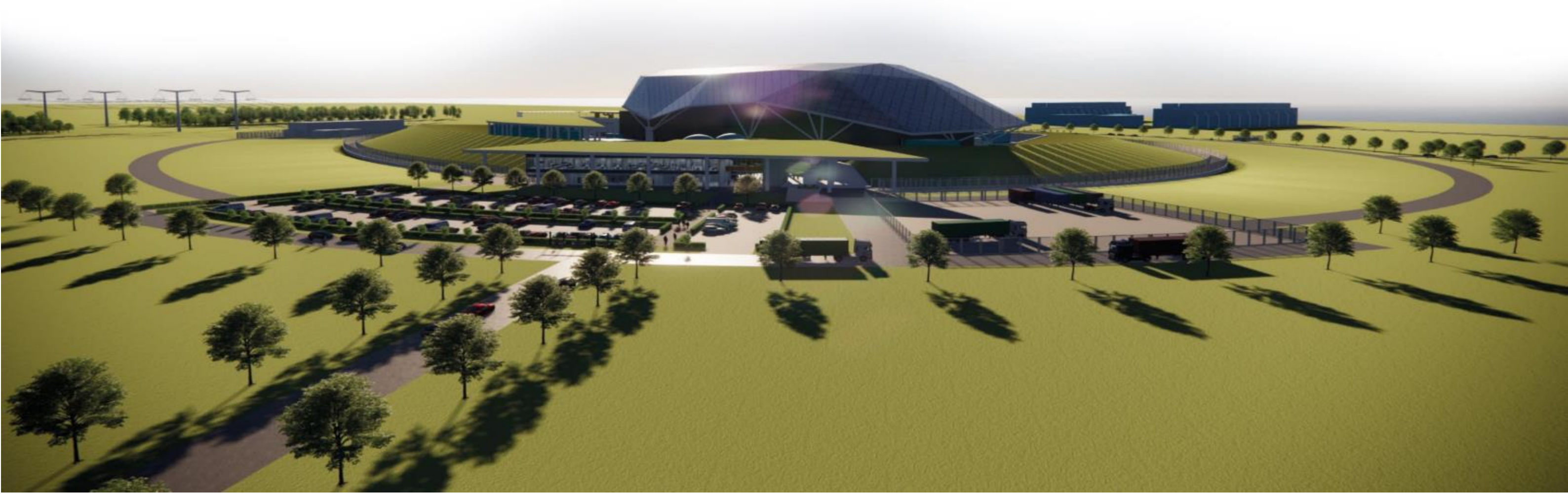
Environment, Safety, Security and Safeguards Case Version 2, Tier 1, Chapter 4: Reactor (Fuel and Core)



Record of Change

Date	Revision Number	Status	Reason for Change
March 2023	1	Issue	First issue of E3S Case
February 2024	2	Issue	Incorporates new design developments and analysis at Reference Design 7, aligned to Design Reference Point 1, including: • Core design optimisation at iteration 7 • Fuel design and analysis • Nuclear design and analysis • Thermal hydraulic design and analysis
May 2024	3	Issue	Updated to correct revision history status at Issue 2. Chapter changes include: • Updated references to include regulatory guidance in section 4.0.4 • Improved cross-referencing to other E3S Case chapters throughout • Updated design basis and design limits for cladding failure, stress, oxidation and hydriding, pellet clad interaction and stress corrosion cracking, in sections 4.2.2.1, 4.2.2.5, 4.2.2.7, and 4.2.2.8, reflecting Tier 2 fuel and core documents • Clarifications to rod cluster control assembly materials added in sections 4.2.4.4 and 4.2.4.5 • New section in 4.2.4.7 added clarifying current position on thimble plugs • Updated design basis and design limits for reactivity coefficients and linear heat generation rate in sections 4.3.1.2 and 4.3.1.5, reflecting Tier 2 fuel and core documents • Updated commentary on fuel temperature doppler coefficient values and uncertainties • Updated section 4.3.3.4 confirming limits on shutdown margin are met • Design limit section added for rod bow in section 4.4.1.6 • Removed section on model boundary conditions in section 4.4.4.11 • Additional detail within conclusions section for how arguments and evidence presented meet the generic E3S case objective Also minor template/editorial updates for overall E3S Case consistency.

