shaft, and RCS Hot Leg and Cold Leg pipework are candidate higher reliability components [13], therefore, meeting this claim to the maturity expected at PSR. Throughout safety case and design development, higher reliability candidates will be reassessed as safety analysis matures.

Claim 2.2.4.3: The RCS design incorporates instrumentation which ensures that relevant safety functions are delivered.

Argument 2.2.4.3-A1: RCS instrumentation provides data to the PSS to actuate reactor protection functions and ESFs as required.

RCS instrumentation is responsible for measuring conditions of the RCS and providing measurement data to the PSS, which actuates reactor protection functions and ESFs as required [19]. Therefore, meeting the claim to the expected level of maturity at PSR.

Claim 2.2.4.4: The RCS design incorporates electrical systems which ensure that relevant safety functions are delivered.

Argument 2.2.4.4-A1: Components and instrumentation required to conduct safety functions are supplied to ensure redundancy and reliability requirements are met.

Safety-related valves and instrumentation are supplied with safety-related Class 1E power, direct from battery packs to meet requirements for redundancy and reliability [19]. Therefore, meeting the claim to the expected level of maturity at PSR.

Claim 2.2.4.5: The safety components of the RCS are designed to withstand expected environmental conditions and a single failure with redundancy and independence.

Argument 2.2.4.5-A1: RCS is designed to ensure components required to conduct safety functions meet redundancy and reliability requirements.

Redundant pressuriser safety valves are included for overpressure protection [19]. Redundant valves in series are included to isolate the RCS and maintain the RCPB [19]. Therefore, meeting the claim to the expected level of maturity at PSR.

Claim 2.2.4.6: The RCS is designed with appropriate Inspection and Testing procedures to ensure that its design intent is met.

Programs will be developed for in-service inspection and testing activities during detailed design in accordance with ASME Boiler and Pressure Vessel Code Section XI [19]. This claim is not yet fully met but at this stage of safety case and design development, a maintenance schedule is not expected. However, as outlined in Part B Chapter 9 [38], the SMR-300 design management arrangements have been developed to ensure that an effective EIMT schedule will be delivered. The EIMT schedule will include appropriate activities to enable SSCs to continue to deliver their design basis. The SMR-300 layout design takes into account space requirements for the safe conduct of EIMT activities, including inspection methods to identify potential degradation of SSCs. Therefore, with an early understanding of what is required from a maintenance program, the claim demonstrated to a maturity level appropriate for PSR.

1.6 ENGINEERED SAFETY FEATURES

1.6.1 Engineered Safety Features Overview

ESFs are the systems which mitigate the consequences of postulated DBAs and protect workers and the public in the unlikely event of an accidental release of radioactive fission products from the RCS. The ESFs function is to limit, control, mitigate, and terminate accidents, while maintaining the radiation exposure levels to the public below applicable regulatory limits and guidelines. The following are defined as ESFs:

- PCH [20].
- PCC [23].
- MCH [22].
- CIS [21].

1.6.1.1 System Functions

1.6.1.1.1 Safety Functions

The SMR-300 ESFs are designed to:

- Maintain the plant in a safe shutdown configuration.
- Provide emergency heat removal to the core.
- Provide emergency makeup water to the core and Spent Fuel Pool (SFP) to prevent fuel element damage.
- Isolate the Containment from the environment to prevent radioactive effluent releases.
- Ensure a habitable environment for plant operators.

1.6.1.1.2 Non-Safety Functions

The ESFs have many non-safety functions, described in detail in the following sub-chapters.

1.6.1.2 System Description

The SMR-300 incorporates multiple levels of defence-in-depth to remove heat from the reactor and assure safety. All safety-systems, excluding the MCH (located in the RAB), are located inside the robust CES, rendering them secure and safe from external threats, both natural and man-made, as described in Part B Chapter 20 Civil Engineering [11]. The systems are simpler than those in current operating reactors, eliminating active pumps from the safety functions, thus making them more reliable. No operator actions are required to place and maintain the reactor in a safe shutdown condition. All makeup water needed for a postulated LOCA is inside the CS, thus making the Containment fully isolable, eliminating dose to the public and effects on the environment from this event.

During normal operations, such as when the reactor is generating electricity or during refuelling, heat is removed from the RCS by rejecting heat to the steam turbine and condenser (while at power) or through highly reliable active systems operating with pumps and heat exchangers. For response to abnormal events, the SMR-300 safety systems are actuated.

1.6.1.3 Materials

Suitable materials for ESF SSCs will be selected based on the US SMR-300 category and classification of the components, and applicable codes and standards. Substantiation of material selection and assessment is described in Part B Chapters 18 Structural Integrity [13], Part B Chapter 19 Mechanical Engineering [10], and Part B Chapter 23 Reactor Chemistry [14].

1.6.1.4 Systems Reliability

The requirements of plant safety measures are developed through fault studies. These requirements drive the design of safety measures, to ensure that they are suitably reliable to perform their functions. A robust methodology for identification and assessment of fault conditions for the SMR-300 has been performed, based on the approach used in the US. This will be expanded to be in-line with UK RGP and the relevant ONR SAPs and TAGs, as presented in Part B Chapter 14 Safety and Design Basis Accident Analysis [12].

For system reliability, the safety components of the ESFs are designed against a single failure with redundancy and independence, and in accord with the Design Standard for Application of Single Failure Criterion [35]. This ensures that safety functions required for DBAs can be accomplished with acceptable reliability. System reliability is demonstrated through protection against:

- Single active failure.
- Single passive failure.
- Spurious valve actuation.
- Damage from fire, flood, dynamic effects.
- Environmental effects.

Differences between the US and UK legislation with regards to single failure criterion and discussed further in sub-chapter 1.7.2.3.3.

To prevent a common failure and assure protection for pipe breaks, missiles, and fires, safety-related valves and components will meet the grouping and separation requirements described in the SMR-300 Design Standard for Grouping and Separation [36].

Design descriptions of the ESFs are presented in the following sub-chapters, providing explanations of their reliability-related design aspects.

1.6.2 Passive Containment Heat Removal System (PCH)

Claim 2.2.2.1: Core decay heat removal is ensured following credible initiating events in all plant states.

Argument 2.2.2.1-A4: The PCH removes heat from the Containment atmosphere following an RCS energy release.

Evidence for Argument 2.2.2.1 – A4 is:

- **HI-2240170, System Design Description for Passive Containment Heat Removal System** [20]: This document provides a high-level description of the functional, performance and safety requirements of the PCH. It provides the classification, safety related and non-safety related functions of the PCH.

Claim 2.2.3.2: The RCS and ESF SSCs ensure the integrity of the Containment Structure following credible initiating events in all plant states.

Argument 2.2.3.2-A1: The PCH is designed to maintain the CS integrity by maintaining containment pressure and temperature below its design limits with sufficient margin during DBAs.

Argument 2.2.3.2-A2: The PCH instrumentation penetrations are designed to withstand containment design basis pressure.

Evidence for Arguments 2.2.3.2 – A1 and A2 is:

- **HI-2240170, System Design Description for Passive Containment Heat Removal System** [20]: This document provides a high-level description of the functional, performance and safety requirements of the PCH. It provides the classification, safety related and non-safety related functions of the PCH.

1.6.2.1 System Functions

1.6.2.1.1 Safety Functions

The safety functions of the PCH are to:

- **Maintain containment pressure and temperature below its design limits**

  As part of the general function to prevent or limit uncontrolled release of fission products to the environment, the PCH is designed to maintain the CS integrity by maintaining containment pressure and temperature below its design limits with sufficient margin during DBAs. The PCH reduces containment pressure by rejecting heat from the containment atmosphere to the environment.

- **Mitigate pressure-assisted release of fission products to the environment**

  As part of the general function to prevent or limit uncontrolled release of fission products to the environment, the PCH reduces containment pressure during DBAs. Reducing containment pressure reduces the driving force (pressure difference between the containment and environment) of uncontrolled release of fission products.

- **Maintain the Containment Structure required leak tightness**

  As part of the general function to prevent or limit uncontrolled release of fission products to the environment, the PCH instrumentation penetrations are designed to withstand containment design basis pressure.

- **Measure containment pressure**

  As part of the general function to prevent or limit uncontrolled release of fission products to the environment, the PCH measures containment pressure and provides measurement data to the PSS.

1.6.2.1.2 Non-Safety Functions

The non-safety functions of the PCH are to:

- **Provide chemistry control for the annular reservoir water.**

  Chemicals are injected into the PCH using the Chemical Addition Tank and Chemical Addition Pump and are circulated into the AR water, via the Recirculation Pump, to prevent biological growth and inhibit corrosion to the CES, CS, and SDH heat exchanger.

- **Provide freeze protection for the annular reservoir water.**

  The recirculation heater functions in conjunction with the recirculation pump to provide freeze protection for the AR water.

- **Provide a connection for makeup water source to the annular reservoir.**

  The PCH contains connections to the Demineralised Water System (DWS) and Diverse and Flexible Coping Strategies (FLEX) for the ability to supply makeup water to the AR. Note that the PCH can perform its safety functions without replenishing the AR water inventory.

- **Provide a means to monitor containment temperature.**

  The PCH monitors containment temperature to verify system performance. Containment temperature measurements are not a safety related input to PSS.

- **Provide a means to monitor AR water level.**

  The PCH monitors the water level in the AR to verify adequate inventory. AR water level measurements are not a safety related input to PSS.

- **Provide a means for de-watering the AR.**

  The PCH contains a de-watering line and associated equipment to provide the ability to remove water from the AR in order to perform inspections and repairs.

1.6.2.2 System Description

The PCH maintains the CS atmospheric pressure and temperature within design limits in the event of a postulated accident by utilising the metal CS and the water inventory in the AR. The PCH is a completely passive system that removes heat from the Containment atmosphere after an RCS energy release to Containment. It does not require any actuations or an actuation signal to perform its functions.

During a postulated high-energy release, steam is released into the Containment atmosphere. The steam condenses on the inside surface of the CS wall as heat is transferred through the CS wall to the AR. The large heat transfer area and high conductance of the steel CS results in rapid heat rejection from the Containment atmosphere to the AR maintaining the integrity of the CS. As the water in the AR is heated, it rejects heat to the environment through the discharge of clean vapor (no radioactivity) through the vent at the top of the CES. Water inventory is maintained in the AR during normal operation to assure PCH is available after a postulated accident without any operator action. The water inventory can be replenished using an accessible connection outside of Containment as part of FLEX implementation. Note that the PCH can perform its safety functions without replenishing the AR water inventory.

A Design Challenge Paper has been raised to address the potential risks associated with the loss of the annular reservoir. Discussion on this can be found in PSR part B Chapter 20 (Civil Engineering).

Figure 3 displays a simplified Process Flow Diagram [32] for the PCH system.

REDACTED

Figure 3: PCH Process Flow Diagram

The Annular Reservoir is a novel feature which is described in various PSR chapters, specifically Part B Chapter 14 Fault Studies [12], Part B Chapter 18 Structural Integrity [13], Part B Chapter 19 Mechanical Engineering [10], Part B Chapter 20 Civil Engineering [11] (lead chapter), and this chapter. A comprehensive overview of the Containment Structure system, including the AR, is presented in [48].

1.6.2.2.1 Instrumentation

Claim 2.2.5.3: The ESFs design incorporates instrumentation which ensures that relevant safety functions are delivered.

Argument 2.2.5.3-A1: The PCH has safety-related pressure instruments to measure containment pressure, monitored by the PSS and DAS.

