A Holtec International Company

Holtec Britain Ltd	HI-2240776
Sponsoring Company	**Document Reference**
1	23 September 2025
Revision No.	**Issue Date**
Report	Non-proprietary
Record Type	**Proprietary Classification**
ISO 9001	No
Quality Class	**Export Control Applicability**

Record Title:

PSR Part B Chapter 2 Reactor

Proprietary Classification

This record does not contain commercial or business sensitive information.

Export Control Status

Export Control restrictions do not apply to this record.

Revision Log

Revision	Description of Changes
0	First Issue to Regulators to support PSR v0
1	Second Issue to Regulators to support PSR v1

Table of Contents

List of Figures

List of Tables

2.1 INTRODUCTION

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

PSR Part B Chapter 2 Reactor presents the Claims, Arguments and Evidence (CAE) for the Reactor topic.

2.1.1 Purpose and Scope

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

This chapter of the PSR, Part B Chapter 2, 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 which are responsible for the design and development of relevant Systems, Structures, and Components (SSCs).

This chapter presents the Reactor system, directly supporting the claims made on the overall design and architecture of Reactor (including Fuel and Core) SSCs (Claim 2.2.12). Furthermore, these SSCs function to support the three critical safety functions (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 2.2.12: The objective of preventing damage to the fuel and core components is appropriately accounted for within their design and safety function.

Further discussion on how the Level 3 claim is broken down into Level 4 claims and how the Level 4 claims are met is provided in Sub-chapter 2.3.

Sub-chapter 2.2 provides an overview of the reactor SSCs and reactor operation for normal and fault conditions.

Sub-chapter 2.4 presents the codes and standards associated with the reactor topic.

Sub-chapter 2.5 provides an overview of the safety case, in relation to the reactor topic, while sub-chapter 2.6 and 2.7 define and assess the safe envelope for fuel and core.

Finally, sub-chapter 2.8 provides a technical summary of the chapter and a summary of the considerations against the ALARP principle, alongside any commitments that have arisen.

2.1.1.1 Physical Scope of the Reactor Topic

The physical scope of this chapter, Reactor system, is set out in Sub-chapter 2.2, which includes the following SSCs:

- Reactor Internal Structure (RIS)
- Fuel Assemblies
- Neutron Source Assemblies (NSAs)
- Rod Cluster Control Assemblies (RCCAs) (aka Control Rod Assembly (CRA))
- Control Rod Control System (CRC) [3]
    - Control Rod Drive System (CDS)
        - Control Rod Drive Mechanism (CRD)
    - Rod Position Indicator (RPI)
- Excore Instrumentation System (EIS) [4]
- Incore Instrumentation System (IIS) [5]

2.1.1.2 Operational Scope of the Reactor Topic

The operational scope of the Reactor topic covers:

- All aspects of in-reactor operation.
- Certain out-of-reactor operations, such as criteria and limits associated with the shipping, handling, and storage of fresh and spent fuel, RCCAs and NSAs.

2.1.1.3 Safety Assessment Scope of the Reactor Topic

The overall scope of safety analysis covered by the Reactor topic is largely based upon the physical and operational scope of the topic.

- Description of Reactor (overview provided at Step 2 GDA).
- Design Basis for Fuel and Core (advance development at Step 2 GDA).
- Fuel Rod performance (follows Step 2 GDA).
- Fuel Assembly mechanical performance (follows Step 2 GDA).
- RCCA thermal-mechanical performance (follows Step 2 GDA).
- Neutron Source Assemblies (follows Step 2 GDA).

- Nuclear Design (limited scope at Step 2 GDA).
- Thermal-hydraulic Design (limited scope at Step 2 GDA).
- Life-cycle aspects (follows Step 2 GDA).
- Reactor Internal Structure Design Substantiation (limited scope at Step 2 GDA).

2.1.1.4 Aspects which are Out of Scope of PSR Part B Chapter 2

The following items are out of scope of the Reactor topic at PSR Revision 1:

- Manufacturing of fuel and core items, such as materials specifications, manufacturing drawings, description of quality control processes, manufacturing inspection programme etc.
- Dry Storage: Consideration has been given to the types of failure modes which fuel and RCCAs could potentially experience during dry storage operations, although no detailed assessment is presented in PSR Revision 1. The topic of dry storage is described in Part B Chapter 24 Fuel Transport and Storage [6].
- Derivation of operating procedures requiring an input from the Reactor topic area.
- Derivation of operational support data having an origin within the Reactor topic area (e.g. low power physics testing).

A master list of definitions and abbreviations relevant to all PSR Chapters can be found in Part A Chapter 2 General Design Aspects and Site Characteristics [7].

2.1.2 Assumptions

There are no assumptions for this chapter.

2.1.3 Interfaces with other SSEC Chapters

This chapter interfaces with many other chapters within the PSR. The following highlights key interfaces.

For specific aspects of SSCs relating to control and instrumentation, Part B Chapter 4 Control and Instrumentation [8] provides the substantiation. Similarly, SSC aspects relating to electrical engineering are covered in Part B Chapter 6 Electrical Engineering [9]; mechanical engineering is covered in Part B Chapter 19 Mechanical Engineering [10]; and reactor chemistry is covered in Part B Chapter 23 [11]. For SSCs which are designated as higher reliability, Part B Chapter 18 Structural Integrity [12] describes the specific substantiation requirements.

From a safety analysis perspective, this chapter interfaces with Part B Chapter 14 Design Basis Analysis (Fault Studies) [13] and Part B Chapter 15 Beyond Design Basis Analysis, Severe Accident Analysis, and Emergency Preparedness [14] for design basis and beyond design basis challenges to the fuel system and reactor integrity. Part B Chapter 2 defines the design basis limits for the fuel system and reactor to which the analysis undertaken in these chapters can be compared and assessed. Part B Chapter 2 interfaces with Part B Chapter 16 Probabilistic Safety Assessment [15] by providing input data for analyses.

The overall generic design aspects and site characteristics, within which the SSCs from this chapter sit, are reported in Part A Chapter 2.

The scope of Part B Chapter 2 directly interfaces with many of the systems described in Part B Chapter 1 Reactor Coolant System (RCS) and Engineered Safety Features (ESFs) [16] and Part B Chapter 5 Reactor Supporting Facilities [17], specifically the Reactor Pressure Vessel (RPV).

While Part B Chapter 2 addresses certain fuel criteria associated with fuel handling and storage, the specific details of this topic area are described in Part B Chapter 24 Fuel Transport and Storage [6].

Part A Chapter 5 Summary of ALARP [18] 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 [19] and GDA Scope [20], 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 2 will help to support the Approach and Application of BAT Demonstration. Part B Chapter 11 Environmental Protection [21], serves as the interface between PSR and Preliminary Environmental Report (PER), and mainly summarises 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.

2.2 REACTOR OVERVIEW

This sub-chapter provides an overview of the reactor SSCs and reactor operation, within the scope of this chapter. Further detailed description of the reactor SSCs discussed within this sub-chapter is presented in the Overview of SMR-300 Fuel Design and Core Components document [22], including details of the SSC classifications.

The SMR-300 reactor incorporates the Framatome GAIA fuel assembly. This fuel assembly is new to the UK but has extensive Operating Experience (OPEX), outlined in the SMR-300 GDA Thermal and Mechanical Design Basis report [23], with no novel operational limits or criteria (see SMR-300 GDA Fuel Design Criteria and Limits [24]).

Asymmetric cold leg inlet nozzles bring reactor coolant into the RPV. Coolant flow maldistribution is addressed with the use of a Flow Distribution Device (FDD). Further details of the FDD design, optioneering and development are presented in this chapter and in the SMR-300 GDA Thermal and Mechanical Design Basis report [23]. Future work relating to the FDD is described in sub-chapter 2.8.3.

The nuclear design of the reactor is supported by the SMR-300 GDA Nuclear Design Basis Report [25] and the SMR-300 GDA Fuel Design Criteria and Limits [24].

2.2.1 Reactor Parameters

Table 1 displays key reactor parameters for the SMR-300 reactor design.

Table 1: Reactor Parameters

Parameter	Value / description	Notes or units
Nuclear Steam Supply System (NSSS) Thermal Power	1050	(MWt) [26]
Fuel Assembly model	Framatome GAIA 17 x 17	Manufactured by Framatome.
Fuel type	Uranium oxide	
Fresh fuel enrichment	5.0% U-235	Maximum enrichment.
Cycle length	[REDACTED]	Refuelling outages every [REDACTED].
Refuelling outage	[REDACTED]	Nominal
Design life of plant	80 years	N/A
Neutron energy spectrum	Thermal	N/A
Main reactivity control	Soluble boron and RCCAs	N/A
Core configuration	69 FAs with 29 RCCAs	N/A

The SMR-300 is designed to have a nominal [REDACTED] fuel cycle length, with core reloads of a duration of [REDACTED] days between each cycle. During a refuelling outage, the fuel is offloaded from the reactor and placed into the spent fuel pool. The reactor is then reloaded with a combination of fresh and irradiated fuel for the new cycle. Following Cycle 1 the intention is for the fuel loading pattern to change from one cycle to the next until an equilibrium fuel loading scheme is reached. A detailed description of the fuel management scheme is presented in the Nuclear Design Basis Report [25].

2.2.2 Reactor Internal Structure

The SMR-300 RPV is constructed from heavy steel forgings. The RPV comprises a cylindrical middle section which is closed off at the bottom by a welded hemispherical lower section and is closed off at the top by a removable hemispherical upper section, which is flanged and bolted.

The principal RPV internal structures are the lower core internal assembly and the upper core internal assembly. The lower internals consist of the core barrel flange, core barrel cylinder and lower fuel support plate, heavy reflector, irradiation specimen baskets, radial key inserts and FDD. The FDD is located below the lower support plate. The upper internals consist of the upper support assembly (including the flange, shell, and upper support plate), the upper core plate, control rod guide tubes and cards, and support columns. Details of the upper internal assembly are still to be finalised. Figure 1 displays the reactor internal components, excluding the upper internals.

[REDACTED]

Figure 1: Reactor Internal Components

Reactor coolant flows into the RPV through two cold leg inlet nozzles separated azimuthally by [REDACTED] degrees (not shown in Figure 1). The inlet flow is mainly directed down an annulus between the RPV inner wall and the core barrel. The reactor coolant then passes through the FDD and up through the lower support plate and into the core. Reactor coolant passing through the core is heated by the fuel before exiting the core, flowing upwards through the upper core plate and into the volume between the support columns and the upper support assembly. The reactor coolant main flow then passes radially through core barrel ports and into two RPV hot leg exit nozzles which are separated azimuthally by 180°.

2.2.3 Fuel Assemblies

The SMR-300 core comprises 69 GAIA 17 x 17 Pressurised Water Reactor (PWR) fuel assemblies, each containing 264 fuel rods, 24 guide tubes and a single central instrumentation tube, as seen in Figure 2. Each of the fuel rods comprises a stack of fuel pellets contained within a tube of M5® clad. The tube is pressurised with helium gas and is sealed at each end. Each fuel pellet comprises UO_2 or UO_2-Gd_2O_3 enriched with U-235 up to a maximum of 5 wt%.

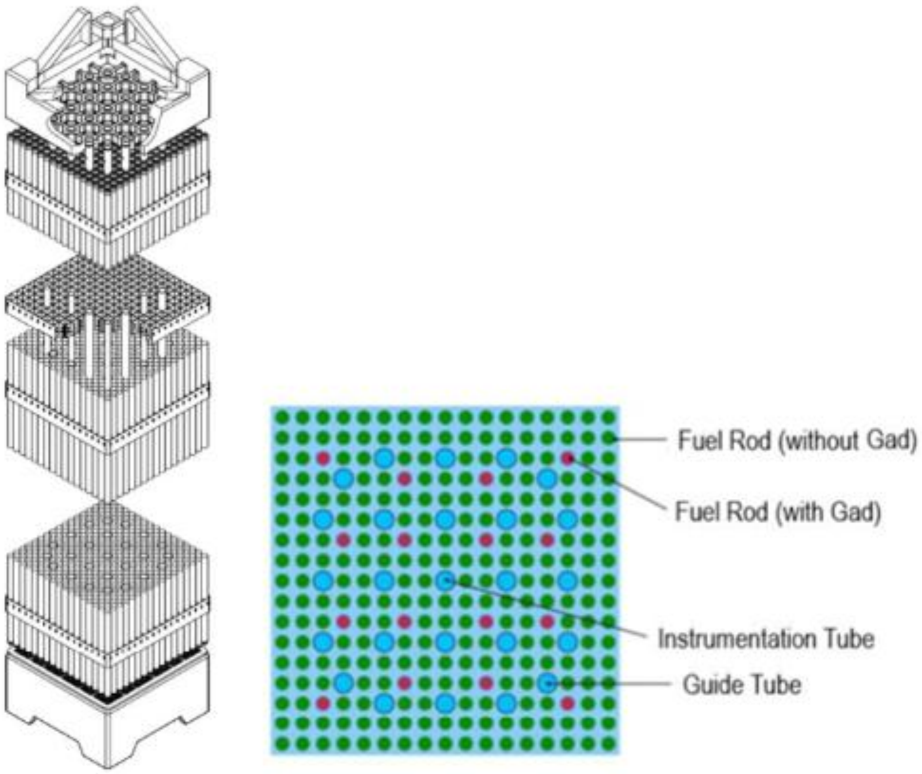

Figure 2: Fuel Assembly Overview

The fuel rods are maintained in a regular array by the fuel assembly skeleton, comprising the top nozzle, guide tubes, spacer grids and bottom nozzle.

2.2.4 Neutron Source Assemblies

NSAs are static components inserted into certain fuel assemblies that are not located under RCCAs. Two types of neutron source can be used within the SMR-300 core: primary neutron sources and secondary neutron sources.

Primary neutron sources provide a detectable neutron flux at the source range detectors during initial core loading. This allows for monitoring neutron count rate trends during fuel loading and the approach to criticality. These trends help operators infer changes in reactivity and verify that the core is responding as expected. Secondary sources serve the same function in subsequent cycles but must be activated prior to use.

[REDACTED]

2.2.5 Rod Cluster Control Assemblies

The SMR-300 core comprises 29 RCCAs (HARMONI type) which are used for reactivity control and shutdown. Each RCCA has [REDACTED]. The RCCA rods are arranged spatially so that they can enter the fuel assembly guide tubes upon insertion and their bottom tips are always engaged with the guide tubes for this purpose. Each RCCA rod contains [REDACTED] and a top plenum spring which provides a downward force on the absorber stack.

[REDACTED]

[REDACTED]

Figure 3: Fuel Assembly and RCCA

There are 29 CRDs, each comprising a drive shaft which connects to a single RCCA at its hub assembly. Each drive shaft penetrates the hemispherical upper section of the RPV where it is enclosed within a pressure retaining enclosure that forms an extension of the primary circuit pressure boundary.

2.2.6 Control Rod Control System

The CRC comprises the CDS, CRD, and the RPI. The CRC controls the motion of the RCCA in response to the demand signal from reactor control subsystems, Plant Control System (PCS) and Plant Safety System (PSS), to control reactivity in the reactor core during both normal operating and accident conditions [3]. The CRC also includes the RPI cabinets and cables, and CRD power cables.

2.2.6.1 Control Rod Drive System

The CDS supports the RCCA by latching, holding, and manoeuvring the RCCA during reactor start-up, power operation, and shutdown in response to signals from the CRC, and in releasing the RCCA during a reactor trip.

2.2.6.1.1 Control Rod Drive Mechanism

CRDs provide positioning for normal insertion and withdrawal of the control rods and rapid control rod insertion for abnormal operating conditions (e.g., reactor trip). There is a single CRD for each RCCA. The CRD includes the control rod drive shaft, which extends to the coupling interface with the RCCA inside the RPV.

The CRD pressure housing is attached to the RPV head and is part of the Reactor Coolant Pressure Boundary (RCPB), maintaining the reactor pressure in all modes of operation. More information about the RCPB is presented in Part B Chapter 1 [16].