Executive Summary

Chapter 4 of the generic Environment, Safety, Security and Safeguards (E3S) Case presents the fuel and core design information for the Rolls-Royce Small Modular Reactor (RR SMR)

The chapter outlines the arguments and evidence to underpin the top-level claim that the reactor (fuel and core) is conservatively designed and verified to deliver E3S functions through-life, in accordance with the E3S design principles, to reduce risks to as low as reasonably practicable (ALARP) with application of best available techniques (BAT), secure by design and safeguards by design.

The fuel and core design are optimised at 'Iteration 7', representing a mature design where the total core layout and design limits are defined. A suite of nuclear, fuel, and thermohydraulic analyses is undertaken to demonstrate that acceptance criteria and design limits can be achieved. Core components are designed in accordance with relevant good practice (RGP) and operating experience (OPEX), with design to codes and standards according to the safety classification, and down-selection of options in accordance with criteria to ensure risks are reduced to ALARP, apply BAT, and are secure by design and safeguards by design.

Version 2 of the generic E3S Case is developed in support of the reference design 7 (RD7) design, corresponding to design reference point 1 (DRP1) for the generic design assessment (GDA). Further arguments and evidence are to be developed to underpin the top-level claim, including ongoing optimisation of the core design to meet the design bases, further iterations of analyses for all modes of operation, development of a complete set of non-functional system requirements for the core components from the E3S design principles, and verification and validation of all E3S requirements.

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4.0 Introduction to Chapter

4.0.1 Introduction

Chapter 4 of the Rolls-Royce Small Modular Reactor (RR SMR) generic Environment, Safety, Security and Safeguards (E3S) Case presents the overarching summary and entry point to the design and E3S information for the fuel and core design of the RR SMR.

4.0.2 Scope and Maturity

The scope of this chapter for fuel and core covers the following physical components:

- Fuel assemblies
- Neutron sources
- Control rods.

The scope covers justification of these core components for equilibrium cores, an initial core load and for any transitional cores, based on a single design concept of an 18 month, three batch cycle. All modes of operation are included. The selection of materials and justification of the integrity of SSCs is covered in E3S Case Version 2, Tier 1, Chapter 23: Structural Integrity [1].

The chapter presents a high-level summary of the core and design requirements (section 4.1), an overview of the fuel design (section 4.2), the nuclear design and analysis (section 4.3), the thermal hydraulic design and analysis (section 4.3), and a description of the mechanical design of the reactor core components (section 4.5).

The following systems and components are excluded from the scope; however, some discussion may be included to provide relevant context:

- Reactor Pressure Vessel (RPV) and internal mechanical structures, covered within E3S Case Version 2, Tier 1, Chapter 5: Reactor Coolant System & Associated Systems [2]
- Control Rod Drive Mechanisms (CRDMs), covered within E3S Case Version 2, Tier 1, Chapter 5: Reactor Coolant System & Associated Systems [2]
- Fuel Handling Systems [F], covered within E3S Case Version 2, Tier 1, Chapter 9A: Auxiliary Systems [3]
- Neutron and Temperature Sensors [JY], covered within E3S Case Version 2, Tier 1, Chapter 7: Instrumentation & Control [4].

The Reactor Core [JAC] design is undergoing constant optimisation to improve performance, increase safety and to reflect the changing requirements of the wider programme. As part of the core design optimisation, minor snapshots of the core design (known as ‘iterations’) are taken to provide a design baseline and to communicate any updated performance characteristics to any interfacing disciplines.

Version 2 of the generic E3S Case is based on reference design 7 (RD7), corresponding to design reference point 1 (DRP1) for the generic design assessment (GDA). RD7 presents the core design at 'Iteration 7', which represents a mature design where the total core layout and design limits are defined. As the detailed design develops, only minor optimisations are expected. A suite of fuel and core analysis is presented for RD7/DRP1, noting this is iterative in nature and so will continue to be updated alongside the fuel and core detailed design.