Evidence for Argument 2.2.5.3 –A1 is:

- **HI-2240170, System Design Description for Passive Containment Heat Removal System** [20]: This document provides a high-level description of the functional, performance and safety requirements of the PCH. It provides the classification, safety related and non-safety related functions of the PCH.

The PCH safety-related pressure instruments measure containment pressure and are monitored by the PSS and DAS. During most plant operations, if the PSS detects High Containment Pressure it will generate an “S-Signal” to align ESFs to mitigate the effects of a High Energy Line Break [20].

1.6.2.2.2 Electrical

Claim 2.2.5.4: The ESFs design incorporates electrical systems which ensure that relevant safety functions are delivered.

Argument 2.2.5.4-A1: The PCH safety-related instruments are supplied with safety-related Class 1E power.

Evidence for Arguments 2.2.5.4 – A1 is:

- **HI-2240170, System Design Description for Passive Containment Heat Removal System** [20]: This document provides a high-level description of the functional, performance and safety requirements of the PCH. It provides the classification, safety related and non-safety related functions of the PCH.
- **HI-2240339, Part B Chapter 6, Electrical Engineering** [9]: Part B Chapter 6 presents the Claims, Arguments and Intended Evidence for Electrical Engineering that underpins the design of the generic SMR-300.

The PCH is a completely passive system. The system does not rely on electrical power supply to perform its safety functions related to containment heat removal.

The PCH instrumentation important to safety receive Class 1E DC power supply. The Class 1E I&C Power Distribution System (1E ICE) provides redundant, safety related DC power to plant loads that are relied upon to perform safety related functions. The Class 1E DC system comprises two redundant and independent divisions (Division-A and Division-B) to provide power to plant DC loads associated with the respective division [20].

1.6.2.2.3 Reliability

Claim 2.2.5.5: The safety components of the ESFs are designed to withstand expected environmental conditions and a single failure with redundancy and independence.

Argument 2.2.5.5-A1: PCH is designed to ensure components required to conduct safety functions meet redundancy and reliability requirements.

Evidence for Argument 2.2.5.5 – A1 is:

- **HI-2240170, System Design Description for Passive Containment Heat Removal System** [20]: This document provides a high-level description of the functional, performance and safety requirements of the PCH. It provides the classification, safety related and non-safety related functions of the PCH.

The PCH is designed in accordance with IEEE Standard 603 to address separation, diversity, and independence of SSCs important to safety. The portion of the PCH that performs safety-related functions is containment pressure instrumentation.

The AR is formed by the CS and CES which are seismic Category I structures. Penetrations for the recirculation line through the CS and CES are above the minimum required water level to prevent inadvertent draining of the AR water. Instrument penetrations below the minimum required water level are part of a closed system both inside and outside the CS. This system is designed to meet the applicable service piping standards and codes and continuously monitored for leaks to ensure the integrity of the AR water inventory against inadvertent draining.

The PCH recirculation piping inside the CES is segregated from seismic Category I equipment and components, such as the SDH heat exchanger, so that a recirculation pipe-break or rupture does not cause adverse system interactions or affect the integrity of the CS and CES.

Instruments required to measure containment pressure are redundant to meet the single failure criterion requirements of IEEE Standard 379 [20].

1.6.2.2.4 Inspection and Testing

Claim 2.2.5.6: The ESFs are designed with appropriate Inspection and Testing procedures to ensure that its design intent is met.

Evidence for Claim 2.2.5.6 is:

- **HI-2240170, Design Specification for Passive Containment Heat Removal System** [20]: This document provides a high-level description of the functional, performance and safety requirements of the PCH. It provides the classification, safety related and non-safety related functions of the PCH.
- **HPP-160-3016, SMR-300 Design Standard for Human Factors: Maintenance, Inspection and Testing** [37]: This document is a Design Guide, for the SMR-300, to assist designers in ensuring that human capabilities and limitations are considered during the design and layout of equipment.

The PCH is designed to permit appropriate periodic inspections to ensure the safety functions of the system including the ability to drain the AR for inspection. Testing and verifications are performed prior to plant operation to ensure that the system has adequate capacity to perform its intended function [20]. Further discussion regarding EMIT philosophy and management can be found in Part B Chapter 9 [38].

1.6.2.3 PCH Interfaces

The PCH interfaces with the following systems to assure that the design requirements are satisfied:

Table 9: System Interfaces with the PCH

System	Description	Function	PSR Chapter
CES	Containment Enclosure Structure	All ESFs are located inside the CES, The PCH utilises the CES top vent to transfer heat from the Containment atmosphere to environment.	Part B Chapter 20 Civil Engineering
CS	Containment Structure	The PCH transfers heat from the containment structure to the external environment.	Part B Chapter 20 Civil Engineering
SDH	Secondary Decay Heat Removal System	The AR water is the heat sink for the SDH.	Sub-chapter 1.6.3.2
DCE	DC Power Distribution System	The Class 1E DC System supplies power to the safety-related instrumentation of the PCH.	Part B Chapter 6 Electrical Engineering
PCS	Plant Control System	The PCS monitors the non-safety recirculation pump and heater parameters to establish and maintain the plant operating conditions within prescribed limits. The PCS provides control and display of non-safety related plant components.	Part B Chapter 4 Control and Instrumentation Systems
PSS	Plant Safety System	The PSS receives signals from the containment pressure instrumentation. The PSS provides plant safety related displays and initiates Reactor Trip and ESFAS in response to signals received from safety related instrumentation.	Part B Chapter 4 Control and Instrumentation Systems
[2]CIS	Containment Isolation System	The CIS allows or isolates normal or emergency passage of fluids through the containment envelope to prevent or limit the release of fission products from the containment atmosphere to the environment.	Sub-chapter 1.6.5
[2]CBV	Containment Ventilation System	The CBV provides containment pressure, energy, and radionuclide management for operational states. The CBV is designed to keep containment temperature within normal operating range. The CBV also consists of low and high purge systems to reduce containment global dose rate by venting airborne contamination	Part B Chapter 5 Reactor Supporting Facilities
[2]CGC	Combustible Gas Control System	The CGC monitors and controls the concentration of combustible gases that might be generated in the containment atmosphere following a Beyond Design Basis Accident (BDBA).	Part B Chapter 5 Reactor Supporting Facilities

1.6.3 Passive Core Cooling System (PCC)

Claim 2.2.1.1: There is provision in the design to ensure that the reactor can be shutdown, via boron control in the RCS, in all relevant plant modes.

[2] The PCH does not physically interface with the systems, but functions in conjunction to fulfil the design requirements of the containment system.

Argument 2.2.1.1-A1: The PCC is designed to provide passive, highly concentrated borated water safety injection and provide a means of negative reactivity insertion into the reactor core, providing a separate means of shutdown diverse from the CDS, following a postulated DBA.

Evidence for Argument 2.2.2.1 – A1 is:

- **HI-2240146, System Design Description for Passive Core Cooling System** [23]: This document provides a high-level description of the functional, performance and safety requirements of the PCC. It provides the classification, safety related and non-safety related functions of the PCC.
- **HI-2250569, UK GDA SMR-300 Secondary Shutdown Design** [49] This document provides the description and demonstration of the SMR-300 secondary means of shut down capability. It also presents discussion of areas of future further work for ALARP considerations.

Claim 2.2.3.1: The RCS and ESF SSCs ensure the integrity of the Reactor Coolant Pressure Boundary following credible initiating events in all plant states.

Argument 2.2.3.1-A2: ESFs SSCs which form the RCPB (ADS, PDH, and associated valves and piping) are designed to ensure its integrity following credible initiating events, isolating the reactor coolant from the Containment atmosphere and secondary cooling systems.

Evidence for Argument 2.2.3.1 – A2 is:

- **HI-2240146, System Design Description for Passive Core Cooling System** [23]: This document provides a high-level description of the functional, performance and safety requirements of the PCC. It provides the classification, safety related and non-safety related functions of the PCC.

1.6.3.1 System Functions

1.6.3.1.1 Safety Functions

The safety functions of the PCC are to:

- **Maintain Reactor Coolant Pressure Boundary Integrity**

  The PCC is designed such that any part of the system that cannot be isolated from the RCS, which is classified as part of the RCPB, is maintained to prevent or mitigate leakage in excess of the normal reactor coolant makeup capability.

- **Maintain Containment Boundary Integrity**

  The PCC is designed such that the closed system inside Containment provides a barrier and the closed system outside Containment as the second barrier for containment isolation. The PCC Containment penetrations are closed loops that function to effectively isolate and preserve the integrity of the Containment envelope.

- **Reactor Coolant System Inventory Control**

  The PCC is designed to provide abundant borated makeup to maintain RCS inventory to preclude the potential for significant core heat-up for the spectrum of postulated design basis LOCAs.

- **Reactor Coolant System Depressurisation**

  The PCC is designed to provide means of controlling the depressurisation of the RCS to permit passive safety injection following DBAs. The system also provides means to vent non-condensable gases from the RCS and prevent re-pressurisation of the RPV following a postulated DBA.

- **Reactor Coolant System Reactivity Control**

  The PCC is designed to provide passive, highly concentrated borated water safety injection and provide a means of negative reactivity insertion into the reactor core, separate and diverse from the CDS, following a postulated DBA.

- **Decay Heat Removal to Safe Shutdown Conditions**

  The PCC is designed to provide for the transfer of sensible and decay heat from the reactor core coolant to the ultimate heat sink. The passive decay heat removal systems provide for safety-grade decay heat removal from the RCS to transfer plant operations from full RCS operating pressures and temperatures to a safe stable condition for all plant conditions.

  The PCC is also designed to provide core decay heat removal and SFP makeup during a postulated DBA or whenever normal heat removal paths are lost.

- **LOCA Containment Flood-up Debris Filtration**

  The PCC is designed to filter any debris from the LOCA flood-up flow to the RPV to prevent blockage in the core and core cool-able geometry for emergency core cooling.

- **LOCA Long-term Cooling Water pH Maintenance**

  The PCC is designed to provide LOCA flood-up water pH control as required to maintain iodine fission products in solution

1.6.3.1.2 Non-Safety Functions

The non-safety functions of the PCC are to:

- **Provide Post-Accident Monitoring Capability**

  The PCC is designed to provide a reliable means of monitoring plant variables and systems by control room operators during accident situations.

- **Primary Decay Heat Removal System Degasification**

  The PCC is designed to provide the capability to perform system high-point venting and to continuously monitor gas accumulation in the RCPB portion of the PDH.

- **LOCA Long-term Cooling Water Sampling**

  The PCC is designed to allow for sampling of the LOCA flood-up as needed in the LOCA long-term cooling configuration.

- **Reactor Cavity Fill and Drain**

  The PCC is designed to provide a means of transferring borated water from the PCMWT to the SFP prior to refueling operations by gravity drain or by using SFC pumps to support refueling operations. The water from the SFP is returned to the PCMWT or the Refuelling Water Storage Tank (RWST) following refueling using the SFC pumps.

- **Emergency Borated Makeup to the Spent Fuel Pool**

  The PCC is designed to provide safety related makeup to the SFP located inside the Containment. The emergency makeup isolation valves and piping connected to the SFP are classified to maintain the integrity of the SFP boundary. Emergency borated makeup to the SFP is provided using water inventory of the PCMWT.

- **Leak Detection**

  The PCC is designed for detection of leakages from or to major PCC equipment, to include the PDH and SDH Heat Exchanger (HX), and the PCMWT.