The CRD is an electromagnetic jacking system which enables the RCCAs to be moved upwards and downwards in small steps. The movable and stationary gripper latches engage the drive shaft when respective latch coils are energised. The movable grippers are used to raise or lower the drive shaft when the lift coil is energised or de-energised. Stationary grippers are used to hold the drive shaft in position whenever rod motion is not required.

The CRD mechanism is designed to require a continuous electrical current in order to function. Loss of current causes the RCCAs to drop into the core.

Preliminary UK classification work, described in Part B Chapter 14 [13], has been performed and determined the CRD as a Class 1 component. Associated codes and standards used in the design of the CRD, based on US classification and presented in sub-chapter 2.4, are still applicable and considered RGP.

2.2.6.2 Rod Position Indicator

The RPI system senses the elevation of each RCCA in the core and transmits the position signal to the data cabinet. The data cabinet receives the position signal and converts it to a digital signal to transmit the position signals to the PCS.

2.2.7 Instrumentation

The EIS and IIS functions are described in this sub-chapter. Further description is provided in the Overview of SMR-300 Fuel Design and Core Components document [22], and Instrumentation and control aspects are described and substantiated in Part B Chapter 4 Control and Instrumentation [8].

2.2.7.1 Excore Instrumentation System

[REDACTED]

2.2.7.2 Incore Instrumentation System

[REDACTED]

2.2.8 Reactor Operation

2.2.8.1 Normal Reactor Operation

Table 2 shows the SMR-300 modes of operation and associated parameters.

Table 2: SMR-300 Operating Modes

Mode	State	K_{eff}	Rated Power (%)	RCS T_{avg} (°F)	Electric Power Research Institute (EPRI) Classification
1	Power Operation	≥ 0.99	> 5	[REDACTED]	Power Operation
2	Start-up	≥ 0.99	≤ 5	[REDACTED]	
3	Hot Standby	< 0.99	-	[REDACTED]	Start-up
4	Safe Shutdown	< 0.99	-	[REDACTED]	
5	Cold Shutdown	< 0.99	-	[REDACTED]	Cold Shutdown
6	Refuelling	•	-	-	
•	Core Empty	•	•	•	•

Power operation covers the power range of 5% to 100% of full power and is the intended long-term mode of operation during which the reactor is critical. Operation between 0% and 5% of full power is regarded as start-up, for which the reactor is also critical. During start-up and power operation, reactivity is controlled by both soluble boron and the RCCAs, and the reactor coolant temperature and pressure are at nominal operating values for these modes.

Hot standby is the state in which the reactor is held sub-critical by having all of the RCCAs fully inserted into the core and the boron concentration in the reactor is sufficiently high to maintain an adequate shutdown margin.

Safe shutdown is also a sub-critical state in which all RCCAs are fully inserted, and boron concentration is sufficiently high to maintain an adequate shutdown margin. However, the reactor coolant temperature and pressure are lower than for hot standby.

Cold shutdown is the state of operation for which the reactor is sub-critical, and the coolant temperature and pressure are each very low. Cold shutdown is a sub-critical state in which all of the RCCAs are fully inserted into the core and the boron concentration is sufficiently high to maintain an adequate shutdown margin.

Refuelling is the state of operation for which the reactor vessel head, reactor upper internals and CRDs are removed from the reactor. Prior to removal of the head, the reactor internals and the reactor cavity is flooded with a higher concentration of borated water and the CRDs are disengaged from the RCCAs so that they remain within the fuel assemblies.

The reactor is de-fuelled and made empty by transferring fuel, RCCAs and neutron sources to the spent fuel pool.

A detailed overview of the SMR-300 plant operations is provided in the SMR-300 Plant Operation Overview document [27] and Part B Chapter 9 Conduct of Operations [28].

2.2.8.2 Reactor Trip and Shutdown

In the case of a fault or hazard event occurring during reactor operation, it is necessary for the reactor to be taken to a safe, stable, shutdown state and to be maintained in that state for as long as required. Any short-term consequences of the fault or hazard event should be suitably limited.

A safe and stable shutdown state is one in which reactor conditions are steady or only varying slowly, such that there is control of reactivity, sufficient removal of decay heat and control of release of activity.

The primary means by which the SMR-300 achieves these emergency shutdown objectives is by allowing the RCCAs (and drive) to drop under the force of gravity, following a reactor trip, so that the RCCAs fully insert into the core. Control of reactor trip signals is described in detail in the Part B Chapter 4 Control and Instrumentation Systems [8]. A separate and diverse means of shutdown is provided via the injection of borated water to the core, this is described in Part B Chapter 1 RCS and ESFs [16].

Longer term control of reactivity is ensured by adjusting the boron concentration within the reactor coolant. Removal of decay heat is managed by the primary or secondary decay heat removal systems.

Ongoing work is being conducted to confirm the robustness of the SMR-300 design to ensure diverse and independent means of shutdown in all plant states, specifically in regard to the secondary means of shutdown. Details of this work are described in Part B Chapter 1 RCS and ESFs [16] under the sub-chapter for Design Challenge 4.

2.3 REACTOR CLAIMS, ARGUMENTS AND EVIDENCE

This chapter presents the Reactor aspects for the generic SMR-300 and therefore directly supports Claims 2.2.1, 2.2.2, 2.2.3, and 2.2.12.

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 2) and within Part B Chapter 1 RCS and ESFs.

Claim 2.2.1 is associated with independent means of reactivity control and the ability to shutdown the core and guarantee its safe long-term shutdown, for all modes of operation. 2.2.1 is decomposed into Claim 2.2.1.1, relating to reactivity control via borated water, described in Part B Chapter 1, and Claim 2.2.1.2, relating to reactivity control via control rods, described in this chapter.

Table 3 shows the breakdown of Claim 2.2.1 and identifies in which chapter of this PSR these claims are demonstrated for PSR v1.

Table 3: Claim 2.2.1 breakdown

Claim No.	Claim	Chapter Sub-chapter
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.	Part B Chapter 1 RCS and ESFs
2.2.1.2	Control rods provide appropriate reactivity control during normal reactor operation and the means for reactor shutdown during a trip.	Sub-chapter 2.7.3 Demonstration of Reactivity Control

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 2) and within Part B Chapter 1 RCS and ESFs.

Claim 2.2.2 asserts that, although not explicitly stated, in all relevant plant modes heat is removed from the reactor core and spent fuel pool. Claim 2.2.2 is decomposed into four subclaims. 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, as described in Part B Chapter 1. Claim 2.2.2.4 asserts that the core design ensures that it is always maintained in a coolable geometry, described in this chapter.

Table 4 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 4: Claim 2.2.2 breakdown

Claim No.	Claim	Chapter / sub-chapter
2.2.2.1	Core decay heat removal is ensured following credible initiating events in all plant states.	Part B Chapter 1 RCS and ESFs
2.2.2.2	Sufficient coolant inventory is maintained following credible initiating events in all plant states.	Part B Chapter 1 RCS and ESFs
2.2.2.3	Spent fuel heat removal is ensured following credible initiating events in all plant states.	Part B Chapter 1 RCS and ESFs
2.2.2.4	The core is always maintained in a coolable geometry.	Sub-chapter 2.7.4 Demonstration of Coolable Geometry

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), within Part B Chapter 1 RCS and ESFs and within 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 RCPB, and components which comprise the RCPB, as a barrier to radiation release, described within Part B Chapter 1 and Part B Chapter 5. Claims 2.2.3.2 and 2.2.3.5 declare the Containment Structure (CS), and components which comprise the CS, as a barrier to radiation release, described within Part B Chapter 1 and Part B Chapter 5. Claim 2.2.3.3 asserts the ability of the Main Control Room Habitability System (MCH) to ensure safe habitability of the main control room following credible initiating events, described in Part B Chapter 1. Claim 2.2.3.6 declares the fuel rod cladding as a barrier to radiation release, described in this chapter.

Table 5 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 5: Claim 2.2.3 breakdown

Claim No.	Claim	Chapter / sub-chapter
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.	Part B Chapter 1 RCS and ESFs
2.2.3.2	The RCS and ESF SSCs ensure the integrity of the Containment Structure following credible initiating events in all plant states.	Part B Chapter 1 RCS and ESFs
2.2.3.3	Habitability of the main control room is ensured following credible initiating events in all plant states.	Part B Chapter 1 RCS and ESFs
2.2.3.4	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

Claim No.	Claim	Chapter / sub-chapter
2.2.3.6	The fuel rod clad integrity is maintained during normal operation and AOOs.	Sub-chapter 2.7.5 Demonstration of Confinement

Claim 2.2.12: The objective of preventing damage to the fuel and core components is appropriately accounted for within their design and safety function.

Claim 2.2.12 has been further decomposed within this chapter to capture those aspects of the Reactor topic which support this claim. Claim 2.2.12 has been decomposed into two subclaims, 2.2.12.1 and 2.2.12.2, which assert that a safe envelope of operation for fuel and core components has been established and that operation within the safe envelope is demonstrated for all modes of operation. In each of these sub-claims, reference to the safe envelope is made via a set of requirements and associated criteria which define the envelope.

Table 6 shows the breakdown of Claim 2.2.12 and identifies in which chapter of this PSR these claims are demonstrated to be met to a maturity appropriate for PSR v1.

Table 6: Claim 2.2.12 breakdown

Claim No.	Claim	Chapter / sub-chapter
2.2.12.1	Safety case requirements relating to fuel and core components are established which define the safe envelope of operation.	Sub-chapter 2.6 Defining the Safe Envelope of Operation for Fuel and Core
2.2.12.2	The safety case requirements for the fuel and core components are demonstrated to be satisfied.	Sub-chapter 2.7 Assessing the Safe Envelope of Operation for Fuel and Core

Appendix A provides a full Claims, Arguments and Evidence mapping for Part B Chapter 2, which includes any lower-level claims, arguments and evidence needed to support the claims in the table above. This includes identification of evidence available at PSR v1 and aspects for future development of evidence to support these claims beyond PSR v1.

2.4 REACTOR CODES AND STANDARDS / METHODOLOGIES

The process for identifying and classifying SSCs is presented in Part B Chapter 14 [13]. The applied codes and standards for the Reactor topic have been selected on the basis of their longstanding application to PWR technology either within the United States (US), the UK, or both. The applied codes and standards for fuel, neutron sources, RCCAs and other reactor internal components and structures, are intended to provide confidence in the safety of the design of the SMR-300.

This sub-chapter provides further information on the codes and standards applicable to the Reactor topic. Furthermore, a summary of the reactor topic codes and standards are presented in the Overview of Core Components document [22].

2.4.1 Fuel Assemblies, RCCAs and NSAs

The fuel assembly and control rod components are designed to conform to the acceptance criteria in NUREG-0800 Section 4.2 [29], with reference to other codes. More generally, the following codes and standards are applied to the design and manufacture of fuel assemblies, RCCAs and neutron source assemblies:

- Fuel Pellets: fuel vendor internal codes and standards (American Society for Testing and Materials (ASTM) standards), approved by the United States Nuclear Regulatory Commission (USNRC).
- Fuel Rod Cladding: fuel vendor internal codes and standards (ASTM standards), approved by the USNRC. USNRC 10 Code of Federal Regulations (CFR) 50.46.
- Fuel Assembly Skeleton: American Society of Mechanical Engineers (ASME) III various.
- RCCAs and Neutron Source Assemblies: ASME III various, ASTM standards.

Specific fuel vendor codes and standards are presented in the fuel vendor's Topical Reports, reviewed in the Applicability Report for the SMR-300 [30]. This report summarises the outcome of reviews of certain fuel vendor Topical Reports for GAIA and explains why the methodologies covered by those Topical Reports are considered to be applicable to the SMR-300.

2.4.2 Reactor Internal Structures

The following codes and standards are applicable to the reactor internal structure [22]:

- Materials: ASME Boiler and Pressure Vessel Code (BPVC) Section II, covering aspects such as resistance to inter-granular corrosion, carbon content, chromium content, toughness, weldability, testing etc.
- Class 1, 2, 3 components: ASME BPVC Section III Subsections NB, NC, ND.
- RPV internal support structures: ASME BPVC Section III Subsection NG.
- Welding: ASME BPVC Section IX.
- Fabrication: ASME Section III, covering aspects such as markings, cuttings, forgings, surface treatments, cleanliness, heat treatments etc.

The majority of the reactor internal components are to be manufactured in accordance with the ASME BPVC Section III (ASME III), as set out under US NRC CFR 10 Part 50.55a. The use of ASME III within the UK has precedent at Sizewell B. It is also noted that the UK European Pressurised Reactor (EPR) applies the RCC-M mechanical design code which is largely based upon ASME III. Hence application of ASME III is considered to represent relevant good practice.

The ASME code also specifies service levels (i.e. criteria) which are applicable to conditions of operation. Application of those service levels to the SMR-300 will be finalised following Step 2 of GDA.

Applicable sections of the ASME code to the SMR-300 reactor are summarised in Table 7.

Table 7: Applicable ASME Sections

Component / Process	Applicable ASME Section
Reactor Components	Section III
Materials	Section II
Examination methods	Section V
Welding	Section IX
Fabrication	Various parts of Section III
General requirements	NCA
Class 1 components	NB
Class 2 components	NC
Class 3 components	ND
Reactor internals	NG
Supports	NF
Technical appendices	Appendices

2.4.3 Appropriateness of the Selected Codes and Standards

Relevant codes and standards are selected based on the categorisation and classification of the SSCs used within the reactor. The SSC classification methodology is described within the SSC Classification Standard [31], and conforms with the USNRC Regulatory Guide (RG) 1.26, Revision 6 [32] and related guidance documents. This methodology is described in Part A Chapter 2 [7], which describes RG 1.26 Quality Groups A through D, and the link to the SSC Classification Standard “SMR Class”.

Justification and appropriateness of codes and standards selection is described in Part B Chapter 18 Structural Integrity [12], Part B Chapter 19 Mechanical Engineering [10], and the Mechanical Codes and Standards Report [33]. Further details of Electrical, and Instrumentation and Control (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 [8].

Design requirements defined for fuel, RCCAs and neutron source assemblies identify relevant codes where applicable as criteria to be satisfied by the safety assessments, as described in the Fuel Design Criteria and Limits document [24].

Holtec acknowledges the existence of differences in the approach to safety categorisation and classification between the NRC regulatory guides and the UK expectations. Further detail on the strategy for the categorisation and classification process is described in Part A Chapter 2 [7].

Codes and standards selected are considered to represent good practice across a wide range of international regimes and are judged to be appropriate for use for the SMR-300

2.4.4 Adaptations

No adaptations to the codes and standards relevant to the Reactor topic are proposed at Step 2 of GDA. This is because the use of codes and standards within the safety case requirements described in sub-chapter 2.6 are considered to be suitably conservative and generally optimised for the SMR-300. This position will be reviewed following Step 2 of GDA and any adaptations to codes or standards considered necessary will be addressed within a future version of the safety case.

2.4.5 Exemptions

No exemptions to the codes and standards considered to be applicable to the Reactor topic are proposed at Step 2 of GDA. Such exemptions could potentially arise from conflicts between the requirements of a code or standard and the ALARP principle. This position will be reviewed following Step 2 of GDA and any required exemptions to codes or standards will be addressed within a future version of the safety case.

2.4.6 IAEA Standards

The following International Atomic Energy Agency (IAEA) standards are considered to be pertinent to the Reactor topic at Step 2 of GDA:

- IAEA SSR-2/1: Safety of Nuclear Power Plants: Design, especially in regard to the reactor core requirements:
    - Requirement 43: Performance of fuel elements and assemblies
    - Requirement 44: Structural capability of the reactor core
    - Requirement 45: Control of the reactor core
    - Requirement 46: Reactor shutdown

Requirements 43 to 46 of SSR-2/1 are addressed within Part B Chapter 2 via demonstration showing the Reactor topic claims are satisfied.

- IAEA SSG-52: Design of the Reactor Core for Nuclear Power Plants [34].

A review of the design of the SMR-300 reactor core against SSG-52 has been performed, with no shortfalls, the results of which are presented in [35].

2.4.7 Fuel Manufacturing Compliance with Codes and Standards

The primary standards and specifications applied to the manufacture of GAIA fuel assemblies are specific to the fuel vendor. These provide for the inspection standards and controls used during manufacture. More generally, the fuel vendor’s quality programme has been reviewed and approved by the USNRC and has been accepted by each of fuel vendor’s US customers.