4.0.3 Claims, Arguments and Evidence Route Map

The overall approach to claims, arguments, evidence (CAE) and the set of fundamental E3S claims to achieve the E3S fundamental objective are described in E3S Case Version 2, Tier 1, Chapter 1: Introduction [5]. The associated top-level chapter claim for E3S Case Version 2, Tier 1, Chapter 4: Reactor (Fuel and Core) is:

Claim 4: The reactor (fuel and core) is conservatively designed and verified to deliver E3S functions through-life, in accordance with the E3S design principles, to reduce risks to ALARP with application of BAT, secure by design and safeguards by design

A decomposition of this claim into sub-claims, and mapping to the relevant Tier 2 and Tier 3 information containing the detailed arguments and evidence, is presented in the E3S Case Route Map [6]. Given the evolving nature of the E3S Case alongside the maturing design, the underpinning arguments and evidence may still be developed in future design stages; the trajectory of this information, where possible, is also illustrated in the route map, which aligns the anticipated arguments and evidence to future issues of the generic E3S Case (subject to ongoing planning).

A proportionate summary of the arguments and evidence from lower tier information, available at the current design stage, is presented within this chapter. A mapping of the claims to the corresponding sections that summarise the arguments and/or evidence is provided in, Appendix A (section 4.8).

4.0.4 Applicable Regulations, Codes and Standards

The mechanical systems and components summarised in this chapter are designed in accordance with their safety classification, to the codes and standards outlined in Table 4.0-1, based on [7].

Table 4.0-1: Mechanical Design Codes and Standards

Safety Classification	Design Basis Code
VHR	American Society of Mechanical Engineers (ASME) III (Sub-section NB) and beyond code requirements
HR	ASME III (Sub-section NB) and beyond code requirements
Class 1	ASME III
Class 2	ASME III
Class 3	ASME III or Commercial standards e.g., ASME VIII, British Standard (BS) and European Standard (EN) BS EN 13445
Not Applicable (n/a)	Commercial standards e.g., ASME VIII, BS EN 13455

ASME III Code Class CS is the appropriate standard for Reactor Core and internals components.

Additional codes, standards and guidance identified for the Reactor Core [JAC] include:

- International Atomic Energy Agency (IAEA) Safety Specific Requirements (SSR) SSR-2/1 (Rev. 1), Safety of Nuclear Power Plants: Design [8]
- IAEA Safety Specific Guide (SSG) SSG-52, Design of the Reactor Core for Nuclear Power Plants [9]
- IAEA SSG-73, Core Management and Fuel Handling for Nuclear Power Plants [10]
- Office for Nuclear Regulation (ONR) Technical Assessment Guide: Safety of Nuclear Fuel in Power Reactors [11]

4.1 Summary Description

4.1.1 General Design Requirements

The fuel and core design are developed in accordance with the systems engineering design process, with design and performance requirements developed based on relevant good practice (RGP) and operating experience (OPEX).

The fuel and core are designed to achieve functional and associated performance requirements, which ensure the Reactor Core [JAC] will operate safely, interface with the wider plant infrastructure and meet the overall objectives of the power station. At the highest level, the design requirements are summarised as:

- Generate heat – the Reactor Core [JAC] is required to generate 1358 MW of thermal energy for a cycle length of 18 months.
- Transfer heat – the Reactor Core [JAC] is required to transfer thermal energy generated through fission to the reactor coolant. A safe margin to the departure from nucleate boiling ratio (DNBR) is required to be maintained through all operational modes including transients and frequent faults.
- Maintain negative fuel and coolant temperature coefficients of reactivity – by ensuring the Reactor Core has negative power coefficients of reactivity, the core will naturally be tolerant to power increasing faults and maintain load following and self-regulating behaviours.
- Control the release of radioactive material – the Reactor Core shall maintain barriers for limiting the release of radioactive material from the fuel pellets to the primary coolant, in normal and faulted conditions.

4.1.2 Design Summary

4.1.2.1 Core Design

The Reactor Core [JAC] design comprises of 121 fuel assemblies arranged in an approximately circular arrangement. Each fuel assembly is based on a standard 17x17 lattice design utilising standard pressurised water reactor (PWR) fuel materials.

The core sits within a solid radial neutron reflector which improves power peaking at the core periphery as well as providing shielding for the RPV. The neutron reflector sits within the core barrel. When assembled, coolant flow enters the RPV inlet nozzles and flows down between the RPV and the core barrel until it reaches the bottom of the RPV. At this point, the flow passes through the flow distribution device (FDD) which straightens the flow, removes any instabilities, and evenly distributes coolant to the core region.