1.6.3.2 System Description

The PCC is designed to provide emergency core heat removal and makeup water during postulated DBAs. The system uses passive means such as natural circulation, gravity injection, and compressed gas expansion for core makeup and cooling without the use of active components such as pumps. The PCC consists of the following sub-systems:

- PDH.
- SDH.
- ADS.
- PCM.

The PDH and SDH provide primary protection in the form of emergency decay heat removal when the normal heat rejection path is lost. Each decay heat removal system is capable of removing core decay heat and bringing the plant to a safe shutdown condition.

The ADS and PCM provide primary protection in the form of controlled depressurisation and safety injection, respectively, for DBAs resulting in loss of reactor coolant. The PCM and ADS will ensure sufficient RCS inventory to prevent fuel failure during the event and remove decay heat for the first 72 hours with no operator action required.

1.6.3.2.1 Higher Reliability Components

Claim 2.2.5.2: Where the integrity of ESFs SSCs is critical in preventing the release of radioactive material, methodology for High Reliability and Very High Reliability components is followed.

Argument 2.2.5.2-A1: Components within the RCS have been selected as Higher Reliability candidates when appropriate.

Evidence for Argument 2.2.5.2 – A1 is:

- **HI-2240349, Part B Chapter 18, Structural Integrity** [13]: Part B Chapter 18 presents the Claims, Arguments and Intended Evidence for Structural Integrity that underpins the design of the generic SMR-300.

The PCM Accumulators are listed as Candidate Higher Reliability Components, as seen in Part B Chapter 18 [13].

1.6.3.2.2 Instrumentation

Claim 2.2.5.3: The ESFs design incorporates instrumentation which ensures that relevant safety functions are delivered.

Argument 2.2.5.3-A2: The PCC has safety-related instrumentation, monitored by the PSS and DAS.

Evidence for Argument 2.2.5.3 – A2 is:

- **HI-2240338, Part B Chapter 4, Control and Instrumentation Systems** [7]**:** Part B Chapter 4 presents the Claims, Arguments and Intended Evidence for Control and Instrumentation that underpins the design of the generic SMR-300.
- **HI-2240146, System Design Description for Passive Core Cooling System** [23]: This document provides a high-level description of the functional, performance and safety requirements of the PCC. It provides the classification, safety related and non-safety related functions of the PCC.

The PCC has safety-related instrumentation which sends signals to the PSS to actuate automatic protective systems if necessary. PCC instrumentation is used for post-accident monitoring, via the Post Accident Monitoring System (PAM). [23]. Further detail of PCC instrumentation is given in the following sub-system descriptions.

1.6.3.2.3 Electrical

Claim 2.2.5.4: The ESFs design incorporates electrical systems which ensure that relevant safety functions are delivered.

Argument 2.2.5.4-A2: The PCC safety-related instrumentation is supplied with safety-related Class 1E power.

Evidence for Argument 2.2.5.4 – A2 is:

- **HI-2240146, System Design Description for Passive Core Cooling System** [23]: This document provides a high-level description of the functional, performance and safety requirements of the PCC. It provides the classification, safety related and non-safety related functions of the PCC.
- **HI-2240339, Part B Chapter 6, Electrical Engineering** [9]: Part B Chapter 6 presents the Claims, Arguments and Intended Evidence for Electrical Engineering that underpins the design of the generic SMR-300.

The PCC safety related-valves and instruments are supplied with safety-related Class 1E power, direct from battery packs to ensure that they will operate in an accident situation when normal sources of power are not available [23].

In accordance with U.S.NRC Branch Technical Position (BTP) 7-2 and 8-4, power is disconnected from the Accumulator Safety Injection Valves and the PDH Inlet Isolation Valve when required to ensure that inadvertent closure of these valves do not prevent accumulator injection or operation of the PDH.

1.6.3.2.4 Reliability

Claim 2.2.5.5: The safety components of the ESFs are designed to withstand expected environmental conditions and a single failure with redundancy and independence.

Argument 2.2.5.5-A2: PCC is designed to ensure components required to conduct safety functions meet redundancy and reliability requirements.

Evidence for Argument 2.2.5.5 – A2 is:

- **HI-2240146, System Design Description for Passive Core Cooling System** [23]: This document provides a high-level description of the functional, performance and safety requirements of the PCC. It provides the classification, safety related and non-safety related functions of the PCC.

PCC is designed to perform its safety functions despite the effects of fires, flooding, missiles, pipe whip, and external hazards. PCC components are designed to withstand adverse environmental conditions in containment including high ambient temperature, high humidity, and flooding [23].

Actuation valves for the SDH and PDH are arranged in parallel so that a failure of one valve to open will not prevent the systems from operating. PDH and SDH are sized such that either system operating alone can bring the plant to a safe shutdown condition. To protect the PDH decay heat removal capacity, a high point vent can be used to remove non-condensable gases that can accumulate and interfere with natural circulation in the subsystem [23].

ADS valves are arranged in series to prevent spurious actuation and leakage from the RCS. Having these valves in series means that the failure of one ADS valve renders that train of ADS inoperable. However, each stage of ADS is composed of two redundant trains (train A and train B) so that a single failure will not compromise the subsystem's ability to depressurise the RCS [23].

When accumulator injection is required to assure plant safety, the accumulator isolation valves are locked open and de-energised to prevent inadvertent isolation. The piping that connects each accumulator to the DVI line features two check valves. If one accumulator check valve leaks, the other check valve prevents back leakage from the RCS into the accumulator. The accumulator features redundant level and pressure instrumentation. [23].

Both trains (A and B) of the PCM have two parallel PCMWT injection valves in series with a PCMWT injection check valve. The failure of an injection valve to open will not prevent the other injection valve from opening and allowing flow from the PCMWT to the RPV. There are two recirculation lines (A and B) connecting the PCMWT to the SFP. Each line has two long-term cooling valves in series. If one valve opens spuriously the other valve prevents overflowing of the SFP from the PCMWT when long-term recirculation cooling is not required. If one of the recirculation valves fails to open, that train is disabled, but the other train will connect the PCMWT and SFP. Each DVI line connects to its own strainer in the PCMWT to assure a flow path from PCMWT to the RPV if one strainer fails. The PCMWT features redundant temperature and level instrumentation [23].

Further discussion regarding the application of single failure criterion in the UK context can be found in sub-chapter 1.7.2.3.

1.6.3.2.5 Testing and Inspection

Claim 2.2.5.6: The ESFs are designed with appropriate Inspection and Testing procedures to ensure that its design intent is met.

Evidence for Claim 2.2.5.6 is:

- **HI-2240146, System Design Description for Passive Core Cooling System** [23]: This document provides a high-level description of the functional, performance and safety requirements of the PCC. It provides the classification, safety related and non-safety related functions of the PCC.
- **HPP-160-3016, SMR-300 Design Standard for Human Factors: Maintenance, Inspection and Testing** [37]: This document is a Design Guide, for the SMR-300, to assist designers in ensuring that human capabilities and limitations are considered during the design and layout of equipment.

PCC [23] components are inspected and tested in accordance with ASME Boiler and Pressure Vessel Code Section XI. The PCC is designed in accordance with the Design Standard for Human Factors: Maintenance, Inspection and Testing [37]. A testing and inspection schedule will be developed post PSR Rev 1. Further discussion regarding EMIT philosophy and management can be found in Part B Chapter 9 [38].

1.6.3.2.6 Primary Decay Heat Removal System (PDH)

Claim 2.2.2.1: Core decay heat removal is ensured following credible initiating events in all plant states.

Argument 2.2.2.1-A5: The PDH and SDH are designed to remove decay heat from the reactor core whenever normal heat removal paths are lost.

Evidence for Argument 2.2.2.1 –A5 is:

- **HI-2240718 R1, SMR-300 Design Basis of Decay Heat Removal Systems** [50]: This document provides an evaluation of the decay heat removal systems (PDH and SDH) performance under limiting design basis events and AOOs of the SMR-300.
- **HI-2240146, System Design Description for Passive Core Cooling System** [23]: This document provides a high-level description of the functional, performance and safety requirements of the PCC. It provides the classification, safety related and non-safety related functions of the PCC.

The PDH is a single loop, natural circulation system which includes one heat exchanger, associated valves, piping, controls, and instrumentation. The system also includes component supports, attached piping, and vent and drain piping.

The PDH is a safety related system, designed to:

- Remove core decay heat whenever the normal heat removal path is not available.
- Remove core decay heat and RCS sensible heat to bring the plant to a safe shutdown condition.

The PDH system provides passive core cooling for non-LOCA accidents by removing core decay heat directly from RCS and rejecting it to the PCMWT. It does this by direct closed-loop decay heat transfer from the primary RCS to a PDH HX, located in the PCMWT. All system components are safely located and protected within the CS. During boiling of the PCMWT inventory, steam is released into Containment and heat is rejected to the AR, via the CS wall. The AR operates as the UHS and allows for continued decay heat removal. The PDH HX is located at an elevation above the core to maintain conditions between the hot leg and cold leg that contribute to natural circulation through the heat exchanger.

The PDH loop, which branches from the RCS hot leg piping, receives high temperature reactor coolant from the RCS hot leg and returns coolant, cooled by the PDH HX, to the SGE lower head. The PDH loop includes the standpipe high point vent line, the tube side of PDH HX, and vent and drain lines. The PDH actuation valves, which actuate on a PDH signal from the PSS, are also provided in the PDH loop, downstream of the PDH HX. An isolation valve is provided at the PDH inlet to isolate the PDH from the RCS in the event of leakage in the PDH loop. The PDH inlet isolation valve is normally open and de-energised during at-power operations.

The performance of the PDH is evaluated in the Design Basis of Decay Heat Removal Systems document [50].

Figure 4 displays a Process Flow Diagram [32] of the PDH.

[REDACTED]

Figure 4: PDH Process Flow Diagram

1.6.3.2.7 Secondary Decay Heat Removal System (SDH)

Claim 2.2.2.1: Core decay heat removal is ensured following credible initiating events in all plant states.

Argument 2.2.2.1-A5: The PDH and SDH are designed to remove decay heat from the reactor core whenever normal heat removal paths are lost.

Evidence for Argument 2.2.2.1 –A5 is:

- **HI-2240718 R1, SMR-300 Design Basis of Decay Heat Removal Systems** [50]: This document provides an evaluation of the decay heat removal systems (PDH and SDH) performance under limiting design basis events and AOOs of the SMR-300.
- **HI-2240146, System Design Description for Passive Core Cooling System** [23]: This document provides a high-level description of the functional, performance and safety requirements of the PCC. It provides the classification, safety related and non-safety related functions of the PCC.

Claim 2.2.3.2: The RCS and ESF SSCs ensure the integrity of the Containment Structure following credible initiating events in all plant states.

Argument 2.2.3.2-A3: The PCC Containment penetrations (for the SDH) are closed loops that function to effectively isolate and preserve the integrity of the Containment envelope.

Evidence for Argument 2.2.3.2 – A3 is:

- **HI-2240146, System Design Description for Passive Core Cooling System** [23]: This document provides a high-level description of the functional, performance and safety requirements of the PCC. It provides the classification, safety related and non-safety related functions of the PCC.

The SDH is a single loop, natural circulation system which includes one heat exchanger, associated valves, piping, controls, and instrumentation. The system also includes component supports, attached piping, and vent and drain piping.

The SDH equipment is located inside the CS and in the AR. The SDH loop penetrates the CS wall, however the loop is not provided with containment isolation valves, taking credit for the closed system outside the CS as a second barrier for containment isolation.