Appropriate testing of the GAIA fuel pellets for surface defects, chemical composition and impurity checks and examinations for the microstructure are performed during manufacture. Measurements of fuel pellet enrichments are also performed to ensure manufactured fuel assembly enrichments conform to the intended cycle-specific core design.

The fuel rod cladding tubes are inspected for external and internal defects by approved non-destructive methods. Gamma scans are typically used to verify the integrity of the fuel internal components as well as the absence of pellet gaps. Welds in the fuel rod and components are tested with both destructive and non-destructive methods. Fuel assemblies are inspected for length, bow, twist, dimensional envelope. Dimensional inspections are performed using ultrasonic measurements.

Details of fuel manufacturing surveillance and inspections will be provided by the fuel vendor following Step 2 of GDA, closer in time to when fuel assemblies will be manufactured. Furthermore, it is the current intention of Holtec to establish an independent third-party review process of the key manufacturing steps performed by the fuel vendor, details of which will be developed following Step 2 of GDA.

2.4.8 Testing of Fuel and RCCAs

All component materials are procured from approved suppliers using approved material specifications, which may include industry-approved standards (such as ASME and ASTM materials specifications) or from approved Framatome internal specifications. All safety-related materials require a certified material test report and are independently reviewed for conformance to the specification requirements.

In the case of the GAIA fuel assembly, out of pile integral and sub-component testing was performed prior to the loading of GAIA fuel assemblies in PWR cores. These tests assessed fuel assembly response to loads such as compression, bending, vibration etc and were of typical of tests that are usually performed by the fuel vendor before bringing a new fuel assembly type to market.

The RCCAs and CRDs to be used in the SMR-300 are of a standard Framatome design that are used in other PWRs. These components have previously undergone extensive testing and design substantiation, and their use in PWR cores is supported by extensive OPEX. Nevertheless, a bespoke test rig that replicates SMR-300 operation conditions, e.g. flow conditions, will be constructed by Framatome for the purpose of testing RCCA performance and drop times.

Results from tests of individual fuel assemblies, RCCAs and their sub-components, performed to date, have been used to validate certain software tools or hand calculations that are used for safety analysis by the fuel vendor, such as modelling using the ANSYS code. The results of future testing will be accounted for within the modelling of fuel and core, where necessary, following Step 2 of GDA.

2.5 SAFETY CASE OVERVIEW

2.5.1 Safety Case Objectives

The safety case for the SMR-300 is predicated upon demonstrating application of “defence in depth”, the elements of which are set out in IAEA’s INSAG-10 document [36]. In particular, INSAG-10 makes reference to the following barriers to radiological release:

- Fuel matrix
- Fuel rod cladding
- Reactor coolant circuit
- Reactor containment structure

For the SMR-300, [REDACTED], although its presence and ability to retain a certain fraction of fission products is taken into account within the safety assessments of the fuel, as required. Consequently, the contribution of the Reactor topic to the overall radiological aspect of the safety case, and therefore the ALARP demonstration, involves substantiating the fuel clad integrity. The reactor coolant circuit barrier is described in Part B Chapter 1 [16]. The reactor containment structure barrier is presented in various PSR chapters, an overview of the containment system is provided in the following document [37].

The substantiation of the fuel clad is in turn predicated upon demonstrating that the SMR-300 core can be operated safely, such that it satisfies the following safety functions: control of reactivity, management of heat transfer, and confinement of activity. Failing to satisfy these safety functions has the potential to lead to the loss of fuel clad integrity, e.g. via excessive power peaking, melting and phenomenological failure modes that are intrinsic to the fuel clad under certain circumstances, respectively.

2.5.2 Safety Case Strategy

The safety case relates primarily to demonstrating the safe operation of the fuel, RCCAs and neutron sources. Demonstrating the safe operation of other reactor internal components is made with reference to other PSR chapters (see Sub-chapter 2.1.3).

For this reason, the safety case strategy, for the reactor topic, involves the following main steps:

1. Identifying potential failure and degradation mechanisms which could jeopardise the integrity or correct performance of fuel assemblies, RCCAs and neutron sources.
2. Listing those potential failure and degradation mechanisms as a set of safety case requirements and establishing acceptance criteria for them. The safety case requirements and their criteria define the “safe envelope” of operation for the fuel, RCCAs and NSAs.
3. Describing how the safety case requirements relate to the overarching safety objectives for the SMR-300, i.e. reactivity control, heat management and confinement.
4. Demonstrating that each acceptance criterion is satisfied for the plant conditions to which it is applicable, including fault and hazard conditions and throughout the life-cycle of the fuel, RCCAs and neutron sources, as appropriate.
5. Identifying relevant codes and standards applicable to the SMR-300 reactor and outlining how the SMR-300 reactor core SSCs satisfy those codes and standards.
6. Reviewing OPEX for the fuel assembly.

In doing so, it is shown how:

- The core design, RCCAs and soluble boron contribute to ensuring reactivity control.
- The fuel assemblies and other reactor internal structures contribute to ensuring regular core geometry for the effective removal of heat from the fuel to the reactor coolant.
- The fuel clad and other core components contribute to ensuring no release of activity from within the fuel rods.

2.5.2.1 Generic and Cycle-specific Aspects of the Safety Case

The SMR-300 reactor core design includes a nominal Cycle 1 core design, a series of transition cycles and a [REDACTED] equilibrium cycle. The analysis for these designs is considered to be "generic safety analysis" which underpins the generic aspects of the safety case.

The majority of safety assessments performed within the Reactor topic area are 'generic' in the sense that results are not sensitive to changes of core design (fuel loading pattern) from one cycle to the next. Examples of this kind of assessment include fuel assembly mechanical performance, RCCA mechanical performance, neutron source assembly mechanical performance and certain fuel rod performance assessments.

However, it is generally the case that a subset of the generic analysis has to be either confirmed as remaining valid for a reloaded core or updated every time a new core is loaded. Assessments falling into the cycle-specific category include a subset of nuclear design analysis, a subset of fuel rod analysis and life-cycle assessments (e.g. fuel operability for reload).

No attempt has been made to determine the scope of cycle-specific assessments at this iteration of the Part B Chapter 2. Instead, assessments relate to the use of GAIA fuel assemblies within a generic sequence of fuel loading patterns that are intended to show the general viability and safety of the SMR-300 reactor when loaded with that fuel design.

2.5.2.2 Operational Support Requirements

Operational support data which originates within the Reactor topic discipline area is out of scope at this iteration of Part B Chapter 2. Typically, data is generated within the Nuclear Design technical area to support operational aspects such as reactor start-up operations, low power physics testing and routine core flux mapping. These aspects of the Reactor topic area will be developed following Step 2 of GDA.

2.5.2.3 Development of Operating Procedures

Part B Chapter 9 Conduct of Operations [28] describes operational aspects of the SMR-300, including the high-level Outage Strategy for SMR-300 [38] for a refuelling outage. Reactor topic input to operational requirements, such as inspection, maintenance, examination and testing, is out of scope at this iteration of Part B Chapter 2. It is expected that the Reactor topic input to operational procedures for the SMR-300 will include:

- Use of fuel and core criteria for setting safe operational boundaries for the reactor and other plant items, e.g. in relation to reactivity control, heat management and confinement of activity.

- Generation and use of physics data to support routine reactor operations. Low power physics testing and flux mapping to validate physics calculations.
- Specifying aspects to be assessed when confirming the operability of fuel assemblies, RCCAs and neutron sources, such as:
    - Visual inspection of fuel assemblies.
    - Failed fuel detection.
    - RCCA rod wear.
    - RCCA-rod drop tests.

These aspects of the Reactor topic area will be developed following Step 2 of GDA.

2.5.2.4 Use of Operating Experience

OPEX has been obtained via a search of open-source literature and requests for information from the fuel and RCCA vendor. The assessment of OPEX has primarily focused upon the performance GAIA fuel assemblies within PWRs and the performance of M5® fuel cladding. OPEX has also been considered in the validation of core design codes, during the design of core components (specifically the FDD and Heavy Reflector), and assessing core flow stability.

2.5.3 Safety Case Structure

Figure 4 shows the safety case structure for the Reactor topic. The key supporting references to Part B Chapter 2 are:

- Reactor Description Overview (Overview of Holtec SMR-300 Fuel Design and Core Components) [22].
- Criteria and Limits Bases (Holtec SMR-300 GDA Fuel Design Criteria and Limits) [24].
- Nuclear Design (ND) Basis Report [25].
- Thermal-hydraulics and Mechanical Design (TH/MD) Basis Report [23].

The structure of the Reactor topic area reflects its current stage of development, in so far as it relates to generic aspects of the safety case. In due course, further references will be added to the safety case structure to reflect the cycle-specific aspects. Further details of the overall safety case structure is presented in Part A Chapter 1 [1].

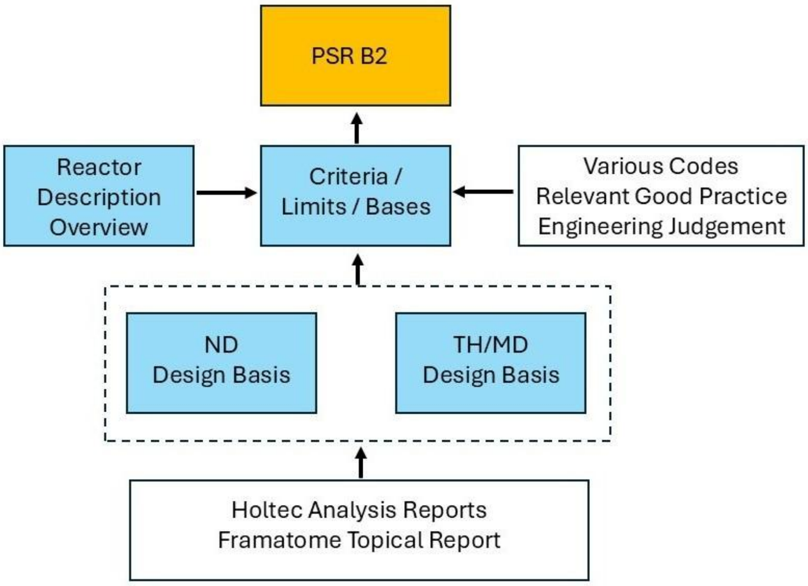

Figure 4: Safety Case Structure for Part B Chapter 2

The criteria and limits document includes all the safety case requirements relating to the fuel clad failure mechanisms and other parameters relating to the safety functions. In broad terms, the safety case requirements are divided into a small number of discipline areas, namely Fuel Rod Performance, Nuclear Design, Thermal-hydraulics, Mechanical Design, Design Basis Accidents, Dry Storage and Life-cycle Aspects.

In the cases of Nuclear Design and Thermal-hydraulic Design & Mechanical Performance, detailed design basis information is provided within the respective design basis documents. No equivalent document has been produced for Fuel Rod Performance because no detailed fuel rod assessments are available at Step 2 of GDA.

2.6 DEFINING THE SAFE ENVELOPE OF OPERATION FOR FUEL AND CORE

This sub-chapter addresses the following claim:

Claim 2.2.12.1: Safety case requirements relating to fuel and core components are established which define the safe envelope of operation.

The claim is supported by two arguments. The first argument describes the process for identifying risks and failure modes for the fuel and core components, and how these are captured, described in 2.6.1. The second argument describes how criteria are applied to the risks and failure modes identified, as described in 2.6.2.

2.6.1 Identification of Risks and Failure Modes

2.2.12.1 -A1: The risks and failure modes for the fuel and core components are defined and captured within a set of safety case requirements.

Evidence for the Argument 2.2.12.1 -A1 is:

- HI-2241327-R1, Holtec SMR-300 GDA Fuel Design Criteria and Limits [24]: This document presents requirements and acceptance criteria associated with the design of the fuel assemblies, RCCAs, and NSAs to be used within the SMR-300 core. These requirements perform the same function as the Specified Acceptable Fuel Design Limits (SAFDLs) in the US context.

The following sub-chapters provide an overview of the position at Step 2 of GDA in relation to establishing the safety case requirements relating to fuel and core components which capture risk and failure modes for those items.

2.6.1.1 Derivation of the List of Safety case requirements for Fuel and Core

The fuel and core components are described in the Overview of Holtec SMR-300 Fuel Design and Core Components document [22]. That document summarises high-level safety and functional requirements applicable to GAIA fuel assemblies, RCCAs, neutron source assemblies and other components that are to be used in the SMR-300 reactor.

The applicability of each safety and functional requirement in [22] is dependent upon the conditions of operation, namely normal operation, fault and hazard conditions, and the location along the fuel route to which these conditions apply, i.e. shipping, handling, reactor operation and storage. This dependency arises because each of the conditions of operation has the potential to give rise to a particular group of failure or degradation mechanisms, such as excessive fuel clad strain, fuel clad corrosion, excessive stress in components and so on.

For these reasons, the following steps have been taken to list and define the safety case requirements relating to fuel, RCCAs and neutron source assemblies [24]:

- Review of the fuel route for SMR-300.
- Review of USNRC documentation, specifically NUREG-0800 Section 4.2 [29].
- Cross-check of requirements and criteria against information published by the Organisation for Economic Cooperation and Development (OECD), EPRI, and within the fuel vendor’s Topical Reports.
- Application of engineering judgement or experience and brainstorming.

- Comparison of SMR-300 operating characteristics to other PWRs, e.g. see [30].

The design and operating characteristics of the SMR-300 core are regarded as being very similar to those of other PWRs in terms of fuel rod linear rating, temperature, and flow characteristics etc. Therefore, the above methodology is regarded as being both sufficient and suitable for specifying requirements for the SMR-300 core. The output from the above process is a list of safety case requirements which account for failure and degradation mechanisms which could challenge the safe use of fuel assemblies, RCCA and neutron source assemblies. The mechanical and chemical (corrosion) aspects of the failure and degradation mechanisms are described in Part B Chapter 19 [10] and Chapter 23 [11] respectively.

2.6.1.2 Demonstration of Completeness

The safety case requirements for fuel, RCCAs and neutron source assemblies have been derived following a review and cross-check of various sources of information produced by the fuel vendor, USNRC, OECD and EPRI.

Use has also been made of fuel requirements listed in Table 1 of the Office for Nuclear Regulation (ONR) Project Assessment Report listing fuel requirements applicable to the Sizewell B dry store [39]. This has been done on the assumption that many of the requirements listed in that table are also applicable to the SMR-300.

A check of the list of safety case requirements against the clauses within IAEA SSG-52 has also been performed [35]. This check did not lead to any omissions being identified.

It is therefore judged that all reasonably practicable steps have been taken to identify safety case requirements and their associated criteria for fuel and core, consistent with the objectives of Step 2 of GDA and commensurate with the current maturity of the reactor design. However, it is possible that further development of the list will be required following Step 2 of GDA, especially in regard to the criteria which have yet to be determined.

2.6.1.3 Grouping of the List of Safety case requirements

The safety case requirements for the fuel and core have been grouped by the following disciplines:

- Fuel Rod
- Fuel Assembly Skeleton
- RCCAs
- Neutron Source Assemblies
- Nuclear Design
- Thermal-hydraulic Design
- Design Basis Accidents
- Dry Storage
- Life-cycle Aspects

Fuel Rod safety case requirements relate mainly to the thermal-mechanical performance of fuel pellets and fuel clad during normal operation and Anticipated Operational Occurrences (AOOs). These requirements mainly support the confinement safety function.

Fuel Assembly Skeleton safety case requirements relate mainly to the thermal-mechanical performance and structural integrity of the fuel assembly skeleton components, apart from fuel rods. These requirements are applicable to shipping, handling, reactor operation and storage conditions, and contribute to supporting the core reactivity control, heat management and confinement safety functions. In particular, satisfying fuel assembly skeleton criteria contributes to the demonstration that the reactor core maintains a regular geometry for the correct insertion of RCCAs and the adequate removal of heat.

RCCA safety case requirements relate to the thermal-mechanical performance and structural integrity of RCCAs for normal operation, AOOs and Design Basis Accident (DBA) conditions. These requirements support the core reactivity control safety function, including for safe shutdown conditions.