The system can satisfy the functional requirement of transferring decay heat to the AR, assuming a single failure in the Class 1E electrical system.

The SDH is a safety-related system, designed to:

- Remove core decay heat whenever the normal heat removal path is not available.
- Remove core decay heat and RCS sensible heat to bring plant to a safe shutdown condition.

The SDH provides additional passive core cooling for non-LOCA accidents, independent to the PDH, by removing core decay heat indirectly from the RCS, via the SGE shell side coolant, and rejecting it to the AR, via the SDH heat exchanger located in the reservoir. The SDH operates by condensing shell side steam and returning the condensate to the feedwater of the SGE, much like auxiliary steam systems of some current reactors but operating entirely passively. All system components are securely located within the CES.

Upon receipt of an actuation signal, the two parallel outlet valves, on the return leg of the SDH loop, automatically open, allowing steam from the shell side of the SGE to circulate to the SDH heat exchanger (SDH HX) located in the AR. The steam condenses in the SDH HX rejecting its latent heat to the water in the AR and the condensate from the SDH HX flows back to the shell side of the SGE. The two parallel valves are powered by Class 1E DC power. Only one valve needs to open for the system to function. The SDH heat exchanger is also sized to have sufficient heat removal capacity to maintain RCS pressure below the design pressure, thus precluding the actuation of PZR safety relief valves during a non-LOCA DBA.

The performance of the SDH is evaluated in the Design Basis of Decay Heat Removal Systems document [50].

SDH has been designed to the requirements of ANSI/ANS 56.2-1984. ANSI/ANS 56.2-1984 provides criteria for using closed systems (either inside or outside containment) as a containment isolation barrier including, in part, designing to Quality Group B and seismic Category I standards, providing overpressure protection, and providing the capability for leak testing.

Therefore, SDH is designed to a suitable code, commensurate to its safety requirement (SMR Class B, ASME Class 2). The pipework is Seismically qualified (SC-1) and design temperature and pressures exceed the containment design temperature and pressures. This will be conformed for both internal and external effects of the temperature and pressure. The SDH is a closed system and has no vent or drain line outside of the containment structure. The system will be protected against, or separated from, the effects of high-energy line breaks, missiles and other hazardous effects (to be confirmed in later hazard studies). The SDH is protected from the effects of overpressure through the main steam safety valves.

The system will be tested for leaks or inspected to confirm its integrity periodically. ANS-56.2 requires leak testing whilst RG 1.141 allows for inspection if operating temperatures and pressures exceed containment design basis. The specific approach to be adopted will be confirmed beyond GDA timescales.

Design and isolation requirements depend heavily on the potential for radioactive release. So, if the closed system outside containment connects to the primary side, strict design rules apply; whereas if it connects to the secondary side, isolation requirements may be relaxed unless a credible fault could cause radioactive contamination. Therefore, as SDH normally contains non-radioactive steam and water, the safety concern is generally lower, and containment isolation requirements are less severe. ANSI/ANS-56.2-1984 does not require the same level of design pressure/temperature qualification as for RCS-connected systems and the system may not be subject to full containment isolation valve requirements, unless a credible failure mechanism could cause the system could become radioactive under accident conditions (e.g. due to Steam Generator Tube Rupture (SGTR).

[REDACTED]

It is acknowledged that further work is required to cement the above claims in the UK regulatory context, therefore GDA Commitment C_Faul_103 has been raised in Part B Chapter 14 [12] to conduct full UK based safety analysis during which SDH failures will be assessed. [REDACTED].

Figure 5 displays a Process Flow Diagram [32] of the SDH.

[REDACTED]

Figure 5: SDH Process Flow Diagram

1.6.3.2.8 Automatic Depressurization System (ADS)

Claim 2.2.2.2: Sufficient coolant inventory is maintained following credible initiating events in all plant states.

Argument 2.2.2.2-A2: The ADS depressurises the RCS, allowing for water injection from the PCM.

Evidence for Argument 2.2.2.2 – A2 is:

- **HI-2240146, System Design Description for Passive Core Cooling System** [23]: This document provides a high-level description of the functional, performance and safety requirements of the PCC. It provides the classification, safety related and non-safety related functions of the PCC.
- **HI-2240165, System Design Description for Reactor Coolant System** [19]: This document provides a high-level description of the functional, performance and safety requirements of the RCS. It provides the classification, safety related and non-safety related functions of the RCS.

The ADS works in conjunction with the PCM, and is designed to provide the following functions:

- Depressurises the RCS, following a LOCA, to allow for water injection from PCM and then vent steam from the RCS to the Containment to remove decay heat from the core for long-term cooling.
- Provide sufficient venting path for long-term cooling to allow steam to vent from the RPV to Containment and prevent re-pressurisation of the RCS.

[REDACTED]

There are two trains of two valves per train in series, for a total of four valves per depressurisation stage. These normally closed, motor-operated valves form part of the RCPB and the configuration of two valves in series for each automatic depressurisation flow path minimises the possibility of leakage through RCPB.

The first stage of automatic depressurisation, ADS Stage 1, is designed to depressurise the RCS to a pressure which ensures safety injection from the accumulators. The injection from the accumulators is key in supporting a diverse means of shutdown (Claim 2.2.1.1-A1). Further discussion regarding the ALARP status maturity is presented in [REDACTED] with evidence presented in [49]. The ADS Stage 1 valves are mounted onto two separate trains of RCS piping, extending from the nozzles at the top of the PZR to the nozzles at the top of the South PCMWT. Piping from the two nozzles at the PCMWT each extend to the lower section of the tank to sparger devices, submerged in the PCMWT. This allows steam to be condensed in the tank, scrubbing radioactive nuclides from the RCS, and minimises the increase in peak pressure and temperature inside Containment.

The second and final stage of ADS, ADS Stage 2, is designed to depressurise the RCS below the static pressure of the PCMWT, near equilibrium with Containment pressure, to permit safety injection from the PCMWT. The ADS Stage 2 valves are mounted on two separate trains of RCS piping. An ADS Stage 2 train connects to each RCS hot leg. The discharge from the RCS hot leg through ADS Stage 2 vents directly to the Containment.

Each train of RCS piping is provided with two motor-operated ADS valves in series, for Stage 1 and Stage 2. The ADS valves are normally in the closed position, with temperature instruments downstream that provide indication of leakage through the valves.

Figure 6 displays a Process Flow Diagram [32] of the ADS.

[REDACTED]

Figure 6: ADS Process Flow Diagram

1.6.3.2.9 Passive Core Makeup Water System (PCM)

Claim 2.2.2.2: Sufficient coolant inventory is maintained following credible initiating events in all plant states.

Argument 2.2.2.2-A3: The PCM provides coolant makeup to the RCS via medium pressure injection (Accumulator), low pressure injection (PCMWT), and long-term cooling (PCMWT to RPV).

Evidence for Argument 2.2.2.2 – A3 is:

- **HI-2240146, System Design Description for Passive Core Cooling System** [23]: This document provides a high-level description of the functional, performance and safety requirements of the PCC. It provides the classification, safety related and non-safety related functions of the PCC.

Claim 2.2.2.3: Spent fuel heat removal is ensured following credible initiating events in all plant states.

Argument 2.2.2.3-A1: The PCM is designed to provide emergency makeup to the SFP.

Evidence for Argument 2.2.2.3 – A1 is:

- **HI-2240146, System Design Description for Passive Core Cooling System** [23]: This document provides a high-level description of the functional, performance and safety requirements of the PCC. It provides the classification, safety related and non-safety related functions of the PCC.

The PCM includes the accumulators, PCMWT, strainers, spargers and associated valves, piping, controls, and instrumentation.

The PCM is a safety related system, designed to perform the following functions:

- Provide coolant makeup to the RCS, following a LOCA, to assure that sufficient water inventory is maintained to limit fuel damage and permit adequate core cooling via:
    a) Accumulator medium pressure injection.
    b) PCMWT low pressure injection.
    c) Long-term cooling.
- Inject sufficient boron into the RCS for reactivity control and to achieve and maintain a safe, stable condition.
- Provide relief and condense steam from the PZR.
- Flood the reactor cavity for refuelling operations.

The PCM is designed to provide core cooling and makeup water to the core for a complete range of LOCAs to make up for coolant loss from the RCS and ensure sufficient RCS inventory to prevent fuel failure in the core and SFP.

Two injection lines, that connect to the RPV DVI nozzles, are provided. The injection lines are located 140 degrees apart, with one injection line per PCM train. Each PCM train connects the PCMWT and one accumulator to the RPV via a DVI line.

<u>Accumulator Medium Pressure Safety Injection</u>

For each train, the accumulator injection line branches from the DVI line. Two swing check valves, which form the RCPB at each accumulator injection line, are provided downstream of the accumulator isolation valve. Each accumulator is provided with a connection from the CVC for inventory makeup to the tanks and a connection from the Nitrogen Supply System (N2S) for gas charging.

Each accumulator tank is provided with a relief connection to protect against over-pressurisation due to back-leakage from the RCS, thermal expansion of fluid caused by changes in ambient conditions, or excessive makeup inventory. Each tank is provided with a drain connection to maintain level within technical specifications and to drain the tank for inspection and maintenance.

A sampling connection which branches from the accumulator drain line is provided downstream of the accumulator drain valve to allow for monitoring of tank water chemistry. The vent to containment atmosphere connection for each tank is provided in the N2S. Instruments are provided to each accumulator tank to monitor tank pressure and level. One accumulator and associated valves are located in each of the accumulator rooms inside Containment.

The actuation of the accumulators provides fresh cooling water that is highly borated. The injection of the highly borated water acts to shut down the core criticality and bring the core into a shutdown state providing a diverse means of shutdown for the plant. The secondary means of passive shutdown is discussed further in [REDACTED].

<u>PCMWT Low Pressure Safety Injection</u>

Each PCM train is provided with a recirculation line that penetrates the lower portion of the PCMWT from the North and South, and associated safety-related recirculation valves used to initiate the long-term cooling phase of PCM injection to the RCS. The Northside of the PCMWT features a recirculation connection that routes to the SFC, with associated isolation valves, to provide recirculation of the PCMWT inventory at power. The RWST purification pump provides the capability to drive the flow necessary to recirculate, filter and purify tank inventory, if necessary. This recirculation flow is sufficient to prevent stratification of boron concentration in the tank, The recirculation flow is returned to the tank through a distribution header to maximise mixing in the tank.

For each train of DVI piping, the PCMWT is provided with a Containment flood-up strainer, submerged in the tank, which filters flow to the DVI connection, from the lower tank.

The PCMWT and accumulators are sufficiently borated to ensure reactor core is maintained subcritical following a LOCA.

Water level is maintained in the PCMWT and accumulators. There is sufficient water in one train of accumulators and the PCMWT to ensure both the core and the SFP remain covered during a LOCA, for a minimum of 72 hours without operator action.

Instrumentation and controls are provided to continuously monitor level and temperature during plant operation to ensure availability during accident conditions.

Non-condensable Gas and Coolant Relief to PCMWT

The flanges provided to the top of the South PCMWT, above grade, connect to the discharge piping from:

- Each train of PSVs.
- Each train of the ADS Stage 1 valves.
- RPV head vent valves.
- PDH high point vent valves.
- RHR LTOP relief valve.
- CVC makeup & recirculation inlet.
- CVC letdown relief.

Figure 7 displays a Process Flow Diagram [32] of the PCM.

[REDACTED]

Figure 7: PCM Process Flow Diagram

1.6.3.3 PCC Interfaces

The PCC interfaces with the following systems to assure that the design requirements are satisfied:

Table 10: System Interfaces with the PCC

System	Description	Function	PSR Chapter
CVC	Chemical and Volume Control System	The CVC provides inventory make up for lost inventory, due to leakage, and makeup to PCM Accumulators and PCMWT.	Part B Chapter 5 Reactor Supporting Facilities
DAS	Diverse Actuation System	The DAS control safety-related valves upon PSS failure to actuate.	Part B Chapter 4 Control and Instrumentation Systems
DCE	DC Power Distribution System	The Class 1E DC System supplies power to the safety-related valves and instruments.	Part B Chapter 6 Electrical Engineering
ICE	I&C Power Distribution System	The ICE supplies power to all solenoid operated valves and instruments.	Part B Chapter 6 Electrical Engineering
LRW	Liquid Radwaste System	Drains in the PCC are routed to the equipment and floor drains subsystems of the LRW for processing.	Part B Chapter 13 Radioactive Waste Management
LVE	Low Voltage AC Distribution System	The PCC non-safety related valve motors are powered by the non-safety portion of the LVE (480 VAC).	Part B Chapter 6 Electrical Engineering
MFS	Main Feedwater System	The MFS supplies feedwater and interfaces with the secondary side of the SGE. The SDH condenses shell-side steam and returns it to the MFS.	Part B Chapter 5 Reactor Supporting Facilities
MSS	Main Steam System	The MSS safety valves (MSSV) provide overpressure protection for the SDH.	Part B Chapter 5 Reactor Supporting Facilities
N2S	Nitrogen Supply System	The N2S provides nitrogen to the PCM accumulators to maintain the required pressure of the tanks.	N/A
PAM	Post-Accident Monitoring System	The PAM receives instrument signals from the PCC for post-accident monitoring requirements.	Part B Chapter 4 Control and Instrumentation Systems
PCH	Passive Containment Heat Removal System	The PCH features the AR which removes heat from the SDH.	Sub-chapter 1.6.2
PCS	Plant Control System	Receives instrument signals (independent from the PSS) from the PCC but does not control the PCC.	Part B Chapter 4 Control and Instrumentation Systems
PSL	Primary Sampling System	The PSL allows for monitoring of parameters of the PCM accumulators and flooded Containment.	Part B Chapter 5 Reactor Supporting Facilities
PSS	Plant Safety System	The PSS provides signals to automatically open PCC actuation valves. The PSS also receives signal from the PCMWT level instrument to, coincident with actuation of ADS2, actuate the PCM recirculation valves.	Part B Chapter 4 Control and Instrumentation Systems
RCS	Reactor Coolant System	The RCS PSV provide overpressure protection for the PDH. The PDH and SDH remove decay heat from RCS during non-LOCA and small LOCA events. The PCM supplies emergency borated water to the RCS when the normal makeup supply is unavailable or insufficient.	Sub-chapter 1.5

System	Description	Function	PSR Chapter
		The ADS Stage 1 depressurises the RCS following a LOCA to ensure injection of emergency makeup water. The ADS Stage 2 allows for further, controlled blowdown of the RCS following accident conditions. Long term cooling configuration allows borated water to flow from the PCMWT to the RPV when PCMWT head at the DVI elevation exceeds the head in RPV at the DVI elevation. In this configuration, the containment flood-up formed from the SFP, PCMWT, and RCS are merged.	
RHR	Residual Heat Removal System	The RHR LTOP relief valve discharges to the PCMWT. The RHR provides overpressure protection for the PDH during low power operations.	Part B Chapter 5 Reactor Supporting Facilities
SFC	Spent Fuel Pool Cooling System	The PCM provides emergency makeup to the SFP. The PCM recirculation valves connect the PCM to the SFP, initiating long-term cooling. Condensate collected from the Containment wall flows to the SFP and then to the PCMWT.	Part B Chapter 5 Reactor Supporting Facilities

1.6.4 Main Control Room Habitability System (MCH)

Claim 2.2.3.3: Habitability of the main control room is ensured following credible initiating events in all plant states.

Argument 2.2.3.3-A1: MCH has been designed to ensure it delivers relevant safety functions.

Evidence for Argument 2.2.3.3 – A1 is:

- **HI-2146060, System Design Description for the Main Control Room Habitability System** [22]: This document provides a high-level description of the functional, performance and safety requirements of the MCH. It provides the classification, safety related and non-safety related functions of the MCH. This document is for the SMR-160 design but remains relevant for GDA as a description of the SMR-300 functions.

1.6.4.1 System Functions

The MCH provides an environment that is safe and comfortable for human occupancy in the Main Control Room (MCR) by maintaining temperature, pressure, and humidity in the MCR under all conditions. Part B Chapter 9 [51] provides a detailed description of the overall design of the MCR to present that it will provide an appropriate means of managing operational activities and delivery of safety functions. Part B Chapter 9 [38] outlines the claims, arguments and evidence that underpins and supports MCR design, operation and philosophy which this sub chapter supports.

1.6.4.1.1 Safety Functions

The safety functions of the MCH are to:

- **Provide habitability for maximum main control room occupancy in the event when CRV is unavailable for the duration of the accident (72 hours)**

  The BAP maintains:

  1. CO_2 concentration to less than 0.5 percent for the main control room occupants

- Temperature and humidity of the MCR within acceptable limits

2. Control Room Emergency Zone (CREZ) at a continuous positive pressure with respect to the surrounding areas.

- **Provide capability to isolate main control room**

  The CRV provides the capability to detect and isolate main control room to protect personnel in the CREZ from external fire, smoke, toxic gas, and airborne radioactivity.

- **Provide adequate protection to the main control room personnel to ensure radiation exposure is within acceptable limits**

  The MCH and the MCR design ensures the radiation exposure of main control room personnel throughout the duration of the DBA does not exceed the limits set by General Design Criterion 19* ("radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident.") [28].

  *For UK context on radiation limits, see Part B Chapter 10 Radiological Protection [52].

1.6.4.1.2 Non-Safety Functions

The non-safety functions of the MCH are to:

- **Provide habitability for MCR during all modes of operation (excluding the conditions when CRNVS is unavailable)**

  The CRV provides the following functions during all modes of operation (excluding the conditions when CRV is unavailable):

  1. The CRV maintains the temperature and humidity in the MCR within acceptable limits.
  2. The CRV provides the capability to provide normal air filtration for the main control room.
  3. The CRV maintains the CREZ at a continuous positive pressure with respect to the surrounding areas.
  4. Isolate the main control room from outside and provide 100% recirculation air in the event of smoke in outside air intake.
  5. Provide smoke removal capability from the MCR.
  6. Provide low leakage envelope construction and leak tight penetrations for the CREZ and provide provision to test the leakage from the CREZ.

1.6.4.2 System Description

The MCH is designed to maintain an environment that is comfortable and permits continuous occupancy for personnel within the CREZ. The MCH comprises the following systems:

- CRV.
- BAP.

1.6.4.2.1 Control Room Normal Ventilation System (CRV)

The CRV is designed to provide a reliable source of heating, ventilation, and cooling to the areas served when AC power is available. The CRV serves the CREZ and the remaining areas in the Control Building (CB). The CREZ envelope consists of the main control room, shift supervisor's office, operation work area, operator washroom, kitchen, conference room, filter bank room and the operations break room area.

Outside supply air is provided to the CRV through an outside air intake duct that is protected by an intake enclosure. The outside air intake is located away from the plant vent discharge. The supply, return, and toilet and kitchen exhaust HVAC penetrations in the MCR envelope include redundant safety-related seismic Category I isolation dampers that are physically located within the main control room envelope. Those portions of the normal ventilation system which penetrate the main control room envelope are safety-related and designed as seismic Category I to provide isolation of the main control room envelope from the surrounding areas and outside environment in the event of a DBA.

Redundant safety-related radiation monitors sample line connections are located upstream of the outside air intake isolation dampers. These monitors initiate operation of the non-safety-related supplemental air filtration unit on high gaseous radioactivity concentrations and isolate the CREZ on high-high particulate or iodine radioactivity concentrations.

The CRV equipment and component functional capabilities are to minimise the potential for actuation of the BAP or the potential reliance on passive equipment cooling. This is achieved by the use of redundant equipment and components that are connected to standby onsite AC power sources.

If very high radioactivity levels are detected downstream of the air filtration unit or at the outside air intake are detected by the redundant radiation monitors, the outside air intake duct and the CREZ are automatically isolated. Under this condition, the normal ventilation system is automatically stopped, and the BAP is automatically placed in service.

Upon loss of normal AC power to both CRV air handling units for a short period of time and if the power is not restored within this short period, then the BAP is placed in service. The time delay provides a reasonable time for power restoration and therefore may prevent an unnecessary initiation of the BAP. Within 72 hours, if the power is restored and if the CRV becomes available, then CRV can be utilised to provide heating, ventilation, and cooling to the MCR.

1.6.4.2.2 Breathing Air and Pressurisation System (BAP)

The BAP is capable of providing emergency ventilation and pressurisation for the CREZ for the duration of the accident (72 hours), when CRV is not available. The BAP and the structures surrounding the CREZ maintain the temperature and humidity in the CREZ within acceptable limits for the duration of the accident. The emergency habitability system maintains the CREZ at a continuous positive pressure with respect to the surrounding areas for the duration of the accident.

The BAP consists of several packages of emergency air storage tanks that passively supply air to the CREZ when this system is actuated. The air storage tanks are sized to deliver the required air flow and induce sufficient air flow through the passive filtration line to meet the ventilation and pressurisation requirements for 72 hours.

1.6.4.2.3 Reliability

Claim 2.2.5.5: The safety components of the ESFs are designed to withstand expected environmental conditions and a single failure with redundancy and independence.

Argument 2.2.5.5-A3: The safety-related portions of the CRV and BAP are designed to ensure the reliability of the systems for all abnormal operations and are designed against a single failure with redundancy.

Evidence for Argument 2.2.5.5 – A3 is:

- **HI-2146060, System Design Description for the Main Control Room Habitability System** [22]: This document provides a high-level description of the functional, performance and safety requirements of the MCH. It provides the classification, safety related and non-safety related functions of the MCH.

The safety-related portions of the CRV and BAP [22] are designed to ensure the reliability of systems for all abnormal operations, including the postulated DBA conditions. They are also designed against a single failure with redundancy, in accordance with the SMR-300 Design Standard for Grouping and Separation [36] and the Design Standard for Application of Single Failure Criterion [35]. Further discussion on the suitability of the HVAC system for UK deployment is presented within Part B Chapter 5 [8] and Part B Chapter 19 [10].

Part B Chapter 19 describes the post-GDA Commitment, C_Mech_028, which proposes to review and incorporate UK RGP into the design of HVAC SSCs in the SMR-300 where practicable.

1.6.4.2.4 Inspection and Testing

Claim 2.2.5.6: The ESFs are designed with appropriate Inspection and Testing procedures to ensure that its design intent is met.

Evidence for Claim 2.2.5.6 is:

- **HI-2146060, System Design Description for the Main Control Room Habitability System** [22]: This document provides a high-level description of the functional, performance and safety requirements of the MCH. It provides the classification, safety related and non-safety related functions of the MCH.
- **HPP-160-3016, SMR-300 Design Standard for Human Factors: Maintenance, Inspection and Testing** [37]: This document is a Design Guide, for the SMR-300, to assist designers in ensuring that human capabilities and limitations are considered during the design and layout of equipment.
- **Regulatory Guide 1.197, "Demonstrating Control Room Envelope Integrity at Nuclear Power Reactors"** [53]: The SMR-300 is designed to this U.S.NRC RG which provides methods for demonstrating that the Control Room Envelope (CRE) at nuclear power plants maintains its integrity under both normal and accident conditions.