Neutron Source Assembly safety case requirements relate to the structural integrity of those components and the confinement of source material.

Nuclear Design safety case requirements relate mainly to power peaking factors, reactivity coefficients and shutdown margin. These requirements support the core reactivity control safety function and contribute to supporting the heat management and confinement safety functions.

Thermal-hydraulic safety case requirements relate mainly to the management of heat transfer from the fuel to the reactor coolant, for all conditions of operations, including post fault conditions. These requirements support the heat management safety function.

Design Basis Accident safety case requirements concern the performance of fuel during and following DBA events. These safety case requirements relate mainly to demonstrating adequate management of heat transfer from the fuel to the reactor coolant during DBAs and contribute to demonstrating the number of fuel clad failures is ALARP for these events.

Dry Storage safety case requirements support the safety functions of confinement and heat management during dry storage.

Life-cycle safety case requirements relate to the various means by which the actual condition and design configuration of the fuel assemblies, RCCAs and neutrons sources are shown to be consistent with the assumptions and inputs of the safety assessments. These requirements include manufacturing quality control and surveillance, inspections on-site and demonstration of operability of fuel assemblies and other core components. These requirements contribute to supporting the core reactivity control, heat management and confinement safety functions.

Regarding generic and cycle-specific aspects of reactor operations, fuel assembly skeleton, thermal-Hydraulic Design and DBA analysis are likely to be mostly or fully generic, with little or no cycle-specific element. The other topics will generally have more cycle-specific aspects, especially Nuclear Design and life-cycle aspects, such as demonstrating operability of fuel for a reload.

2.6.2 Applied Criteria to Address Risks and Failure Modes

2.2.12.1 –A2: The set of safety case requirements include acceptance criteria for the failure modes, with reference to internationally recognised sources of information and engineering judgement.

Evidence for the Argument 2.2.12.1 -A2 is:

- HI-2241327-R1, Holtec SMR-300 GDA Fuel Design Criteria and Limits [24]: This document presents requirements and acceptance criteria associated with the design of the fuel assemblies, RCCAs, and NSAs to be used within the SMR-300 core.

The following sub-chapters provide an overview of the position at Step 2 of GDA in relation to defining acceptance criteria for the safety case requirements relating to fuel and core components and their relation to the broader safety objectives.

2.6.2.1 Information used to Establish Acceptance Criteria

Source materials used to establish criteria for the safety case requirements are listed in Sub-chapter 2.6.1.

Fuel Rod criteria are based mainly upon information contained within fuel vendor Topical Reports and information relating to UK and international practice.

Criteria relating to mechanical performance of fuel assemblies, RCCAs and neutron source assemblies are based mainly upon the ASME BPVC, Section III. These criteria are widely applied within the UK and internationally. No attempt has yet been made to specify the ASME service level applicable to each requirement for a given condition of operation; this will be done by the fuel vendor following Step 2 of GDA.

Nuclear design limits for power shaping have been derived on the basis of iteration and convergence between the Nuclear Design and Transient Analysis technical areas.

Core reactivity coefficients are required to be [REDACTED] throughout a cycle of operation and at every power level, and sufficient [REDACTED] reactivity insertion is required for shutdown conditions. These reactivity criteria are intended to demonstrate control of core reactivity for all modes of operation.

Several criteria relate to chemistry control of reactor water, spent fuel pool water and the internal environment of the Multi-Purpose Canister (MPC) for dry storage. In the cases of reactor water and spent fuel pool water, guidance is summarised in the water chemistry strategic plan for the SMR-300 [40].

Criteria applied to DBA assessments re-state the criteria applied by the Deterministic Safety Assessment discipline within Holtec International, further details of this are presented in Part B Chapter 14 [13].

Criteria relating to dry storage are based mainly upon EPRI guidance, engineering judgement or experience (noting Holtec's extensive experience of manufacturing dry storage systems).

No criteria have yet been established for the life-cycle requirements, these will be developed following Step 2 of GDA.

2.6.2.2 Relation of Safety Case Requirements to Safety Objectives

The Top-Level Safety Functions (TLSFs) are supported by a larger set of Lower-Level Safety Functions (LLSFs) which, if satisfied, ensure the TLSFs are also satisfied (see also Part B Chapter 14 [13]). The safety case requirements for fuel and core have therefore been listed individually against each relevant LLSF, presented in a table in the GDA Fuel Design Criteria and Limits document [24], in order to show how they contribute to ensuring the top-level safety functions are satisfied.

Furthermore, relevant LLSFs are listed against each postulated fault within the Preliminary Fault Schedule (PFS) for the SMR-300 [41]. Therefore, the connection between each fault entry within the PFS and the relevant safety case requirements can be inferred using [24]. This facilitates the identification of the “golden thread”, as introduced in [1], between fault identification, fault assessments, SSCs important to safety and the associated safety analysis within the Reactor topic.

2.6.3 CAE Summary

This sub-chapter summarises the arguments and evidence in support of the below claim:

Claim 2.2.12.1: Safety case requirements relating to fuel and core components are established which define the safe envelope of operation.

2.2.12.1 -A1: The risks and failure modes for the fuel and core components are defined and captured within a set of safety case requirements.

Risks and failure modes associated with PWR fuel and other core components have been identified via the review of various pieces of publicly available information, including that produced by the fuel vendor, USNRC, OECD and EPRI. The operating characteristics of the SMR-300 core are sufficiently similar to those of other PWRs, such that the identified risks and failure modes are regarded as being applicable to the SMR-300. These risks and failure modes are presented in the GDA Fuel Design Criteria and Limits document [24].

2.2.12.1 –A2: The set of safety case requirements include acceptance criteria for the failure modes, with reference to internationally recognised sources of information and engineering judgement.

Acceptance criteria and limits applicable to the safety case requirements have been identified, outlined in [24], defining a safe envelope for operation. These criteria, if satisfied, ensure the SMR-300 satisfies the top-level safety functions relating to reactivity control, heat management and confinement of activity. Further work is required to validate the safe envelope of operation through the UK based fault studies program as outlined in Chapter 14 [13]. The process to derive and group safety case requirements, then establish limiting criteria will be applied to all fuel and core components post GDA.

Overall, it is judged that Claim 2.2.12.1 is met to a maturity judged to be appropriate for PSR is supported by the evidence presented in [24] and the above arguments.

2.7 ASSESSING THE SAFE ENVELOPE OF OPERATION FOR FUEL AND CORE

The following level-four claims are addressed in this sub-chapter:

Claim 2.2.12.2: The safety case requirements for the fuel and core components are demonstrated to be satisfied.

Claim 2.2.12.2 is supported by two arguments. The first argument describes the methods which are being defined to assess fuel and core components against safety case requirements, presented in 2.7.1. The second argument states that assessments performed thus far demonstrate the safety case requirements acceptance criteria are satisfied or are likely to be satisfied post Step 2 of GDA, as discussed in 2.7.2.

Claim 2.2.1.2: Control rods provide appropriate reactivity control during normal reactor operation and the means for reactor shutdown during a trip.

Claim 2.2.1.2, relating to reactivity control via control rods, is addressed in 2.7.3 and is supported by three arguments. The first argument relates to the appropriateness of the control rod design in assuring reactivity control is achieved. The second argument relates to the control rod operation during normal operation. The third argument relates to control rod operation during a reactor trip.

Claim 2.2.2.4: The core is always maintained in a coolable geometry

Claim 2.2.2.4, relating to core heat removal, is addressed in 2.7.4 and supported by two arguments. The first argument relates to the integrity of fuel assembly geometry, while the second argument relates to the integrity of non-fuel reactor internal components geometries, in relation to maintaining a coolable core geometry.

Claim 2.2.3.6: The fuel rod clad integrity is maintained during normal operation and AOOs.

Claim 2.2.3.6, relating to confinement of radioactive material, is addressed in 2.7.5 and is supported by one argument. The argument states that criteria relating to the integrity of the fuel cladding are satisfied for normal operation and AOO.

2.7.1 Assessment Methodologies

2.2.12.2 -A1: Sound methods for assessing fuel and core components against the safety case requirements are in the process of being defined.

Evidence for the Argument 2.2.12.2 -A1 is:

- HI-2241458-R1, Holtec SMR-300 GDA, Nuclear Design Basis Report [25]: The purpose of this report is to present an overview of the nuclear design work for the SMR-300 for the GDA. The report presents the safety case context of the nuclear design work, the software codes used to perform the analysis, an overview of the applied nuclear design limits and a summary of the assessments against those limits. Nuclear design assessments to date have been performed in accordance with code vendor methodology guidance and this guidance will continue to be applied as the nuclear design assessments mature to completion.

- HI-2241288-R1, Thermal and Mechanical Design Basis [23]: This document presents the methodology and scope for demonstrating the integrity of the fuel assembly and core components under both mechanical and thermal loads. It introduces the key thermal and mechanical design limits, and assessment methodologies. Additionally, the conclusions drawn from a subset of casualty analysis are presented to provide confidence that there are no fundamental shortfalls in the design of the SMR-300 fuel and core system. Thermal-hydraulics assessments to date have been performed in accordance with accepted international good practice and with support from the fuel vendor.
- Framatome, Applicability of Framatome Fuel Methods for Holtec Fuel, ANP-4139NP, Revision 0, 2025 [30]: This document presents a review of the applicability of fuel vendor methodologies to the SMR-300. The review evaluated Topical Reports relating to aspects such as the use of the COPERNIC fuel performance code, methodologies applied to the assessment of M5® fuel rods, methodologies applied to the assessment of fuel assembly mechanical and thermal-hydraulics performance. These Topical Reports have previously been submitted to the USNRC by the fuel vendor, for licensing GAIA fuel assemblies for use in Westinghouse PWRs in the US. This review concluded that the fuel vendor methodologies are valid for use in assessing the SMR-300 core.

The following sub-chapters provide an overview of the position at Step 2 of GDA in relation to the methodologies being developed within the Reactor topic area for fuel and core assessments.

2.7.1.1 Nuclear Design Methodology

Nuclear design supports the following safety functions: control of reactivity, heat transfer from the fuel to the reactor coolant and the confinement of activity. Requirements relating to nuclear design assessments are specified in [24].

The core design at the start of each cycle of operation contains sufficient excess reactivity to ensure power generation for the intended duration of the cycle. End-of-cycle typically corresponds to the point in time at which the boron concentration in the reactor coolant is approximately zero. Burnup limits are specified for the peak fuel assembly average burnup and the peak fuel rod average burnup. The first of these burnup limits is intended to ensure mechanical performance assessments are valid, whereas the second of these burnup limits is intended to ensure fuel rod performance assessments are valid. Different fuel rod burnup limits apply to non-Gd fuel rods and Gd-doped fuel rods.

Reactivity coefficients are specified to be [REDACTED] for all power conditions for which the RCCAs are withdrawn and for all times during a cycle of operation, ensuring inherent stability and reactivity controllability from start-up to full power operating conditions. The [REDACTED] fuel Doppler temperature coefficient provides rapid feedback in response to changes to fuel temperature. A [REDACTED] moderator temperature coefficient is ensured by suitable use of burnable absorber. Furthermore, the nuclear design analysis is required to demonstrate sufficient shutdown margin exists for an at-power trip and for subsequent longer-term changes to core parameters.

Reactivity control is further demonstrated by assessing the rates of reactivity change resulting from RCCA insertions and variations in boron concentration in the reactor coolant. The inherent stability of the core is also verified through its response to perturbations, including those caused by RCCA axial movements.

Nuclear design limits for power peaking factors have been specified to ensure local power peaking is acceptable, no fuel melt occurs, and no Departure from Nucleate Boiling (DNB) occurs. Power peaking limits have been derived and converged upon following an iterative process of analysis between Nuclear Design and Transient Analysis technical areas. Constraining power peaking and fuel burnup contributes to demonstrating the integrity of the fuel clad.

Nuclear design analysis has been performed for Step 2 of GDA by Holtec International using methodologies recommended by the reactor physics code vendor. These methodologies are described within the code vendor manual for the CASMO-SIMULATE reactor physics package and are summarised in the Nuclear Design Basis Report [25].

Benchmarking of [REDACTED] calculations is currently ongoing. The purpose of the benchmarking exercise is to demonstrate that derived uncertainties for SMR-300 reactor physics parameters are consistent with those assessed by the USNRC as part of the licensing process for the software. Hence no formal uncertainties have been defined for most SMR-300 reactor physics parameters at Step 2 of GDA. Consequently, most reactor physics parameters have been assessed on a best-estimate basis. The main exception is shutdown margin, where reasonable assumptions have been made concerning applied uncertainties [25].

Refinement of the applied methodologies within the nuclear design area is required following Step 2 of GDA, especially in respect to validation and derivation of uncertainties.

2.7.1.2 Thermal-Hydraulic Design Methodology

Thermal-hydraulic design supports the following safety functions: heat transfer from the fuel to the reactor coolant and confinement of activity. Requirements relating to thermal-hydraulics design assessments are specified in [24].

Thermal-hydraulic analysis for Step 2 of GDA has been performed by Holtec International using methodologies presented in Thermal and Mechanical Design Basis document [23]. The applied methodologies are consistent with best practice guidance produced by the OECD and cover the following high-level topics:

- Minimum Departure from Nucleate Boiling Ratio (MDNBR).
- Hydrodynamic and hydraulic stability.
- Crud deposition and induced corrosion.

The core thermal-hydraulic code COBRA-FLX is the primary software for thermal-hydraulics analysis. This software was developed by the fuel vendor and is approved by the USNRC for use in licensing applications in the US. The code utilises the subchannel analysis concept developed for reactor core analysis, solving conservation equations for mass, momentum and energy to obtain fluid enthalpy and flow distributions as well as momentum pressure drop.

COBRA-FLX is used to calculate the margin for MDNBR. To perform this calculation, COBRA-FLX applies inputs for core flow, core inlet temperature, core exit pressure and total core power derived using the RELAP5-3D software.

The applicability of COBRA-FLX to analysing the thermal-hydraulics of the SMR-300 core has been assessed by the fuel vendor [30]. The review involved an evaluation of a Topical Report for core thermal hydraulics and methodologies, submitted to the USNRC by the fuel vendor, for licensing GAIA fuel assemblies for use in Westinghouse PWRs in the US. The applicability review therefore focused on comparing the in-reactor operating conditions of the Westinghouse PWRs with those of the SMR-300. In the case of thermal-hydraulic design, the following Topical Report was reviewed:

- Core Thermal-hydraulic Codes and Methodologies, the aspects of which include:
    - COBRA-FLX code.
    - ORFEO Critical Heat Flux (CHF) correlations.
    - Fuel rod bow and fuel assembly bow.
    - DNB propagation.

The applicability review by the fuel vendor concluded that COBRA-FLX is acceptable for use in assessing core thermal-hydraulics for the SMR-300, primarily on the basis of the similarity of its operating conditions to those of the Westinghouse reactors. [REDACTED]. Compliance with these limitations and conditions will be assessed following Step 2 of GDA as the thermal-hydraulic methodology is developed further.

Inputs to assessments of heat transfer assume suitably bounding or pessimistic operating conditions for the reactor and accounts for uncertainties within various factors, including power peaking. Account is taken of various parameters such as coolant temperature, pressure, flow bypass, flow maldistribution, pressure drop and hydraulics loads. Justification for the inputs, assumptions and applied uncertainties within the thermal-hydraulic assessments is presented in the Thermal and Mechanical Design Basis document [23].

For SMR-300, which incorporates GAIA fuel assemblies, the ORFEO-GAIA and ORFEO-NMGRID CHF correlations have been applied within the assessments of MDNBR. The assessment of DNB is used to assess the performance of the fuel rod for the conditions modelled and therefore contributes to demonstrating the integrity of the fuel clad.

A feature of the SMR-300 core is the location of the two reactor cold leg inlet nozzles. These are separated azimuthally by [REDACTED], leading to [REDACTED]. A FDD has been designed and included within the reactor vessel to homogenise the coolant flow that enters the core from below [22].

The approach taken to designing the FDD and the methodology used to assess its thermal-hydraulic performance are presented in [23].