The safety-related mechanical components of the MCH systems [22] are inspected in accordance with the ASME BPV Code Section XI.

Testing includes pre-operational and startup testing. The integrity of the CREZ pressure boundary will be demonstrated in accordance with RG 1.197 [53].

In-service inspection and testing schedule will be developed during detailed design. Further discussion regarding EMIT philosophy and management can be found in Part B Chapter 9 [38].

1.6.4.3 MCH Interfaces

The MCH interfaces with the following systems to assure that the design requirements are satisfied:

Table 11: System Interfaces with the MCH

System	Description	Function	PSR Chapter
CAI	Instrument and Service Air System	The CAI provides an air supply to Air Operated Valves and Air Operated Dampers	N/A
CWS	Chilled Water System	The CWS provides cooling water supply to Air Handling Units	N/A
DCE	DC Power Distribution System	The Class 1E DC System supplies power to the safety-related motor operated valves.	Part B Chapter 6 Electrical Engineering
ICE	I&C Power Distribution System	The ICE supplies power to the instrumentation and AOVs and AODs.	Part B Chapter 6 Electrical Engineering
LVE	Low Voltage AC Distribution System	The LVE supplies power to non-safety valve motors	Part B Chapter 6 Electrical Engineering
MVE	Medium Voltage AC Distribution System	The MVE supplies power to the CRV fan motors.	N/A
PCS	Plant Control System	Receives instrument signals (independent from the PSS) from the PCC but does not control the PCC.	Part B Chapter 4 Control and Instrumentation Systems
PFA	Plant Fire Alarm System	Smoke detectors located in the CRV provides signals to the PFA.	N/A
PSS	Plant Safety System	The PSS provides signals to automatically open PCC actuation valves. The PSS also receives signal from the PCMWT level instrument to, coincident with actuation of ADS2, actuate the PCM recirculation valves.	Part B Chapter 4 Control and Instrumentation Systems
RMS	Radiation Monitoring System	RMS monitors outside air intake and provides signals to perform isolation/actuation.	Part B Chapter 4 Control and Instrumentation Systems

1.6.5 Containment Isolation System (CIS)

Claim 2.2.3.2: The RCS and ESF SSCs ensure the integrity of the Containment Structure following credible initiating events in all plant states.

Argument 2.2.3.2-A4: The safety function of the CIS is to isolate Containment to prevent or limit the release of fission products to the public and the environment in the event of a design basis accident.

Evidence for Argument 2.2.3.2 – A4 is:

- **HI-2146278, System Design Description for the Containment Isolation System** [21]: This document provides a high-level description of the functional, performance and safety requirements of the CIS. It provides the classification, safety related and non-safety related functions of the CIS.

1.6.5.1 System Functions

The CIS is an ESF that isolates the containment boundary to prevent or limit the release of fission products to the public and the environment during accidents. The CIS is not a discrete system; it is a grouping of valves and piping penetrating containment from other process systems that perform the containment isolation function.

1.6.5.1.1 Safety Functions

The safety function of the CIS is to isolate Containment to prevent or limit the release of fission products to the public and the environment in the event of a design basis accident.

The U.S.NRC requires piping systems, that penetrate the primary reactor containment, to be provided with isolation capabilities. With requirements for containment isolation function as per General Design Criteria 54, 55, 56, and 57 from 10 CFR 50 Appendix A [28].

1.6.5.1.2 Non-Safety Functions

The CIS does not perform any non-safety functions. The size of CIS valves and the associated piping are determined as part of the performance functions of the system in which they belong.

1.6.5.2 System Description

Containment isolation is typically provided by two valves, in series, in each containment penetration line, with one valve inside (inboard) and one valve outside (outboard) the Containment. For other CIS valve arrangements, only one isolation valve is provided, such as the MSS header and Steam Generator Blowdown line. In addition, there are closed systems outside Containment that meet certain criteria and are credited as one of the two containment isolation barriers i.e. SDH. The other barrier is the closed system inside containment.

Power operated valves that perform the containment isolation function are automatically closed via a signal from the PSS. Most valves are closed on a Safeguards Actuation Signal (S-Signal) or as a result of an S-Signal. however, some valves that are part of non-safety systems that can provide inventory to the RCS will only isolate when high radioactivity is detected in containment. All power operated valves can also be manually closed remotely from the MCR.

The actuators for power-operated CIS valves inside Containment are protected from containment flood water. All CIS valves and piping inside Containment are protected from the effects of pipe rupture and missiles.

The motor operated CIS valves and solenoids for air operated CIS valves are powered by the Class 1E DC System.

1.6.5.2.1 Reliability

Claim 2.2.5.5: The safety components of the ESFs are designed to withstand expected environmental conditions and a single failure with redundancy and independence.

Argument 2.2.5.5-A4: CIS is designed to ensure components required to conduct safety functions meet redundancy and reliability requirements.

Evidence for Argument 2.2.5.5 – A4 is:

- **HI-2146278, System Design Description for the Containment Isolation System** [21]: This document provides a high-level description of the functional, performance and safety requirements of the CIS. It provides the classification, safety related and non-safety related functions of the CIS.

CIS electrical, I&C, and mechanical equipment redundancy shall be incorporated in the design of the CIS [21]. Mechanical redundancy shall be provided by two barriers. When actuation of two power operated isolation valves on the same penetration is required, electrical redundancy shall be provided by independent power sources to each valve. Typically, the inboard CIS valve is powered and controlled by Division A of the electrical and I&C systems; and the outboard isolation valve is powered and controlled by Division B of the electrical and I&C systems.

1.6.5.2.2 Inspection and Testing

Claim 2.2.5.6: The ESFs are designed with appropriate Inspection and Testing procedures to ensure that its design intent is met.

Evidence for Claim 2.2.5.6 is:

- **HI-2146278, System Design Description for the Containment Isolation System** [21]: This document provides a high-level description of the functional, performance and safety requirements of the CIS. It provides the classification, safety related and non-safety related functions of the CIS.
- **HPP-160-3016, SMR-300 Design Standard for Human Factors: Maintenance, Inspection and Testing** [37]: This document is a Design Guide, for the SMR-300, to assist designers in ensuring that human capabilities and limitations are considered during the design and layout of equipment.

In-service inspection and testing of the CIS [21] valves shall be in accordance with ASME BPV Code Section XI. Pre-service inspection and pre-operational testing requirements will be addressed during detailed design. Further discussion regarding EMIT philosophy and management can be found in Part B Chapter 9 [38].

1.6.5.3 CIS Interfaces

The CIS interfaces with penetrations on the following systems to assure that the design requirements are satisfied:

Table 12: System Interfaces with the CIS

System	Description	PSR Chapter
CAI	Instrument and Service Air System	N/A
CBV	Containment Ventilation System	Part B Chapter 5 Reactor Supporting Facilities
CCW	Component Cooling Water	N/A
CLR	Containment Leak Rate Testing System	N/A
CS	Containment Structure	Part B Chapter 20 Civil Engineering
CVC	Chemical and Volume Control System	Part B Chapter 5 Reactor Supporting Facilities
CWS	Chilled Water System	N/A
DAS	Diverse Actuation System	Part B Chapter 4 Control and Instrumentation Systems
DWS	Demineralised Water System	N/A
FPS	Fire Protection System	Part B Chapter 5 Reactor Supporting Facilities

System	Description	PSR Chapter
GRW	Gaseous Radioactive Waste System	Part B Chapter 13 Radioactive Waste Management
LRW	Liquid Radwaste System	Part B Chapter 13 Radioactive Waste Management
MFS	Main Feedwater System	Part B Chapter 5 Reactor Supporting Facilities
MSS	Main Steam System	Part B Chapter 5 Reactor Supporting Facilities
PSL	Primary Sampling System	Part B Chapter 5 Reactor Supporting Facilities
RHR	Residual Heat Removal System	Part B Chapter 5 Reactor Supporting Facilities
SDH	Secondary Decay Heat Removal System	Sub-chapter 1.6.3
SFC	Spent Fuel Pool Cooling System	Part B Chapter 5 Reactor Supporting Facilities
SGB	Steam Generator Blowdown System	N/A
I&C Systems	Typical	Part B Chapter 4 Control and Instrumentation Systems
Electrical Systems	Typical	Part B Chapter 6 Electrical Engineering

1.6.6 ESFs CAE Summary

The ESFs, described in the preceding sub-chapter, demonstrates the following claims:

Claim 2.2.1.1: There is provision in the design to ensure that the reactor can be shutdown, via boron control in the RCS, in all relevant plant modes.

Argument 2.2.1.1-A1: The PCC is designed to provide passive, highly concentrated borated water safety injection and provide a means of negative reactivity insertion into the reactor core, providing a separate means of shutdown diverse from the CDS, following a postulated DBA.

The description of the PCC, as detailed in the SDD [23], demonstrates that the plant can be shut down via boron control, specifically by utilising ADS1 and medium pressure safety injection via the accumulators. The system is designed to provide negative reactivity via boron control in all plant states as required following an initiating event. As UK safety analysis matures, confidence in meeting this claim as required in the UK context will increase i.e. meeting UK single failure criterion requirements. However, it is assessed that this claim is met to the expected maturity at PSR. Further discussion can be found in [REDACTED], including references to supporting documentation.

Claim 2.2.2.1: Core decay heat removal is ensured following credible initiating events in all plant states.

Argument 2.2.2.1-A4: The PCH removes heat from the Containment atmosphere following an RCS energy release.

Argument 2.2.2.1-A5: The PDH and SDH are designed to remove decay heat from the reactor core whenever normal heat removal paths are lost.

SMR-300 is designed to passively remove decay heat following initiating events in all plant states. This is achieved through interaction of the PCC (including but not limited to PDH and SDH) and PCH. The performance of the PDH and SDH is evaluated in the Design Basis of Decay Heat Removal Systems document [50] which outlines the design basis specifics which have been simulated and verified through transient analysis. Given the system design and function is well understood, it is assessed that this claim is met to a maturity appropriate for PSR.

Claim 2.2.2.2: Sufficient coolant inventory is maintained following credible initiating events in all plant states.

Argument 2.2.2.2-A2: The ADS depressurises the RCS, allowing for water injection from the PCM.

Argument 2.2.2.2-A3: The PCM provides coolant makeup to the RCS via medium pressure injection (Accumulator), low pressure injection (PCMWT), and long-term cooling (PCMWT to RPV).

During accident conditions, when required, the ADS depressurises the plant which enables PCM to provide coolant via medium or low-pressure injection followed by long-term cooling. Sufficient coolant inventory is held within the CS and can be injected passively to provide sufficient cooling for 72 hours without operator input [19], [23]. Although further UK-based safety analysis is required to verify fault progression and system responses, it is assessed that this claim is met to a maturity appropriate for PSR.

Claim 2.2.2.3: Spent fuel heat removal is ensured following credible initiating events in all plant states.

Argument 2.2.2.3-A1: The PCM is designed to provide emergency makeup to the SFP.

The PCC, specifically PCM, is designed such that during a LOCA the PCMWT and SFP are connected through the long-term recirculation cooling line. The PCMWT can then supply water to the SFP for emergency makeup via the Spent Fuel Pool Cooling System [23], meeting the above claim to the maturity expected at PSR.