Holtec International is currently developing a strategy for validating the methodologies and assessments for the SMR-300. To date, a limited set of calculations have been performed which benchmark COBRA-FLX against ANSYS. The results of this benchmarking show that both codes produce results which are consistent with each other [23]. Commitment C_Fuel_129 has been raised to extend the current substantiation to include time-dependent modelling of coolant flow and sensitivity studies, as described in sub-chapter 2.8.3.

It is Holtec International’s intention to perform instrumented scaled tests of reactor core flow prior to operation of the first SMR-300. Holtec is presently assessing the viability of using an existing test rig (such as those operated by the fuel vendor) and is also giving consideration to constructing a test rig which is bespoke to the SMR-300. Such a test rig will be used to validate numerical analysis performed for the SMR-300. Thermal-hydraulic validation of the UK SMR-300 will therefore benefit from these various tests following Step 2 of GDA.

2.7.1.3 Fuel Rod Performance Methodology

Fuel rod design supports the following safety functions: control of reactivity, heat transfer from the fuel to the reactor coolant and the confinement of activity. Requirements relating to nuclear fuel rod performance are specified in [24].

Fuel vendor methodologies will be applied to the assessment of fuel rod performance for the SMR-300. These methodologies are presented in the Thermal and Mechanical Design Basis document [23].

The applicability of fuel vendor methodologies to the SMR-300 has been performed by the fuel vendor [30]. The review involved the evaluation of certain Topical Reports relating to the assessment of M5® fuel rods submitted to the USNRC, by the fuel vendor, for licensing GAIA fuel assemblies for use in Westinghouse PWRs in the US. The applicability review of those Topical Reports therefore focused on comparing the in-reactor operating conditions of the Westinghouse PWRs with those of the SMR-300.

Pertinent to fuel performance, the review assessed:

- M5® Material Topical Report [42], aspects of which include:
    - Plant operating parameters.
    - Applicability of bounding fuel rod rating histories.
    - Inputs to assessments: fuel rod design, back-fill pressure, linear heat generation rate, fast flux and densification kinetics.
- COPERNIC Fuel Rod Design Topical Report [43], aspects of which include:
    - Fuel rod design.
    - Fuel rod manufacturing characteristics.
    - Fuel rod thermal-mechanical performance.
    - Fuel rod irradiation history.
    - Thermal-hydraulic conditions.
    - COPERNIC fuel performance code and models (e.g. fission gas release, mechanical, corrosion etc.).
    - Limitations of use.
    - Validation.
- Program to Determine In-reactor Performance of BWFC Fuel Cladding Creep Collapse [44], supporting sub-chapter 2.7.1.3. Aspects covered include:
    - CROV clad collapse code.

The applicability review by the fuel vendor concluded that applied methodologies for assessing fuel rod performance were valid for the SMR-300, primarily on the basis of the similarity of its operating conditions to those of the Westinghouse reactors.

2.7.1.4 Fuel Assembly Mechanical Performance Methodology

Fuel assembly mechanical design supports the following safety functions: control of reactivity, heat transfer from the fuel to the reactor coolant and the confinement of activity. Requirements relating to fuel assembly skeleton mechanical design assessments are specified in [24].

Fuel vendor methodologies will be applied to the assessment of fuel assembly mechanical performance for the SMR-300. These methodologies are presented in the Thermal and Mechanical Design Basis document [23].

The scope of safety assessments for fuel assembly mechanical performance mainly involves the assessment of loads imparted upon fuel assemblies as a result of manufacturing, normal shipping, handling, in-reactor, storage conditions, faults and hazards.

For the in-reactor situation, the main loads imparted upon the fuel assembly arise from compression between the top and bottom core plates, coolant flow, RCCA drop impact during reactor trip, fault and hazard conditions, such as seismic-Loss Of Coolant Accident (LOCA) events. The mechanical performance assessment is required to demonstrate that fuel assemblies retain a regular geometry for all conditions of operation and design basis events, meaning RCCAs can be correctly and fully inserted into the core, heat transfer from the fuel to the reactor coolant is not impeded and fuel rod integrity is not jeopardised.

A review of the applicability of the fuel vendor methodology in assessing the mechanical performance of fuel assemblies to be used in the SMR-300 core has been performed by the fuel vendor [30]. The review involved an evaluation of certain Topical Reports relating to the assessment of fuel assembly mechanical performance, submitted to the USNRC by the fuel vendor for licensing GAIA fuel assemblies for use in Westinghouse PWRs in the US. The applicability review of those Topical Reports therefore focused on comparing the in-reactor operating conditions of the Westinghouse PWRs with those of the SMR-300.

Pertinent to fuel assembly mechanical performance, the following Topical Reports were reviewed:

- GAIA Mechanical Fuel Design Topical Report [45], aspects of which include:
    - Use of ANSYS software and hand calculations.
    - Application of the ASME III code.
    - Stress, strain, loading limits, strain fatigue, fretting wear.
    - Fuel rod / fuel assembly bow and growth.
    - Fuel assembly lift-off.
    - Hydriding.
- Q12 Structural Material Topical Report [46], concerning:
    - Burnup limit applicable to material assessment.
- Seismic Topical Report [47], aspects of which include:
    - Fuel assembly response to seismic events (fuel assembly characteristics).
    - Plant-specific aspects of fuel seismic assessment.

The applicability review by the fuel vendor concluded that applied methodologies for assessing fuel assembly mechanical performance were valid for the SMR-300, primarily on the basis of the similarity of its operating conditions to those of the Westinghouse reactors.

2.7.1.5 RCCA and NSA Integrity Methodology

RCCA and NSA design assessment supports the following safety function: control of reactivity.

The scope design substantiation for RCCA performance mainly involves the assessment of loads imparted upon RCCAs as a result of normal shipping, handling, in-reactor operations, storage operations, faults and hazards. For the in-reactor situation, the most significant loads imparted upon the RCCAs are judged to arise from RCCA position stepping, RCCA drops (impact with fuel assembly top nozzle) and seismic-LOCA events.

RCCA and NSA vendor methodologies will be applied to RCCA and NSA design substantiation for the SMR-300. The scope of assessments to which the methodologies apply is presented within the Fuel Design Criteria and Limits document [24]. The methodologies involve:

- Use of ANSYS software and hand calculations.
- Application of the ASME III code.
- Assessment of stress, strain, loading limits, strain fatigue, fretting wear.

Methodologies for substantiating the design of the RCCAs and NSAs will be presented following Step 2 of GDA.

2.7.1.6 Design Basis Accidents Methodology

Design basis accident methodologies are outlined in the Thermal and Mechanical Design Basis Report [23] and Part B Chapter 14 [13]. Requirements relating to DBA assessments specifically in relation to fuel performance are specified in [24].

Within the US context, DBA analysis applies to postulated events having initiating event frequencies within the range 1×10^{-2} /yr > IEF ≥ 1×10^{-4} /yr. For DBA events, the US context permits a small number of fuel rod failures provided the dose consequences satisfy specified limits. By contrast, within the UK context, no fuel clad failures (except for a small number of random failures) are permitted for an event having an IEF ≥ 1×10^{-3} /yr and the number of fuel rod failures for events having IEFs in the range of 1×10^{-3} /yr > IEF ≥ 1×10^{-5} /yr must be demonstrated to be ALARP [48]. A Commitment, C_Fuel_127 (see sub-chapter 2.8.3), is therefore raised to identify the initiating events with IEFs within the range of 1×10^{-2} /yr > IEF ≥ 1×10^{-3} /yr and to evaluate their impact upon fuel clad integrity.

Table 8 presents an overview of the fault groups considered in the SMR-300 design. Information on Fault Classifications is available in Part B Chapter 14 [13].

Table 8: Design Conditions

Fault Classification	Description
Normal Operation (NO)	Operation within specified operational limits and conditions. For a nuclear power plant, this includes start-up, power operation, shutdown, maintenance, testing and refuelling.
AOO	Condition of Normal Operation which is expected to occur one or more times during the life of the nuclear power unit.
DBA	A postulated accident that a nuclear facility must be designed and built to withstand without loss to the systems, structures, and components necessary to ensure public health and safety.
Design Extension Condition (DEC)	A subset of beyond-design-basis accidents that are considered in the design process of the facility in accordance with best-estimate methodology to keep releases of radioactive material within acceptable limits. Design extension conditions could include severe accident conditions.

Calculations of Departure from Nucleate Boiling Ratio (DNBR) have been performed for a small number of faults within the transient analysis technical area. These faults have not been formally categorised as per Table 8; although they are regarded as representing a sufficiently broad range of fault categories as to provide confidence in the safety of the SMR-300 and its robustness against a wide range of fault conditions

The main purpose of DBA assessments is to demonstrate that core geometry remains coolable and that fuel rod failures remain ALARP and within specified dose limits. It is implicitly assumed within DBA assessments that the core retains a coolable geometry and that RCCAs insert correctly into the core. The DBA assessment therefore interfaces with fuel assembly mechanical performance assessments (sub-chapter 2.7.1.4), RCCA integrity assessment (sub-chapter 2.7.1.5) and assessment of the CRD integrity (sub-chapter 2.2.6.1). This particular group of assessments will be performed by the vendor of those items following Step 2 of GDA.

2.7.1.7 Life-cycle Aspects Methodology

A description of the life-cycle aspects specific to the Reactor topic area are presented in the Fuel Design Criteria and Limits document [24].

The following capture (at a very high level) activities which are required to be performed to confirm (at various life-cycle stages) that the condition of fuel and RCCAs remain consistent with the inputs and assumptions of the safety case and its associated safety analysis.

- Inspections and surveillance programme for the manufacturing of fuel and core components, for quality assurance and adherence to design.
- Visual inspections of fresh fuel and core components being brought onto site, to confirm safe transportation to site and acceptability of use of fresh fuel.
- Visual inspections of irradiated fuel and general core performance review prior to core reload, to confirm acceptability of re-use of irradiated fuel.
- Fuel sentencing requirements for dry storage.

Thus, the above life-cycle aspects provide a signpost for future safety case development within the Fuel and Core technical area following GDA. Methods for performing the above life-cycle operations will be developed following Step 2 of GDA.

2.7.2 Assessment Results

2.2.12.2 -A2: Assessments performed thus far demonstrate the safety case requirements acceptance criteria are satisfied or are likely to be satisfied post Step 2 of GDA.

Evidence for the Argument 2.2.12.2 -A2 is:

- HI-2241458-R1, Holtec SMR-300 GDA, Nuclear Design Basis Report [25]: The purpose of this report is to present an overview of the nuclear design work for the SMR-300 for the GDA. The report presents the safety case context of the nuclear design work, the software tools used to perform the analysis, an overview of the applied nuclear design limits and a summary of the assessments against those limits.
- HI-2241288-R1, Thermal and Mechanical Design Basis [23]: This document presents the methodology and scope for demonstrating the integrity of the fuel assembly and core components under both mechanical and thermal loads. It introduces the key thermal and mechanical design limits, and assessment methodologies. Additionally, the conclusions drawn from a subset of casualty analysis are presented to provide confidence that there are no fundamental shortfalls in the design of the SMR-300 fuel and core system.
- Framatome, Applicability of Framatome Fuel Methods for Holtec Fuel, ANP-4139NP, Revision 0, 2025 [30]: This document presents the applicability of fuel vendor methodologies to the SMR-300. The review involved the evaluation of certain Topical

Reports relating to the assessment of M5® fuel rods, submitted to the USNRC, by the fuel vendor, for licensing GAIA fuel assemblies for use in Westinghouse PWRs in the US.

The safety assessments for the Reactor topic are intended to demonstrate the following:

- No fuel clad failures occur during normal operations or as a consequence of AOO.
- Core reactivity remains controllable for all conditions of operation within the design basis, e.g. power operation, hot shutdown, cold shutdown.
- Core maintains a regular geometry for all conditions of operation within the design basis, such that it remains coolable and does not jeopardise the correct insertion of RCCAs (e.g. during a reactor trip).
- Structural integrity of RCCAs in-reactor is maintained for all conditions of operation within the design basis.
- Requirements and criteria applicable to ex-reactor conditions are satisfied (e.g. shipping, handling, storage of fuel assemblies, RCCAs and neutron source assemblies).

The complete safety demonstration outlined above is mostly outside the scope of Step 2 of GDA. Nevertheless, a certain amount of safety assessment within the scope of the Reactor topic has been performed at this iteration of the PSR, the results of which are summarised in the following sub-chapters.

2.7.2.1 Nuclear Design Assessment

The nuclear design assessments for the SMR-300 are presented in the Nuclear Design Basis Report [25]. These assessments are based on a 'generic' fuel management strategy that starts with Cycle 1 and ends at Cycle 5 (equilibrium cycle).,

The analysis presented in [25] covers various power and shutdown conditions and demonstrates the following:

- Assessed reactivity coefficients, in particular fuel Doppler temperature coefficient and moderator temperature coefficient, are negative at all power levels throughout the cycle.
- Power peaking factors satisfy specified limits.
- Positive shutdown margin exists for both short and long periods following reactor trip.
- RCCA reactivity insertion rates are acceptable (to be confirmed following further iteration between Nuclear Design and Transient Analysis).
- Core stability is demonstrated as acceptable (perturbations are controllable).

Applied limits for fuel assembly average burnup, and non-Gd fuel rod average burnup, are satisfied with positive margin. However, the limit for Gd fuel rod average burnup is slightly exceeded. This negative margin for Gd fuel is considered small enough to be reversible with minor core design optimisation following Step 2 of GDA (e.g. quadrant-to-quadrant shuffling of fuel assemblies). A Commitment (C_Fuel_128) is raised to progress analysis in this area in accordance with the Design Management process [49]. Target for Resolution - Issue of Pre-Construction SSEC. Further discussion on this issue is presented in Sub-chapter 2.8.3.

It is concluded that the nuclear design of the SMR-300 core is broadly acceptable subject to satisfactory conclusion of validation, derivation of uncertainties and core optimisation to ensure positive margin to all burnup limits.

2.7.2.2 Thermal-Hydraulic Design Assessment

The thermal-hydraulics design assessments for the SMR-300 are presented in the Thermal-Mechanical Design Basis Report [23].

Several representative fault transients have been assessed for MDNBR using the core thermal-hydraulic code COBRA-FLX. The analyses include a range of postulated faults which, collectively, are assumed to represent a set of AOOs, DBAs and Beyond Design Basis Accident (BDBA) that are typically considered in the safety assessments of PWRs, namely:

- Small Break (SB) LOCA.
- Inadvertent actuation of the Primary Decay Heat Removal System (PDH).
- Inadvertent actuation of the Secondary Decay Heat Removal System (SDH).
- Turbine Trip coincident with a Loss of Offsite Power (TTLOOP).
- Anticipated Transient Without Scram (ATWS).

Results show that MDNBR for these events show positive margin to the limit, accounting for pessimisms and uncertainty. Hence the thermal-hydraulic properties of the SMR-300 core are shown to be acceptable in terms of providing effective heat removal from the fuel to the reactor coolant. In particular, the results for the ATWS event indicates that the inherent feedback effects of the SMR-300 plant are effective in mitigating flow reduction and temperature rise.

Additionally, the core thermal-hydraulic steady-state analysis shows the FDD is effective at reducing flow maldistribution to acceptably low values such that inputs and assumptions of the thermal-hydraulics assessments are valid [23]. Additionally, consequences of the thermal-hydraulic design of the SMR-300 within the nuclear design technical area have been assessed and found to be negligible [23].

It is concluded that the thermal-hydraulic design of the SMR-300 core is broadly acceptable subject to satisfactory conclusion of testing and validation, as outlined in Sub-chapter 2.7.1.2.

2.7.2.3 Fuel Rod Performance Assessment

No fuel rod performance analysis for the in-reactor situation has been undertaken for the SMR-300 at Step 2 of GDA. It is judged that fuel rod and clad integrity criteria will be satisfied for the SMR-300 for the following reasons, see Thermal-Mechanical Design Basis Report [23]:

- Applied fuel rod performance criteria presented in [24] are similar to, or identical to, equivalent criteria applied to many other PWRs by the fuel vendor.
- SMR-300 in-core operating conditions are similar to those of other PWRs for which fuel rod performance has been assessed by the fuel vendor. Hence, the inputs and boundary conditions applied to fuel rod performance assessments for the SMR-300 will likewise be similar to those applied to other PWRs by the fuel vendor.
- Software codes and methodologies applied by the fuel vendor for assessing fuel rod performance have a proven track record for reliably predicting the performance of fuel in PWRs. These methods are underpinned by a substantial validation dataset.
- The applicability of fuel vendor methodologies has been assessed by the fuel vendor and found to be applicable to the SMR-300 (see sub-chapter 2.7.1.3).

- Results of fuel rod performance analysis for normal operating conditions and AOOs, produced by the fuel vendor for those other PWRs, show positive margins to limits.
- Positive OPEX exists concerning the use of GAIA fuel assemblies in other PWRs. In particular, no fuel rod failures have been identified to date (see sub-chapter 2.7.2.8).

It is concluded that the fuel rod design for the SMR-300 is acceptable and that the fuel clad will remain intact for normal operating and AOO conditions for the SMR-300, thus demonstrating the integrity of the fuel clad as a containment barrier.

2.7.2.4 Fuel Assembly Mechanical Performance Assessment

No fuel skeleton mechanical performance analysis has been undertaken for Step 2 of GDA. It is judged that fuel mechanical performance criteria will be satisfied for the SMR-300 for the following reasons, see Thermal-Mechanical Design Basis Report [23]:

- The lower-level mechanical performance criteria which underpin the fuel assembly load limits are similar to, or identical to, equivalent criteria applied to many other PWRs, in particular for guide tube and spacer grid deformation.
- It is assumed that standard fresh fuel containers and modes of transportation will be used for shipping fresh fuel to site, meaning no adverse latent structural conditions are anticipated for this phase.
- Fuel assembly handling equipment on site will be designed to ensure that the handling load limits for fresh fuel assemblies and irradiated fuel assemblies will be satisfied.
- SMR-300 in-reactor load conditions will be similar to those of other PWRs, e.g. loads upon fuel assemblies arising from dropped RCCAs during a reactor trip and seismic/LOCAs (accounting for UK context).
- Software codes and methodologies applied by the fuel vendor for assessing the mechanical performance of fuel assemblies have a proven track record for reliably predicting the performance of fuel in PWRs. These methods are underpinned by a substantial validation dataset, mainly in the form of mechanical tests.
- Results of fuel skeleton mechanical performance analysis reported in fuel vendor Topical Reports show positive margins to limits. These results are considered by the fuel vendor to be applicable to the SMR-300.

It is concluded that the GAIA fuel assembly design to be used in the SMR-300 is acceptable and that the mechanical performance criteria will be satisfied for all normal operating conditions and all design basis events, thus demonstrating fuel assemblies will retain a regular geometry which permits correct and full insertion of RCCAs into the core, does not impede the transfer of heat from the fuel to the reactor coolant and does not jeopardise the integrity of the fuel rods.

As stated in 2.7.1.6, a Commitment, C_Fuel_127 (see sub-chapter 2.8.3) has been raised to identify the initiating events with IEFs within the range of 1×10^{-2} /yr > IEF ≥ 1×10^{-3} /yr and to evaluate their impact upon fuel clad integrity.

2.7.2.5 RCCA and NSA Assessment

No design substantiation for RCCAs and NSAs has been undertaken for Step 2 of GDA. It is judged that applicable criteria specified for these core components, as presented in [24], will be satisfied because:

- RCCAs and NSAs to be used in the SMR-300 core will be of a standard design used in other PWRs.
- Conditions of operation of those other PWRs are similar to those of the SMR-300, i.e. neutronic, mechanical and thermal-hydraulic loads are similar.
- RCCAs are of the HARMONI design and have been used extensively in PWRs internationally and for which positive OPEX exists (see Sub-chapter 2.7.2.8).

It is therefore judged that RCCAs will retain their structural integrity for normal operating conditions and following design basis accidents. Formal design substantiation of RCCAs and NSA will be performed following Step 2 of GDA.

2.7.2.6 DBA Assessment

Formal assessment of the SMR-300 core for design basis accidents will be presented in Part B Chapter 14 following Step 2 of GDA.

Assessment of MDNBR for a subset of fault events has been performed and is summarised in Sub-chapter 2.7.2.2. These results show positive margin to the limit for MDNBR, providing confidence that the SMR-300 core is resilient to design basis accidents, subject to completion of all DBA assessments (see [13]).

Identification, categorisation and assessment of faults will be performed following Step 2 of GDA and summarised in Part B Chapter 14 within a future iteration of the safety case.

2.7.2.7 Life-cycle Aspects Assessment

No substantiation of life-cycle requirements, as presented in [24], has been performed for Step 2 of GDA. Therefore, this aspect of the safety case will be developed following Step 2 of GDA.

2.7.2.8 Operational Experience

OPEX concerning the use of GAIA fuel assemblies within PWRs has been supplied to Holtec Britain by the fuel vendor. To date, over 500 GAIA fuel assemblies [REDACTED] have been loaded into PWRs at [REDACTED]. This OPEX shows that the GAIA fuel assembly has had no loss of cladding integrity or other component failure while in operation at any time (this includes Lead Fuel Assembly (LFA) operation and batch operation to date).

In general terms, the operating conditions of the reactors loaded with GAIA fuel are considered by the fuel vendor to be consistent with the operating conditions of the SMR-300, as they stand at this point in its design development. The maximum fuel assembly average burnup associated with the OPEX is 58 GWd/MTU, which is greater than the fuel assembly average burnup limit applied in the nuclear design assessments (sub-chapter 2.7.1.1).

The fuel vendor has also confirmed that it considers there to be nothing unusual about the operations of those PWRs which would lead it to conclude that the OPEX for GAIA is not valid for use by Holtec in relation to the expected performance of the fuel when loaded into the SMR-300. Moreover, the OPEX is regarded by the fuel vendor as providing a broad range of experience that would support its applicability to the SMR-300.

Additionally, the fuel vendor has supplied over 7,700 HARMONI RCCAs for use in 97 reactors around the world, including the United States. Performance of the HARMONI RCCA type is generally regarded to be very good.

2.7.3 Demonstration of Reactivity Control

2.2.1.2 -A1: Appropriate design of the RCCA rod allows for control of reactivity in the core.

Evidence for the Argument 2.2.1.2 -A1 is:

- HI-2240845-R0, System Design Description for the Control Rod Control System [3]: This document provides a high-level description of the functional, performance, and safety requirements of the CRC which controls the movement of RCCAs into and out of the core.

This argument is supported by evidence presented in Sub-chapters:

- Sub-chapter 2.7.2.1 Nuclear Design Assessment, where it is shown that RCCAs, in combination with other core design features, are able to control reactivity and provide adequate short term and long-term shutdown margin.
- Sub-chapter 2.7.2.5 RCCA and NSA Assessment where it is shown that RCCAs will maintain their structural integrity for normal operations and DBAs.

On the basis of these assessments, it is concluded that the RCCA HARMONI design is appropriate for use in the control of reactivity within the SMR-300 core.

2.2.1.2 -A2: Regulating groups of RCCAs can be stepped in or out of the core to control reactivity during normal operation.

Evidence for the Argument 2.2.1.2 -A2 is:

- HI-2240845-R0, System Design Description for the Control Rod Control System [3]: This document provides a high-level description of the functional, performance, and safety requirements of the CRC which controls the movement of RCCAs into and out of the core.

This argument is supported by evidence presented in Sub-chapter 2.2.6 concerning the design of the CRC. The CRC including CDS, CRD and RPI, facilitates control of reactivity during normal operation by inserting and withdrawing control rods into the core as per operator demand. Further information regarding the design, operation and regulatory requirements and codes that the CRC is designed to is presented in [3].

2.2.1.2 -A3: RCCAs are fully inserted when a reactor trip occurs (insertion by gravity).

Evidence for the Argument 2.2.1.2 -A3 is:

- HI-2240845-R0, System Design Description for the Control Rod Control System [3]: This document provides a high-level description of the functional, performance, and safety requirements of the CRC which controls the movement of RCCAs into and out of the core.

This argument is supported by evidence presented in Sub-chapter 2.7.2.4 addressing fuel assembly mechanical performance, which concludes that fuel assemblies will retain a regular geometry during normal operating conditions and following design basis accidents, permitting correct and full insertion of RCCAs into the core following a trip. Furthermore, it is judged that

RCCA integrity will be assured for these conditions, as outlined in sub-chapter 2.7.2.5 concerning the assessment of RCCA integrity.

2.7.4 Demonstration of Coolable Geometry

2.2.2.4 -A1: Fuel assemblies maintain a regular geometry for all design basis events.

Evidence for the Argument 2.2.2.4 -A1 is:

- HI-2241327-R1, Holtec SMR-300 GDA Fuel Design Criteria and Limits [24]: This document presents requirements and acceptance criteria associated with the design of the fuel assemblies, RCCAs, and NSAs to be used within the SMR-300 core.
- HI-2241288-R1, Thermal and Mechanical Design Basis [23]: This document presents the methodology and scope for demonstrating the integrity of the fuel assembly and core components under both mechanical and thermal loads. It introduces the key thermal and mechanical design limits, and assessment methodologies. Additionally, the conclusions drawn from a subset of casualty analysis are presented to provide confidence that there are no fundamental shortfalls in the design of the SMR-300 fuel and core system.

This argument is supported by evidence presented in Sub-chapter 2.7.2.4 addressing fuel assembly mechanical performance, which concludes that fuel assemblies will retain a regular coolable geometry for normal operating conditions and design basis accidents.

2.2.2.4 -A2: Non-fuel reactor internal structures retain a regular geometry so as not to impede the required transfer of heat from the fuel.

Evidence for the Argument 2.2.2.4 -A2 is:

- HI-2240961, Holtec SMR-300 Reactor Pressure Vessel Internals Design Description [50]: This document provides a description of the Reactor Internal Structure, including design criteria and a description of the coolant flow path.

This argument is supported by evidence presented in Sub-chapter 2.2.2, which provides a high-level description of the RIS. To provide a high degree of integrity and reliability, the RIS is designed to codes outlined in Sub-chapter 2.4.2. This ensures that RIS meets the guidelines of ASME NG-3000. The following General Design Criteria (GDC) apply to the RIS:

GDC 1 [51] and 10 CFR 50.55a [52] require that SSCs important to safety be designed to quality standards commensurate with the importance of the safety functions performed.

GDC 2 [51] requires that SSCs important to safety be designed to withstand the effects of earthquakes without loss of capability to perform their safety functions. This GDC makes the Internals structure Seismic Category 1.

GDC 4 [51] requires that SSCs important to safety be designed to defined environmental conditions of normal operation, maintenance, testing, and postulated accidents, including LOCAs. Additionally, Holtec SMR-300 design applies the Leak Before Break (LBB) methodology to eliminate the dynamic effects of certain pipe ruptures.

GDC 10 [51] requires that RCS systems be designed with appropriate margin to assure that SAFDLs are not exceeded during any condition of normal operation, transients, or postulated accidents.

Through meeting the above GDC, confidence is provided that the RIS does not impede coolant flow path or negatively impact the transfer of heat through geometry during normal operation, transients, or postulated design basis accidents. Detailed information on RIS design is presented in [22] and [50].

2.7.5 Demonstration of Confinement

2.2.3.6 -A1: Criteria relevant to demonstrating the integrity of the fuel clad are satisfied for normal operations and AOO.

Evidence for the Argument 2.2.3.6 -A1 is:

- HI-2241327-R1, Holtec SMR-300 GDA Fuel Design Criteria and Limits [24]: This document presents requirements and acceptance criteria associated with the design of the fuel assemblies, RCCAs, and NSAs to be used within the SMR-300 core.
- HI-2241288-R1, Thermal and Mechanical Design Basis [23]: This document presents the methodology and scope for demonstrating the integrity of the fuel assembly and core components under both mechanical and thermal loads. It introduces the key thermal and mechanical design limits, and assessment methodologies. Additionally, the conclusions drawn from a subset of casualty analysis are presented to provide confidence that there are no fundamental shortfalls in the design of the SMR-300 fuel and core system.
- Framatome, Applicability of Framatome Fuel Methods for Holtec Fuel, ANP-4139NP, Revision 0, 2025 [30]: This document presents the applicability of fuel vendor methodologies to the SMR-300. The review involved the evaluation of certain Topical Reports relating to the assessment of M5® fuel rods, submitted to the USNRC, by the fuel vendor, for licensing GAIA fuel assemblies for use in Westinghouse PWRs in the US.

This claim and argument are supported by evidence presented in sub-chapter 2.7.2.3 in relation to fuel rod performance, where is concluded that fuel rods within the SMR-300 core will remain intact for normal operations and AOOs. Additional evidence in support of this argument is presented in Sub-chapters:

- Sub-chapter 2.7.2.1 Nuclear Design Assessment, where it is shown that the limit for peak fuel rod average burnup (non-Gd fuel) is satisfied and the corresponding limit for Gd rods is likely to be satisfied following further optimisation of the fuel management scheme, and that that local power peaking factors satisfy specified limits such that there will be no melting of fuel clad or fuel pellets.
- Sub-chapter 2.7.2.2 Thermal-Hydraulic Design Assessment, where it is shown that effective heat transfer from the fuel to the reactor coolant occurs within the SMR-300 core, avoiding any challenge to fuel clad thermal-mechanical limits.
- Sub-chapter 2.7.2.4 Fuel Assembly Mechanical Performance Assessment, where it is shown that fuel assemblies retain a regular geometry such that the integrity of fuel rods is not jeopardised.

2.7.6 CAE Summary

This sub-chapter summarises the arguments and evidence in support of Claims 2.2.12.2, 2.2.1.2, 2.2.3.6, and 2.2.2.4.

Claim 2.2.12.2: The safety case requirements for the fuel and core components are demonstrated to be satisfied.

2.2.12.2 -A1: Sound methods for assessing fuel and core components against the safety case requirements are in the process of being defined.

Methodologies for performing nuclear design analysis are based upon recommendations of the vendor of the nuclear design software tools used for that analysis [24].

Methods for assessing the thermal-hydraulics characteristics of the SMR-300 reactor are presented and justified in [23]. These methods are consistent with internationally recognised best practice guidance produced by the OECD. The use of COBRA-FLX for thermal-hydraulic assessments is supported by an applicability review of that software by the fuel vendor.

Fuel vendor methodologies for assessing thermal-mechanical performance of fuel rods and fuel assembly components have been assessed and shown to be acceptable [23].

Fuel vendor methodologies for assessing RCCA and NSA performance are judged to be acceptable given the extensive use of those components within PWRs internationally.

2.2.12.2 -A2: Assessments performed thus far demonstrate the safety case requirements acceptance criteria are satisfied or are likely to be satisfied post Step 2 of GDA.

Assessments performed within the Nuclear Design, thermal-hydraulics, Fuel Rod Performance and Fuel Assembly Performance technical areas are presented in Sub-chapters 2.7.1.1, 2.7.1.2, 2.7.1.3 and 2.7.1.4, respectively. These assessments indicate that respective criteria listed in [24] are either shown to be satisfied or are likely to be satisfied for GAIA fuel assemblies used in the SMR-300 reactor. Certain DBA assessments are summarised in Sub-chapter 2.7.2.6, where it is shown that the limit for MDNBR is satisfied for the fault events which have been assessed.

Peak fuel rod average burnup for Gd-doped fuel rods is found to exceed the specified limit for this parameter. However, it is expected that positive margin to this particular limit will be produced following further optimisation of the core design (see GDA Commitment C_Fuel_128).

The assessment results are broadly supported by the positive OPEX of GAIA fuel assemblies used in PWRs.

It is therefore judged that Claim 2.2.12.2 is demonstrated to a level of maturity appropriate for PSR. This judgement is supported by operational experience concerning the use of GAIA fuel assemblies within PWRs.

Claim 2.2.1.2: Control rods provide appropriate reactivity control during normal reactor operation and the means for reactor shutdown during a trip.

2.2.1.2 -A1: Appropriate design of the RCCA rod allows for control of reactivity in the core.