Claim 2.2.3.1: The RCS and ESF SSCs ensure the integrity of the Reactor Coolant Pressure Boundary following credible initiating events in all plant states.

Argument 2.2.3.1-A2: ESFs SSCs which form the RCPB (ADS, PDH, and associated valves and piping) are designed to ensure its integrity following credible initiating events, isolating the reactor coolant from the Containment atmosphere and secondary cooling systems.

ESF SSCs including ADS, PDH and associated valves and piping are designed to ensure RCPB integrity following credible initiating events, isolating the reactor coolant from the Containment atmosphere and secondary cooling systems [19]. Following further UK safety analysis, a definitive list of initiating events will be produced. However, utilising preliminary UK analysis [46], initial UK DBAA [47] and US based work, confidence is provided that this claim will be fully met throughout the next phase of safety case development. This claim is currently assessed as met to a maturity appropriate for PSR.

Claim 2.2.3.2: The RCS and ESF SSCs ensure the integrity of the Containment Structure following credible initiating events in all plant states.

Argument 2.2.3.2-A1: The PCH is designed to maintain the CS integrity by maintaining containment pressure and temperature below its design limits with sufficient margin during DBAs.

Argument 2.2.3.2-A2: The PCH instrumentation penetrations are designed to withstand containment design basis pressure.

Argument 2.2.3.2-A3: The PCC Containment penetrations (for the SDH) are closed loops that function to effectively isolate and preserve the integrity of the Containment envelope.

Argument 2.2.3.2-A4: The safety function of the CIS is to isolate Containment to prevent or limit the release of fission products to the public and the environment in the event of a design basis accident.

Through the arguments above and supported by the system safety functions and SDDs [20], [23], [21], it is evaluated that the above claim is met to a maturity appropriate at PSR. The importance of maintaining a secure CS is paramount to nuclear safety in all plant states during all credible events. To further demonstrate the importance of the CS to nuclear safety, the Containment Structure Systems Based View document has been produced [48] which provides a holistic view regarding the integrity of the CS and the role ESFs play in maintaining a secure structure.

[REDACTED]. Further information and discussion about the CS and this design challenge paper is contained PSR part B chapter 20.

As outlined in Section 1.6.3.2.7, further work is required to substantiate SDH within UK context with regards to preventing radioactive releases. This work is captured under GDA Commitment C_Faul_103 raised in Part B Chapter 14 [12] to conduct full UK based safety analysis during which SDH failures will be assessed.

Claim 2.2.3.3: Habitability of the main control room is ensured following credible initiating events in all plant states.

Argument 2.2.3.3-A1: MCH has been designed to ensure it delivers relevant safety functions.

The MCH has been designed with clear safety functions to enable the habitability of the MCR following credible initiating events [22]. Further design substantiation and UK based safety analysis will be performed to ensure the system meets UK legislative requirements including radiation limits [52]. Part B Chapter 19 describes the post-GDA Commitment, C_Mech_028, which proposes to review and consider UK RGP in the design of HVAC SSCs for deployment of the SMR-300 in the UK, where practicable. Despite the further work to be conducted to develop confidence in the MCH, it has been assessed that this claim is met to level of maturity appropriate at PSR.

Claim 2.2.5.2: Where the integrity of ESFs SSCs is critical in preventing the release of radioactive material, methodology for High Reliability and Very High Reliability components is followed.

Argument 2.2.5.2-A1: Components within the ESFs have been selected as Higher Reliability candidates when appropriate.

The PCM Accumulators are candidate higher reliability components [13], therefore, meeting this claim to the maturity expected at PSR. Throughout safety case and design development, higher reliability candidates will be reassessed as safety analysis matures.

Claim 2.2.5.3: The ESFs design incorporates instrumentation which ensures that relevant safety functions are delivered.

Argument 2.2.5.3-A1: The PCH has safety-related pressure instruments to measure containment pressure, monitored by the PSS and DAS.

Argument 2.2.5.3-A2: The PCC has safety-related instrumentation, monitored by the PSS and DAS.

The PCH has safety-related pressure instruments to measure containment pressure, monitored by the PSS and DAS. During most plant operations, if the PSS detects High Containment Pressure it will generate an “S-Signal” to align ESFs to mitigate the effects of a High Energy Line Break [20]. The PCC has safety-related instrumentation which sends signals to the PSS to actuate automatic protective systems if necessary [23]. Therefore, meeting this claim to the maturity expected at PSR. Throughout safety case and design development, safety functions of instrumentation will be reassessed as safety analysis matures.

Claim 2.2.5.4: The ESFs design incorporates electrical systems which ensure that relevant safety functions are delivered.

Argument 2.2.5.4-A1: The PCH safety-related instruments are supplied with safety-related Class 1E power.

Argument 2.2.5.4-A2: The PCC safety-related instrumentation is supplied with safety-related Class 1E power.

The PCH safety-related instruments are supplied with safety-related Class 1E power, direct from battery packs to ensure that they will operate in an accident situation when normal sources of power are not available [20]. The PCC safety related-valves and instruments are supplied with safety-related Class 1E power, direct from battery packs to ensure that they will operate in an accident situation when normal sources of power are not available [23]. Therefore, meeting this claim to the maturity expected at PSR. Throughout safety case and design development, electrical system classification and required supplies to instrumentation will be reassessed as safety analysis matures.

Claim 2.2.5.5: The safety components of the ESFs are designed to withstand expected environmental conditions and a single failure with redundancy and independence.

Argument 2.2.5.5-A1: PCH is designed to ensure components required to conduct safety functions meet redundancy and reliability requirements.

Argument 2.2.5.5-A2: PCC is designed to ensure components required to conduct safety functions meet redundancy and reliability requirements.

Argument 2.2.5.5-A3: The safety-related portions of the CRV and BAP are designed to ensure the reliability of the systems for all abnormal operations and are designed against a single failure with redundancy.

Argument 2.2.5.5-A4: CIS is designed to ensure components required to conduct safety functions meet redundancy and reliability requirements.

Throughout sub-chapter 1.6, system reliability and redundancy are demonstrated and the evidence presented. Further work will be performed to ensure the design satisfies redundancy and reliability requirements, including single failure criterion, with regards UK context. This work is currently being managed via the DAC and is explained further in sub-chapter 1.7.2.3. At this stage, the claim is not fully met due to the requirement to provide additional evidence to substantiate the design for UK deployment and verify that system designs are appropriate through UK based safety analysis.

Claim 2.2.5.6: The ESFs are designed with appropriate Inspection and Testing procedures to ensure that its design intent is met.

ESFs have been designed with inspection and testing requirements in mind. As detailed in the sub-chapter above, an understanding of inspection and testing requirements is being developed in accordance with the appropriate codes and standards. A full maintenance schedule will be developed as both the design of ESFs and supporting safety case are developed. This claim is not yet fully met noting that at this stage of safety case and design development, a maintenance schedule is not expected. However, as outlined in Part B Chapter 9 [38], the SMR-300 and its design management arrangements have been developed to ensure that effective EIMT will be delivered, ensuring that appropriate activities to support EIMT are included, that the schedule for these activities is appropriate for SSCs to continue to deliver their design basis, that there is sufficient space to conduct these activities, material selection is appropriate and that the inspection methods incorporated can identify degradation of SMR-300 SSCs. Therefore, with an early understanding of what is required from a maintenance program, the claim demonstrated to a maturity level appropriate for PSR.

1.7 CHAPTER SUMMARY AND CONTRIBUTION TO ALARP

This sub-chapter presents an overall summary and conclusion of the Reactor Coolant System and Engineered Safety Features chapter, and how this chapter contributes to the overall demonstration of ALARP for the generic SMR-300. Part A Chapter 5 [15] sets out the overall approach for demonstration of ALARP and how contributions from individual chapters are consolidated.

This sub-chapter therefore consists of the following elements:

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

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

1.7.1 Technical Summary

Part B Chapter 1 aims to demonstrate the following Level 3 claims:

Claim 2.2.1: Adequate provision for the control of reactivity is incorporated into the design of the reactor systems, engineered safety features, and fuel and core design.

Claim 2.2.1 is partially demonstrated within this chapter and partially demonstrated within other PSR chapters, in accordance with the Level 4 claims summarised in Table 1. The Level 4 claim (2.2.1.1) discussed within this chapter is demonstrated to a maturity expected at PSR Revision 1 by the design of the PCC and relevant safety functions.

Claim 2.2.2: Adequate provision for the removal of heat from the reactor core and spent fuel pool is incorporated into the design of the reactor systems, engineered safety features, and fuel and core design.

Claim 2.2.2 is partially demonstrated within this chapter and partially demonstrated within other PSR chapters, in accordance with the Level 4 claims summarised in Table 2. Core decay heat removal, relating to Claims 2.2.2.1, 2.2.2.2, and 2.2.2.3, is ensured through the RCS, presented in 1.5, and the ESFs, presented in 1.6. Justification of the core design, relating to Claim 2.2.2.4 is presented in Part B Chapter 2 (Reactor). All Level 4 claims discussed within this chapter are met to a maturity expected as PSR Revision 1 through system design and safety functions. It is recognised that post GDA as system design and UK based safety analysis are developed design changes to systems may be required to meet UK regulatory requirements. Where risk is already recognised, a Design Challenge Paper has been raised and is explained further in sub-chapter 1.7.2.3.3 ([REDACTED]).

Claim 2.2.3: Adequate provision for the control of radiation exposure and control of release of radioactive material is incorporated into the design of the reactor systems, supporting facilities, engineered safety features, and fuel and core design.

The above claim is partially demonstrated from this chapter and partially demonstrated within other PSR chapters, in accordance with the Level 4 claims summarised in Table 3. The supporting Level 4 claims discussed within this chapter are demonstrated to an expected maturity appropriate at PSR by the isolation and integrity safety functions of each system described above. Post PSR further work will be conducted to justify system claims to ensure UK regulatory requirements are met.

Claim 2.2.4: The Reactor Coolant System and Connected Systems are designed to ensure they deliver relevant safety features, supported by substantiation which is suitably mature.

Claim 2.2.4 is decomposed into six Level 4 claims which all are discussed within this chapter. All the claims are demonstrated to a maturity appropriate for PSR by system design, functions and safety features highlighted above. Despite no inspection and testing schedule for SSCs being produced, an understanding on the requirements and codes to be used is developed which is line with what is expected at this stage of safety case and design development.

Claim 2.2.5: The Engineered Safety Features are designed to ensure they deliver relevant safety features, supported by substantiation which is suitably mature.

The above claim is decomposed into six Level 4 claims which are all discussed within this chapter. All Level 4 claims except for 2.2.5.5 are demonstrated to a maturity appropriate for PSR. Claim 2.2.5.5 is not fully demonstrated as further work is required to ensure the design meets redundancy and reliability requirements, including single failure criterion, with regards to the UK regulatory context. This work is currently being managed via the DAC and is explained further in sub-chapter 1.7.2.3.

1.7.2 ALARP Summary

1.7.2.1 Review against Relevant Good Practice

The design of the SMR-300 SSCs complies with RGP and U.S.NRC requirements applicable in the US. The design adopts nuclear-specific codes and standards endorsed by the U.S.NRC and internationally recognised bodies such as International Atomic Energy Agency (IAEA). The principal codes and standards identified within sub-chapter 1.4 are considered RGP by the UK nuclear industry. This is based on existing practices adopted on UK nuclear licensed sites, application in earlier and successful GDAs, as well as recognition as RGP by ONR TAGs. The summary statements against specific SSCs and Codes & Standards are aligned with sub-chapter 1.4.