2.2.1.2 -A2: Regulating groups of RCCAs can be stepped in or out of the core to control reactivity during normal operation.

2.2.1.2 -A3: RCCAs are fully inserted when a reactor trip occurs (insertion by gravity).

Sub-chapters 2.7.1.1, 2.7.1.3, 2.7.1.4 and 2.7.2.5 for nuclear design, fuel rod performance, fuel assembly mechanical performance, RCCA integrity outline how core reactivity is controlled for normal operations, AOOs and DBA events. At this stage of plant design, CRD design has not been finalised and so no assessment of RCCA performance has yet been performed, although it is judged that RCCAs will perform as intended given they are of a standard design which has been used extensively in PWRs.

An understanding of the safety functions of the control rod system including supporting sub-systems (CDS, CRD, CRC, RPI) has been developed. As well as how instrumentation interfaces with reactor protection systems to ensure the plant operations remain safe. Once RCCA design has been finalised, demonstration that Claim 2.2.1.2 and its supporting arguments are satisfied will be explicitly confirmed by assessing each RCCA criterion individually for bounding normal operations, AOO and DBA events. This will be conducted post GDA.

It is therefore judged that Claim 2.2.1.2 is demonstrated to a level of maturity appropriate for PSR.

Claim 2.2.2.4: The core is always maintained in a coolable geometry

2.2.2.4 -A1: Fuel assemblies maintain a regular geometry for all design basis events.

2.2.2.4 -A2: Non-fuel reactor internal structures retain a regular geometry so as not to impede the required transfer of heat from the fuel.

Sub-chapters 2.7.2.4 and 2.7.4 outline the demonstration of how fuel assemblies and non-fuel reactor internal structure retain a regular geometry for all plant operating conditions (normal operation, AOO and DBA events).

Evidence in Sub-chapter 2.7.2.4 concerning fuel assembly mechanical performance indicates that the relevant criteria listed in [24] concerning the structural integrity of fuel assemblies will be satisfied.

Sub-chapters 2.2.2 and 2.4 discusses RIS and the criteria that underpins the design. By satisfying these criteria and adhering to codes and standards outlines in Sub-chapter 2.4, it is judged that non-fuel reactor internal structures (within GDA scope) will not impede heat transfer.

It is therefore judged that Claim 2.2.2.4 is demonstrated to a level of maturity appropriate for PSR.

Claim 2.2.3.6: The fuel rod clad integrity is maintained during normal operation and AOOs.

2.2.3.6 -A1: Criteria relevant to demonstrating the integrity of the fuel clad are satisfied for normal operations and AOO.

Evidence in Sub-chapter 2.7.1.3 concerning fuel rod performance indicates that the fuel rod criteria listed in [24] for normal operation and AOOs will be satisfied for GAIA fuel assemblies used in the SMR-300 reactor.

It is therefore judged that Claim 2.2.3.6 is demonstrated to a level of maturity appropriate for PSR. This judgement is supported by operational experience for the GAIA fuel assembly, which to date has not experienced any fuel clad failures.

2.8 CHAPTER SUMMARY AND CONTRIBUTION TO ALARP

This sub-chapter provides an overall summary and conclusion of the Reactor Chapter and how this chapter contributes to the overall demonstration of ALARP for the generic SMR-300. Part A Chapter 5 [18] sets out the overall approach for demonstration of ALARP and how contributions from individual chapters are consolidated.

This sub-chapter therefore consists of the following elements:

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

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

2.8.1 Technical Summary

Part B Chapter 2 contributes to demonstrating 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 decomposed into two Level 4 claims with only one (Claim 2.2.1.2), addressed in this chapter (in Sub-chapter 2.7.3). Claim 2.2.1.2 is addressed by showing how nuclear design, fuel rod performance, fuel assembly mechanical performance, RCCA mechanical performance and RCCA control systems provide reactivity control.

Further work is required to finalise control rod design, including but not limited to, RCCA and CRC design. Once the design is finalised post GDA, RCCA performance assessments will be conducted to ensure that core reactivity can be adequately controlled for normal operations, AOO and DBA events which will ensure the above claim is fully met.

Therefore, it is assessed that this Level 4 claim is met to a maturity expected for PSR, with a full understanding of work to be conducted post Step 2. The remaining work to substantiate this claim has been assessed as normal business within design development, therefore no GDA commitment has been raised.

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 decomposed into four Level 4 claims with only one (Claim 2.2.2.4) addressed in this chapter (in Sub-chapter 2.7.4). Claim 2.2.2.4 is addressed by demonstrating both the fuel, core and reactor internal structures retain their designed geometry so as not to impede heat transfer from the fuel to the reactor coolant. This will be demonstrated by conducting mechanical performance analysis for these technical areas. At this stage, no analysis has been conducted. However, it has been demonstrated in this chapter that fuel assemblies and

the reactor core has been designed to internationally recognised codes and standards. Furthermore, the fuel vendor has conducted comparisons between SMR-300 reactor internal conditions and other operating PWRs in the US for which the GAIA fuel assembly is licenced for use. The similarities between the operating conditions provide confidence that the current design of fuel assembly is fit for purpose with regards to heat removal.

Further work is required to conduct bespoke mechanical performance analysis on both the fuel assembly and reactor internals to ensure the necessary performance criteria are met. Despite this work being scheduled post GDA, utilising codes, standards and fuel vendor report provides assurances that the core will retain a coolable geometry.

Therefore, it is assessed that this Level 4 claim is met to a maturity expected for PSR with a full understanding of work to be conducted post Step 2. Post GDA, mechanical performance analysis will be conducted for the GAIA fuel assembly and reactor internal structures which will be used to substantiate this claim. The remaining work to demonstrate this claim has been assessed as normal business within design development, no GDA commitment has been raised.

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 is decomposed into six Level 4 claims with only one claim (Claim 2.2.3.6) addressed in this chapter (in Sub-chapter 2.7.5). Claim 2.2.3.6 is addressed by demonstrating the fuel clad remains intact for normal operations and AOOs. The number of fuel rod failures occurring as a result of a DBA are to be ALARP and shall not result in dose limits being exceeded.

The fuel vendor has conducted comparisons between SMR-300 reactor internal conditions and other operating PWRs in the US for which the GAIA fuel assembly is licenced for use. The similarities between the operating conditions provide confidence that the current design of fuel rods is fit for purpose with regards to use within the SMR-300. This judgement is supported by operational experience for the GAIA fuel assembly, which to date has not experienced any fuel clad failures. Therefore, there is confidence that the choice of fuel and associated cladding is both appropriate for the SMR-300 and safe with regards to release of radioactive material.

Therefore, it is assessed that this Level 4 claim is met to a maturity expected for PSR with a full understanding of work to be conducted post Step 2. Post GDA, fuel performance analysis will be conducted for the GAIA fuel assembly in the SMR-300 plant which will be used substantiate this claim. As the remaining work to demonstrate this claim has been assessed as normal business within design development, no GDA commitment has been raised.

Claim 2.2.12: The objective of preventing damage to the fuel and core components is appropriately accounted for within their design and safety function.

Claim 2.2.12 is decomposed into two Level 4 claims (Claims 2.2.12.1 and 2.2.12.2), both of which are discussed within this chapter (in Sub-chapters 2.6 and 2.7). The aim of these two Level 4 claims is to demonstrate that a safe envelope of operation for fuel and core components has been established and that the safety case requirements associated with the safe envelope have been satisfied.

It is judged that the set of safety case requirements and their associated acceptance criteria for fuel and core have been established to a level of maturity that is consistent with the objectives of the PSR. Assessments against the derived criteria are incomplete and work to validate methodologies in the nuclear design and thermal-hydraulics technical area will continue following GDA.

Post GDA, limits and criteria for all SSCs will be finalised and demonstrated as satisfied by the design. This substantiation will be undertaken at the Pre-Construction Safety Report (PCSR) stage. It is therefore assessed that both Level 4 claims are met to a maturity expected at PSR for those elements within the GDA scope.

2.8.2 ALARP Summary

A high-level summary of the options considered as part of the SMR-300 reactor core design can be seen in the Overview of Holtec SMR-300 Fuel Design and Core Components document [22].

2.8.2.1 Demonstration of RGP

2.8.2.1.1 Design Codes and Standards

The design of the SMR-300 reactor internal structures complies with RGP and USNRC requirements applicable in the US. The design applies nuclear-specific codes and standards endorsed by the USNRC and internationally recognised bodies such as IAEA.

The principal codes and standards identified within Sub-chapter 2.4 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.

A review of the design of the SMR-300 reactor against each clause in IAEA SSG-52 (Design of the Reactor Core for Nuclear Power Plants) has been performed. This has been undertaken in order to identify any potentially significant shortfalls in design, core design limits, core substantiation, quality control etc.

This review demonstrates that the SMR-300 core is generally compliant with most of the clauses within SSG-52, although the following important points are noted:

- Clauses 3.110 and 3.111 in SSG-52 relate to diverse means of shutdown and reference is made to the following text within SSR-2/1: “The means for shutting down the reactor shall consist of at least two diverse and independent systems.”

The design of the diverse means of shutdown for the SMR-300 is an ongoing topic that extends beyond the physical scope of the Reactor topic. [REDACTED]. Work is underway to assess whether the design of this diverse system is consistent with the ALARP principle. This topic is being primarily tracked via Part B Chapter 1 with a signpost in Sub-chapter 2.8.2.3. 2.8.2.3

The review of the SMR-300 reactor design against SSG-52 highlighted further clauses which are not yet met but which are expected to be met as the design is developed further. Further discussion of these points can be found in [35].

2.8.2.1.2 Criteria and Methodologies for Fuel and Core

The applied criteria for fuel and core are consistent with similar sets of criteria applied to other PWRs that are either planned or operating in the UK, Europe and the US.

Methodologies for assessing fuel and core against nuclear design requirements are being developed jointly between the reactor physics software vendor and Holtec International, in accordance with the vendor's software manual. These methodologies are based on vendor experience of applying its reactor physics code to other operating PWRs.

Methodologies applied for assessing fuel rod design performance and fuel assembly mechanical design performance have been developed and applied by the fuel vendor to other PWRs. Similarly, methodologies for assessing fuel and core against thermal-hydraulics criteria are being developed jointly by the fuel vendor and Holtec and are based on prior fuel vendor experience in performing such assessments for other operating PWRs.

The software packages being used by Holtec for assessing the criteria listed in [24] are used widely within the nuclear industry internationally; none of the applied software tools are novel or unique to the SMR-300 project.

2.8.2.2 Evaluation of Risk and Demonstration Against Risk Targets (where applicable)

The numerical targets against which the demonstration of ALARP is considered can be found in PSR Part A Chapter 2 [7].

The Reactor topic, through the defined safety functions, will contribute to the demonstration of ALARP by comparison against the risk targets in the following ways:

- Safety case requirements and associated criteria have been defined to ensure fuel clad remains intact for normal operations and AOOs, noting the caveat raised in Sub-chapter 2.8.2.3.1 concerning scope of fuel clad integrity assessments. Criteria have also been defined for Deterministic Safety Analysis (DSA) for the purpose of demonstrating the number of fuel clad failures occurring as a result of design basis accidents are ALARP.
- Safety case requirements and associated criteria have been defined for reactivity control, heat management and confinement of activity. As the design matures, requirements and criteria will be established for all SSCs which contribute to these SSCs.

Evaluation of risk is not directly applicable to the Reactor topic. The safety classification of the reactor SSCs 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.

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

2.8.2.3 Options Considered to Reduce Risk

This sub-chapter briefly summarises the margins, judgements and challenges associated with the Reactor topic at this stage of the development of the safety case for the SMR-300. The sub-chapter will discuss issues and challenges that relate to both the project and plant design.

For design associated risks, the process for the assessment of risk reduction options is presented in the Design Management document [49].

Part A Chapter 5 ALARP Summary [18] considers the holistic risk-reduction process for the generic SMR-300.

2.8.2.3.1 Safety Case Objectives and Strategy

Presently, the fuel and core criteria are based upon US practice of demonstrating fuel clad integrity for normal operations and AOO. By contrast, the regulatory expectation within the UK is that fuel clad integrity shall be demonstrated for all postulated frequent fault and hazard events and that fuel failures resulting from infrequent events shall be ALARP. Hence there is a region of fault event parameter space between 10^{-2} per year and 10^{-3} per year for which no assessments are currently planned which have the purpose of demonstrating fuel clad integrity [48]. Furthermore, the development of the fault schedule and the transient analysis (fault analysis) is insufficiently mature to demonstrate that fuel rod failures occurring during infrequent events are reduced to ALARP.

In the case of frequent events, the current shortfall could be resolved by extending the methodologies for assessing AOOs to all faults having an initiating event frequency $\geq 10^{-3}$ per year. Other, potentially more efficient, options for demonstrating clad integrity for all frequent events are described in [48]. In the case of infrequent events, demonstration that the number of fuel failures are ALARP will not be possible until after Step 2 of GDA.

For these reasons, a GDA commitment (C_Fuel_127) has been raised to ensure these shortfalls within the safety case strategy are suitably resolved following Step 2 of GDA (see Sub-chapter 2.8.3). Furthermore, a GDA commitment (C_Faul_103) has been made in Part B Chapter 14 [13] to undertake full Design Basis Accident Analysis (DBAA), from which faults in the reactor topic area will be identified and confirmed.

2.8.2.3.2 Applied Fuel and Core Criteria

It is judged that the list of safety case requirements relating to fuel, RCCAs and neutron sources is complete, as presented in [24] and that the stated applicability of the requirements is accurate. However, further development of some criteria is required following Step 2 of GDA, e.g. to specify specific ASME service levels to be applied to plant conditions.

Criteria relating to power peaking factors have been derived via iteration and convergence between nuclear design analysis and transient analysis. The majority of the remaining criteria are based upon publicly available fuel vendor information, widely applied design codes and widely applied industry practice.

Criteria for a minority of parameters have yet to be determined, although these cases are judged to be not significant at this stage of the project mainly because they relate either to dry storage (currently out of scope of Part B Chapter 2), or to infrequent fault conditions. These will be derived as part of normal future development of the safety analysis.

In the case of thermal-hydraulics analysis, bounding initial conditions are applied to inputs such as power peaking, coolant flow, circuit pressure, flow maldistribution etc. These constitute a set of 'interface parameters' to be satisfied in order for safety assessments to be considered valid. Furthermore, penalties are applied within the assessment of DNB to account for the possibility of fuel rod bow and fuel assembly bow. This approach means the assessments are suitably conservative of actual operating core conditions.

Assessment of the remaining criteria is considered normal business as the safety case develops. Therefore, no GDA commitment has been raised.

2.8.2.3.3 Nuclear Design Analysis

The majority of computational analysis supporting this iteration of the PSR has occurred in the Nuclear Design and thermal-hydraulic design technical areas. In these cases, certain parameters have been assessed against criteria presented in [24].

Nuclear design analysis has been performed on a mostly best-estimate basis because no formal validation of numerical methods has yet been performed that is specific to the SMR-300, hence uncertainties specific to the SMR-300 are generally not yet available. The main exception to this is the calculation of shutdown margin, for which applied uncertainties have been estimated and are judged to be reasonable. Certain parameters have intentionally been assessed on a best-estimate basis, such as fuel assembly average burnup and fuel rod average burnup.

It is shown in [25] that the 'generic' PWR validation base for CASMO-SIMULATE bounds the design and operating parameters of the SMR-300 core, including its aspect-ratio and use of a heavy reflector. Hence it is expected that uncertainties derived specifically for the SMR-300 will be bounded by (or otherwise be consistent with) those derived on a generic basis by the code vendor for PWR applications.

In cases of best-estimate analysis for which the required application of uncertainties is currently lacking, the calculated margins to applied limits are judged to be sufficiently large that future applied uncertainties are unlikely to lead to current conclusions being challenged. Nonetheless, nuclear design validation of methodologies, including confirmation of applied uncertainties, has been identified as an area of further work to be conducted for the reactor topic.

The proposed approach to be taken for addressing this further work is presented in [25]. In summary, the resolution of this issue will involve:

- Confirming the validation base for CASMO-SIMULATE, as presented in the Topical Report for this code package, is applicable to the operation of the SMR-300 core in terms of core design, calculated parameters, heavy reflector etc.
- [REDACTED]
- [REDACTED]
- [REDACTED]

In the case of calculated shutdown margin, the calculated value of sub-criticality greatly exceeds the applied limit, meaning that any future changes to current estimates of uncertainty within the methodology are very unlikely to challenge current conclusions drawn from those assessments.

The only case within the current set of nuclear design assessments for which an applied criterion is exceeded concerns the calculation of peak fuel rod average burnup for Gd-doped fuel rods. This result is considered to be sufficiently important that a GDA Commitment (C_Fuel_128) has been raised to address this current shortfall (see Sub-chapter 2.8.3). Nuclear design assessment of fuel rod average burnup shall be revisited with the aim of restoring positive margin to the limit. Options for addressing the commitment are set out in [25].

2.8.2.3.4 Core Design and Batching

The core designs for [REDACTED] involve shuffling from one cycle to the next. This approach has been found to lead to good margins to design limits, although fuel assemblies shuffled in this manner laid out in [25] can lead to issues at end-of-life. Applying alternative methods of shuffling could potentially “flatten” fuel assembly burnup tilts and reduce the peak rod average burnup.

Thus far, preliminary core design sensitivity studies indicate that the fresh fuel batch can be reduced from 29 fuel assemblies to [REDACTED] fuel assemblies on the basis of assessing a realistic capacity factor <100%. This in turn would have positive implications for fuel utilisation and environmental considerations. Further optimisation of the core design will be explored following GDA.

2.8.2.3.5 Fuel Rod, Fuel Assembly, RCCA and Neutron Source Analysis

No analysis has been presented for fuel rod performance, fuel assembly skeleton mechanical performance, RCCA performance and neutron source performance at Step 2 of GDA.

It is judged that future assessments of fuel rod performance and fuel assembly skeleton mechanical performance, for normal operations and AOO events, are likely to result in acceptable outcomes, i.e. the relevant requirements within [24] will be satisfied. This judgement is based on the outcome of a fuel vendor review of the applicability of its Topical Reports to the SMR-300 design [30].

No assessment of RCCA performance has been made for Step 2 of GDA, for instance, in relation to mechanical loads arising from rod stepping, rod drop and seismic events. The CRDMs and RCCAs are of a standard Framatome design that are used in other reactors for which Framatome has demonstrated acceptable margins to applied criteria. Therefore, it is judged that RCCA performance will be shown to acceptable in future assessments.

Similarly, it is likely that neutron source assemblies will be of a standard design that is used in other PWRs. It is therefore judged that the performance of neutron source assemblies will likewise be shown to be acceptable, with positive margins to limits.

This analysis is considered normal business as the safety case develops. Therefore, no GDA commitment has been raised.

2.8.2.3.6 Change in RCCA Design

Preliminary analysis of rod ejection accidents conducted by the fuel vendor identified that a design with 33 RCCAs resulted in a high rod worth which exceeded the fuel vendors limits. A re-design of RCCA layout identified a solution with 29 RCCAs and highest rod worth being within acceptable limits for rod ejection accident criteria.

The new design reduced the overall number of RCCAs as well as increasing the percentage of shutdown banks and removing the shadowing effect from corner RCCAs on ex-core detectors. Further discussion is provided in [25].

2.8.2.3.7 Diverse Means of Shutdown

The previous iteration of the PSR did not fully articulate the claimed diverse means of shutdown for the SMR-300. This situation has been improved by the production of a report outlining the options available for the SMR-300 [53]. In parallel with that report, a Design Challenge Paper has been raised within Holtec Britain which identifies the possible project risk associated with the diverse shutdown mechanism [54]. This topic is discussed further in Part B Chapter 1 [16].

2.8.2.3.8 Thermal-Hydraulics Analysis

The main challenge perceived within the thermal-hydraulic technical area concerns validation of reactor coolant flow modelling against actual test rig data, especially in regard to the asymmetric reactor inlet nozzles and use of a flow distribution device. A certain amount of uncertainty exists surrounding how validation will be achieved, and several options are currently under consideration, including: making use of existing Framatome test rigs, constructing a scaled or full-sized reactor test rig which is bespoke to the SMR-300, or benchmarking analysis against measurements taken from the first operating SMR-300 (assumed to be at Palisades US), or some combination of these options. Given such uncertainty and the importance of these novel aspects of the SMR-300 design, further work will be required to address this current uncertainty (see Sub-chapter 2.8.3).

Only steady-state thermal-hydraulic analysis has been formally reported to date. In general, the thermal-hydraulic assessments reported in [23], such as the assessments of DNBR, show significant margins to limits based on analysis performed to date, indicating that further measures optimisation of these aspects is unlikely to be necessary. However, no transient analysis has been conducted.

Transient analysis is important for determining and understanding variations in reactor coolant flow conditions that occur on short timescales that are of relevance to other reactor internal components (e.g. vibration and resonance analysis). However, such transient calculations can be complex, computationally expensive, and require validation against data from physical tests or benchmarking exercises before they can be accepted for use in the safety assessments. Given the importance of such calculations in the context of the thermal-hydraulic design of the SMR-300 design, further work will be required to address this current uncertainty (see Sub-chapter 2.8.3). This further work also includes a requirement to perform appropriate assessment of variations in boundary conditions (i.e. sensitivity studies), such as differences in inlet flow speeds.

The approach taken to address the further work above, including the validation of codes and methodologies, and derivation of applied uncertainties will cover the following:

- Obtaining pertinent data from test rig experiments. Options for performing test rig experiments are presented in [23].
- Performing time-dependent thermal-hydraulics analysis.
- Performing sensitivity studies to changes to input boundary conditions, such as pump speeds and differences in inlet flow between the two reactor inlet nozzles.

2.8.2.3.9 Inlet Nozzle Location and FDD Design Evolution

Optioneering studies leading to the selection of asymmetric reactor inlet nozzle location and design of the FDD are considered to be significant contributors to the overall ALARP argument for the SMR-300, that arise from the thermal-hydraulics discipline. These aspects of the reactor design are therefore described further in [23].

The design objectives for the SMR-300 include an intent to minimise the number of components, including for the RCS. This leads to the overall chance of RCS component failure for the 2-loop SMR-300 being nominally less than that for 3-loop or 4-loop PWRs because it has fewer pipe bends, pipe connections, pumps etc. This benefit, however, results in a need to route the hot and cold legs in such a way which leads to asymmetric flow into the reactor core. This in itself is not entirely unusual because other operating PWRs have reactor inlets (and outlets) which are not positioned equidistant from each other about the circumference of the reactor vessel. Nonetheless, the flow maldistribution arising in the SMR-300 in the absence of a FDD was considered to be unacceptably large, hence the FDD was included in the design to homogenise the core inlet flow.

The results presented in [23] show that the use of a FDD reduces flow maldistribution significantly, although it is possible that further minor optimisation of the design will continue beyond Step 2 of GDA.

2.8.3 GDA Commitments

GDA Commitments have been formally captured in the Commitments, Assumptions and Requirements process [55]. Further details of this process are provided in Part A Chapter 4 Lifecycle Management [56].

Below is an overview of the GDA Commitments raised from this chapter:

C_Fuel_127: Fuel clad integrity of the SMR-300 will be maintained during normal operation and Anticipated Operational Occurrences (AOOs) (Claim 2.2.3.6). Differences in UK and US regulatory expectations require that for the UK deployment of the SMR-300, additional analysis work be undertaken to demonstrate with a high degree of confidence that frequent design basis faults (frequency > 10^{-3} /yr) do not result in breaches of the fuel cladding. This demonstration is beyond that required for the US, which is limited to an initiating event frequency (IEF) of > 10^{-2} /yr. A Commitment is raised to identify the initiating events with $1\times10^{-2} > \text{IEF} \geq 1\times10^{-3}$, (known as DBA in the US, or DBC3a in the UK) and to evaluate their impact on fuel rod cladding integrity, ensuring that risks are minimised SFAIRP. Target for Resolution – Issue of Pre-Construction SSEC.

C_Fuel_128: Core design and optimisation work undertaken for the SMR-300 has identified that the Gd-doped fuel rod average burnup exceeds the limit applied by Holtec (fuel vendor limit) if operated conservatively. A Commitment is raised to progress to completion the optioneering undertaken to address this topic in accordance with the Design Management process (HPP-3295-0017-R1.0). Target for Resolution - Issue of Pre-Construction SSEC.

C_Fuel_129: Claim 2.2.12.2 is based upon the requirements presented in HI-2241327-R1 'Holtec SMR-300 GDA Fuel Design Criteria and Limits' and is partially demonstrated for PSR v1. Additional justification is required regarding thermal-hydraulics and fuel mechanical performance in relation to coolant flow stability. A Commitment is raised to extend the current substantiation to include time-dependent modelling of coolant flow and sensitivity studies, in relation to changes to inputs and boundary conditions, validated against measurement data derived from physical tests. Target for Resolution - Issue of Pre-Construction SSEC.

Below is an overview of the further work raised in this chapter.

- Nuclear design methodologies and assessments, including the confirmation of uncertainties, is to be conducted with a validated computer code.
- Validation of reactor coolant flow modelling against actual test rig data is to be conducted including the impact of asymmetric reactor inlet nozzles and use of a flow distribution device.
- Time-dependent transient analysis of reactor coolant flow is to be conducted, including assessment of variations in boundary conditions, such as differences in inlet flow speeds.

2.8.4 Conclusion

This chapter summarises the high-level design and supporting analysis of the fuel and core topic area. It identifies the claims and arguments that will form the basis of the safety case throughout the lifecycle of the SMR-300 to a maturity aligned to a PSR. As the design and safety case develop, further evidence and analysis will be provided to further substantiate the claims.

With the exception of Claim 2.2.12, Level 3 claims discussed within this chapter are also supported by other PSR chapters as well. As outlined in the Technical Summary, all claims have been satisfied to a level of maturity consistent with the objectives of the PSR. A clear understanding of remaining work has been developed to ensure Claim 2.2.1.2 can be met post GDA Step 2.

GDA Commitments have been raised to capture differences between UK and US requirements and regulations or where analysis or design is not to a maturity level expected for PSR.

REFERENCES

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

[2] Holtec Britain, “HI-2240334, SMR-300 GDA PSR Part A Chapter 3 Claims, Arguments and Evidence,” Revision 1, July 2025.

[3] Holtec International, “HI-2240845, System Design Description for the Control Rod Control System,” Revision 0, 2024.

[4] Holtec International, “HI-2240847, System Design Description for Ex-Core Instrumentation System,” Revision 0, 2025.

[5] Holtec International, “HI-2240848, System Design Description for In-core Instrumentation System, Rev 0,” 2025.

[6] Holtec Britain, “HI-2240353, Holtec SMR GDA PSR PART B Chapter 24 Fuel Transport and Storage,” Revision 1, 2025.

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

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

[9] Holtec Britain, “HI-2240339, Holtec SMR GDA PSR PART B Chapter 6 Electrical Engineering,” Revision 1, 2025.

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

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

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

[13] Holtec Britain, “HI-2240345, Holtec SMR GDA PSR PART B Chapter 14 Safety and Design Basis Accident Analysis,” Revision 1, 2025.

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

[15] Holtec Britain, “HI-2240347, Holtec SMR GDA PSR PART B Chapter 16 Probabilistic Safety Assessment,” Revision 1, 2025.

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

[17] Holtec Britain, “HI-2240777, Holtec SMR GDA PSR PART B Chapter 5 Description of the Reactor Supporting Facilities,” Revision 1, 2025.

[18] Holtec Britain, “HI-2240336, Holtec SMR GDA PSR Part A Chapter 5 Summary of ALARP,” Revision 1, 2025.

[19] Holtec Britain, “HI-2240516, Holtec SMR-300 GDA Approach and Application to the Demonstration of BAT,” Revision 1, 2025.

[20] Holtec Britain, “HI-2240121, Holtec SMR GDA Scope,” Revision 1, 2025.

[21] Holtec Britain, “HI-2240342, Holtec SMR GDA PSR PART B Chapter 11 Environmental Protection,” Revision 1, 2025.

[22] Holtec Britain, “HI-2240861, Overview of Holtec SMR-300 Fuel Design and Core Components,” Revision 1, 2025.

[23] Holtec Britain, “HI-2241288, Thermal and Mechanical Design Basis,” Revision 1, 2025.

[24] Holtec Britain, “HI-2241327, Holtec SMR-300 GDA Fuel Design Criteria and Limits,” Revision 1, 2025.

[25] Holtec Britain, “HI-2241458, Holtec SMR-300 GDA Nuclear Design Basis Report,” Revision 1, 2025.

[26] Holtec International, “HI-2240077, SMR-300 Plant Overview Rev. 1,” 2024.

[27] Holtec International, “HI-2250362, SMR-300 Plant Operation Overview,” Revision 0, 2025.

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

[29] U.S. Nuclear Regulatory Commission, “NUREG-0800, Standard Review Plan, 4.2 Fuel System Design,” Revision 3, 2007.

[30] Framatome, “Applicability of Framatome Fuel Methods for Holtec Fuel, ANP-4139NP, Revision 0,” 2025.

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

[32] US Nuclear Regulatory Commission, “Regulatory Guide 1.26, Quality Group Classification and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants, Rev. 6,” 2021.

[33] Holtec Britain, “HI-2241408, Mechanical Engineering Codes and Standards,” Revision 0, 2024.

[34] International Atomic Energy Agency, “Specific Safety Guide. SSG-52, Design of the Reactor Core for Nuclear Power Plants,” 2019.

[35] Holtec Britain, “HI-2250562, SMR-300 SSG-52 Compliance,” Revision 0, 2025.

[36] International Atomic Energy Agency, “Defence in Depth in Nuclear Safety, INSAG-10,” 1996.

[37] Holtec Britain, “HI-2250348, Containment Structure System Based View,” Revision 0, 2025.

[38] Holtec International, “HI-2250400, Outage Strategy for SMR-300,” Revision 0, 2025.

[39] Office for Nuclear Regulation, “ONR-OPF-PAR-16-010 Revision 0, Sizewell B Dry Fuel Store,” 2016.

[40] Holtec International, “HI-2240459, SMR-300 Primary Strategic Water Chemistry Plan,” Revision 0, 2024.

[41] Holtec Britain, “HI-2241322, Preliminary Fault Schedule Revision 1,” 2025.

[42] Framatome, “M5® Material Topical Report (BAW-10227PA).”.

[43] Framatome, “BAW-10231P, COPERNIC Fuel Rod Design Topical Report,” Revision 1.

[44] Framatome, “Program to Determine In-reactor Performance of BWFC Fuel Cladding Creep Collapse (BAW-10084)”.

[45] Framatome, “GAIA Mechanical Fuel Design Topical Report (ANP-10342P-A)”.

[46] Framatome, “Q12 Structural Material Topical Report (ANP-10334P-A)”.

[47] Framatome, “Seismic Topical Report (ANP-10337P)”.

[48] Holtec Britain, “HI-2250581, RQ-01566 Progress Statement at GDA Step 2,” 2025.

[49] Holtec Britain, “HPP-3295-0017, Design Management,” Revision 1, 2025.

[50] Holtec Internationa1, “HI-2240961, Holtec SMR-300 Reactor Pressure Vessel Internals Design Description,” 2024.

[51] US Nuclear Regulatory Commission, “Title 10, Code of Federal Regulations, 10 CFR 50, App. A, General Design Criteria for Nuclear Power Plants”.

[52] U.S. Nuclear Regulatory Commission, “10 CFR 50.55a, Codes and standards”.

[53] Holtec Britain, “HI-2250569, UK GDA SMR-300 Secondary Shutdown Mechanism Options Review,” Revision 0, 2025.

[54] Holtec Britain, “HI-2241431, Design Challenge - Diverse Means of Shutdown,” 2025.

[55] Holtec Britain, “HPP-3295-0013, Holtec SMR-300 Generic Design Assessment CAR Process,” Revision 0, 2025.

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

2.9 LIST OF APPENDICES

Appendix A PSR Part B Chapter 2 CAE Route Map

Table 9: PSR Part B Chapter 2 CAE Route Map

[REDACTED]