Table 13: RGP and OPEX Identification

Item	RGP Description
Codes and standards	ASME BPVC and other ASME standards have been used for the design of SSCs. These codes have been used extensively in the design of operating PWRs, such as the AP1000 [54].
OPEX	The SMR-300 is an advanced PWR with a design informed by decades of operating reactor experience and industry lessons-learned.
Redundancy, diversity, and segregation	Systems design incorporate appropriate levels of redundancy, diversity, and segregation.
Passive Safety Systems	The SMR-300 safety systems are passive and are driven by natural forces (e.g., gravity, conductive and convective heat transfer), with no reliance on pumps, external water, or offsite power.
Containment penetrations and isolation	The number of penetrations through the containment shall be kept to a minimum. Containment isolation valves are provided to maintain containment pressure boundary.
RCP	During Normal Operation, the SMR-300 design uses RCPs to provide forced flow through the core and generate the rated power. The SMR-300 does not rely on the RCPs for circulation during accident conditions. RCPs utilise a flywheel to provide coast-down flowrate during early stages of some DBAs.
RPV	ASME Boiler and Pressure Vessel Code, Section III, Class 1, thick-walled cylindrical pressure vessel.
PCM	The total PCM water inventory shall be sufficient to prevent fuel failure keep the core covered after a LOCA for at least 72 hours.
PCC	The components of the PCC are designed to the requirements of ASME Code, Section III, Division 1, Subsections NB and NC. The SDH has no containment isolation valves in the lines that penetrate the containment, taking credit for the closed system inside containment as one barrier and the closed system outside Containment as the second barrier for containment isolation [21].
MCH	The number and size of emergency air storage tanks provide the required air flow for 72 hours and limit the CO_2 concentration to the level governed by American Society of Heating, Refrigerating and Air-Conditioning Engineers (ASHRAE) Standard 62.1 [55].
MCH	The control room habitability analysis considers all materials expected to be used during routine plant operations from the list of materials in Table 1 of U.S.NRC Regulatory Guide 1.78 [56].

1.7.2.2 Evaluation of Risk and Demonstration Against Risk Targets

The numerical targets against which the demonstration of ALARP is considered can be found in Part A Chapter 2 [3].

RSC SSCs and ESFs, through the defined safety functions, will contribute to the demonstration of ALARP by comparison against the risk targets in two ways:

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

Evaluation of risk is not directly applicable to the RCS SSCs and ESFs. The safety classification of the RCS SSCs and ESFs will be associated with a Probability of Failure on

Demand (PFD) and Probability of Failure per Annum (PFA), which is then used to calculate the overall comparison against the risk targets as described above.

The evaluation of the normal operations and accident risks against Targets 1-9 is summarised in Part A Chapter 5 [15].

1.7.2.3 Options Considered to Reduce Risk

The process for the assessment of risk reduction options is presented in 'HPP-3295-0017-R1, GDA Design Management Procedure' [31].

Part A Chapter 5 'ALARP Summary' [15] considers the holistic risk-reduction process for the Generic SMR-300.

Design Challenges that have been raised and are applicable to the scope of this chapter are summarised below.

1.7.2.3.1 [REDACTED]

[REDACTED].

1.7.2.3.2 [REDACTED]

[REDACTED].

1.7.2.3.3 [REDACTED]

[REDACTED].

1.7.3 GDA Commitments

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

At Revision 1 there are no GDA commitments identified for Part B Chapter 1 Reactor Coolant System and Engineered Safety Features.

1.7.4 Conclusion

This chapter summarises the high-level design of the Reactor Coolant System and Engineered Safety Features. It identifies the claims, arguments and supporting evidence that will form the basis of the safety case for these systems throughout the lifecycle of SMR-300 to a maturity aligned to a PSR. Supporting engineering chapters provide further claims, arguments and evidence pertaining to these systems and these are referenced where relevant.

As the design and safety case matures, further evidence will be provided to substantiate these claims and arguments.

Differences between the reference US design and practices in the UK have been identified. The safety categorisation and classification will be confirmed based on the outcome of the UK DBAA. The application of UK regulatory expectations with regards to single failure criterion have been raised as potential risks against the suitability of UK deployment. Although still under initial investigations, processes have been implemented to manage the risk and ensure the design meets ONR expectations prior to deployment. Any shortfalls/risk will be addressed on a case-by-case basis with an appropriate level of management oversight depending on the risk to UK deployment and impact to design.

As outlined in the Technical Summary, not all claims are met to the expected maturity for PSR Revision 1. Where issues within the scope of this chapter have been identified (Sub Chapter 1.7.2.3), design challenge papers have been raised to manage the gap and ensure the correct level of oversight is maintained.

1.8 REFERENCES

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

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[12] Holtec Britain, “Holtec SMR GDA PSR PART B Chapter 14 Safety and Design Basis Accident Analysis,” Revision 1, July 2025.

[13] Holtec Britain, “HI-2240349, Holtec SMR GDA PSR PART B Chapter 18 Structural Integrity,” Revision 1, July 2025.

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1.9 LIST OF APPENDICES

Appendix A Classification of SSCs

Table 14: RCS and ESFs SSCs Classification [24]

SSC	RG 1.26 Quality Group	SMR Class	Seismic Category	Major Function
RCS	**(Note – RCS Preliminary UK Classification as per** [47] **– Class 1)**			
RPV	A	A	C-I	• RCPB
Piping and components in the RPV head vent line from the RPV up to and including the second RCPB Pressure Isolation valves	A	A	C-I	• RCPB
Piping and components in the RPV head vent line downstream of the second RCPB Pressure Isolation valves up to Passive Core Makeup Water Tank (PCMWT) A	C	C	C-I	• Other Functions (Makeup, Purification etc.)
Piping and components in the RPV Head Flange gasket leak off-line	D	D	C-II	• Other Functions (Makeup, Purification etc.)
Pressuriser, piping, and components in the PZR vent line from the PZR up to and including the second RCPB Pressure Isolation valve	A	A	C-I	• RCPB
Piping and components in the PZR vent line from the PZR downstream of the second RCPB Pressure Isolation valve	D	D	C-II	• Other Functions (Makeup, Purification etc.)
Steam Generator (tube side), Steam Generator bottom head and top portion of Steam Generator above tube sheet up to and including the PZR interface, and Reactor Coolant Pumps. All piping and components in the inlet lines from the hot legs and discharge lines to Coolant Pump Piping.	A	A	C-I	• RCPB
Piping and components in the ADS Stage 1 Valves line, from the PZR up to and including the second RCPB Pressure Isolation valves (ADS Stage 1 Valves)	A	A	C-I	• RCPB • Emergency core cooling*
Piping and components downstream of the second RCPB Pressure Isolation valves (ADS Stage 1 Valves) up to PCMWTs	C	C	C-I	• Pressure Relief/ Depressurisation
Piping and components in the ADS Stage 2 line, up to and including the second RCPB Isolation valves (ADS Stage 2 Valves)	A	A	C-I	• RCPB • Emergency core cooling* • Pressure Relief/ Depressurisation
Piping and components downstream of the second RCPB Pressure Isolation valves (ADS Stage 2 Valves)	C	C	C-I	• Pressure Relief/ Depressurisation
Piping and components in the Safety Valves line from the PZR up to and including the Safety Valves	A	A	C-I	• RCPB • Pressure Relief/ Depressurisation
Piping and components downstream of the Safety Valves up to PCMWTs	C	C	C-I	• Pressure Relief/ Depressurisation
ESFs				

SSC	RG 1.26 Quality Group	SMR Class	Seismic Category	Major Function
PCH	**(Note – PCH Preliminary UK Classification as per** [47] **– Class 1, see 1.4.1)**			
Annular Reservoir (AR)	C	C	C-I	• Post-accident Containment heat removal
Instrument lines penetrating the Containment Structure	B	B	C-I	• Containment Isolation
All other equipment, components, and piping Note: The recirculation piping inside the Containment Structure is to be segregated from safety-related equipment and components to prevent system interactions.	N/A	F	NS	• Other Functions (Makeup, Purification etc.)
MCH	**(Note – MCH Preliminary UK Classification as per** [47] **– Class 1)**			
BAP - Emergency Air Storage Tanks in the Tank Vault, Habitability Filter Banks in the Control Room Emergency Zone (CREZ), HVAC Equipment Room, Pressure Control Valves, piping and components in the BAP Fill connection piping and components including the normally closed fill connection valve	C	C	C-I	• Control room habitability
Dampers and the ductwork in the Normal Ventilation Subsystem (CRV) air supply line and return line used for isolation of the CREZ. Dampers and the ductwork in the CRV associated with smoke purge line used for isolation of the CREZ	C	C	C-I	• Control room habitability
All HVAC Equipment, ductwork and components with in the CREZ	D	D	C-II	• Control room habitability
CRV - Main Control Room Air Handling Units, Main Control Room Filtration Units, Main Control Room Smoke Purge Fan, Toilet/Kitchen Exhaust Fan, dampers, ductwork and components in the in the Auxiliary Building (outside CREZ)	N/A	F	NS	• Control room habitability
PDH	**(Note – PDH Preliminary UK Classification as per** [47] **– Class 1)**			
PDH loop - Piping and components in the PDH loop, including the inlet side (connected to RCS) of the PDH Heat Exchanger (PDH HX) and the outlet side, to the SGE. Vent Line in the PDH loop and the associated components in the Vent Line up to and including the second RCPB Isolation valve	A	A	C-I	• RCPB • Emergency core cooling*
Piping and components downstream of the second RCPB Isolation valve in the PDH loop in the Vent Line	C	C	C-I	• Other Functions (Makeup, Purification etc.)
Piping and components downstream of the orifice in the PDH loop Vent Line	D	E	NS	• Other Functions (Makeup, Purification etc.)
SDH	**(Note – SDH Preliminary UK Classification as per** [47] **– Class 1)**			
SDH HX, piping and components in the SDH both inside and outside Containment	B	B	C-I	• Emergency core cooling • Containment Isolation

SSC	RG 1.26 Quality Group	SMR Class	Seismic Category	Major Function
PCM	**(Note – PCM Preliminary UK Classification as per** [47] **– Class 1)**			
Piping and components in the injection line to RPV, downstream of the second RCPB Isolation valve, including the second RCPB Isolation valve	A	A	C-I	• RCPB
Accumulators and PCMWT Piping and components in the injection line to RPV from the Accumulators and PCMWT, up to and excluding the second RCPB Isolation valve. Piping and components connected to Passive Core Makeup Water Tanks up to and including the isolation valves. Piping and components, connected to Accumulators up to and including the isolation valves, except for the isolation valve in the CVC fill line to Accumulators	B	B	C-I	• Emergency core cooling
Piping and components in the CVC fill line to Accumulators, up to and including the RCPB isolation valves	A	A	C-I	• RCPB
Piping and components in the recirculation line between PCMWT and the SFP including the two recirculation valves	B	B	C-I	• Emergency core cooling

*Applicable to SSCs that provide or ensure emergency core cooling function

Appendix B PSR Part B Chapter 1 CAE Route Map

Table 15: PSR Part B Chapter 1 CAE Route Map

[REDACTED]

Appendix C High-Level, Safety and Non-Safety Functions for RCS and ESFs SSCs

Table 16: RCS SSC Schedule

[REDACTED]

Table 17: ESF SSCs Schedule

[REDACTED]