Coolant flows over the fuel assemblies transferring heat. At the core exit, flow passes into the outlet plenum, passing a series of control rod housing columns and eventually to the RPV outlet nozzles. Several bypass flow paths also exit to ensure adequate cooling is provided to all structural components.

4.1.2.2 Fuel Assembly Design

Each fuel assembly includes 24 guide thimbles that interface with the Rod Control Cluster Assembly (RCCA), a central instrumentation tube which allows instrumentation to be inserted into the core, and 264 fuel pins.

Each fuel pin comprises a cylindrical zircalloy clad material which contains a series of fuel pellets. Each fuel pellet contains uranium dioxide with a maximum fuel enrichment of 4.95 %. Selected fuel pellets contain gadolinia which acts as a neutron poison and helps supress reactivity at the start of cycle and control the axial power distribution within the core.

The fuel assembly is held together with top and bottom nozzles which retain the guide thimbles and interface with the RPV internals. The fuel assemblies also contain a series of grids which provide lateral support to the guide thimbles and fuel pins. Intermediate flow mixing grids also provide additional support as well as providing mixing of the coolant to promote turbulent flow and increase thermal margins.

4.1.2.3 Reactivity Control

Reactivity is controlled through insertion and withdrawal of the RCCAs. The RR SMR design does not utilise soluble boron to control or suppress reactivity, during powered operations or shutdown (see E3S Case Version 2, Tier 1, Chapter 20: Chemistry [12]). There is functionality to inject boron into the core; however, this is only operated in faulted situations following a failure to scram (see E3S Case Version 2, Tier 1, Chapter 6: Engineered Safety Features [13]).

There are three types of RCCA design in the core; shutdown RCCAs which predominately use boron carbide as an absorbing material, control RCCAs which use Silver-Indium-Cadmium Alloy (SINCAD) as an absorbing material and grey rod cluster assemblies (GRCAs) which use stainless steel.

Shutdown RCCAs are fully withdrawn during powered operations and provide the majority of shutdown and hold down reactivity worth. Control RCCAs are used to control reactivity through cycle and during reactivity transients and provide significant shutdown worth. GRCAs are used to control power peaking within the core and support transient control.

During shutdown, all RCCAs and GRCAs are fully inserted into the core. In this state, full shutdown and reactivity hold down is maintained over all possible operating conditions including the limiting temperatures, pressures, fission product inventories and with the RCCA of highest worth fully withdrawn.

4.1.2.4 Instrumentation

The primary mechanism for monitoring core power is through the ex-core neutron detectors. These detectors cover the entire power range, from low power shutdown modes, through to nominal full power.

In-core neutron detectors are present within the core and provide indication of the local power distribution. Thermocouples at the core exit also provide functionality to monitor the temperature distribution within the core.

4.2 Fuel Design

4.2.1 System and Equipment Functions

The E3S functions for the core, including the fuel assemblies, are described in Section 4.5.1. Rolls-Royce SMR Limited has tasked Westinghouse, as the fuel vendor, with designing the typical RR SMR fuel assemblies and core components. The detailed information of the fuel design presented in this chapter is primarily sourced from the Westinghouse fuel design report [14]. It will therefore not be referenced every time it is used in Section 4.2 and its subsections.

4.2.2 Design Bases

4.2.2.1 RIA Cladding Failure and PCMI

Description

Reactivity insertion accidents (RIA) are events that introduce a significant amount of positive reactivity in a short timescale, leading to cladding failure. The primary cladding failure mechanism is PCMI (pellet-clad mechanical interaction), which differs from PCI-SCC (pellet-clad interaction – stress corrosion cracking) in that it is related to stress and/or strain on the cladding from an expanding pellet during transients.

The criterion takes the form of an upper limit on maximum fuel enthalpy rise, to prevent cladding failure. The maximum fuel enthalpy is referred to as the radially-averaged peak fuel enthalpy (RAPFE), i.e. the maximum (axially and temporarily) value of fuel enthalpy (averaged over the pellet cross-sectional area) rise during the transient.

Alongside the prompt enthalpy rise failure mode described above, there are also clad failure mechanisms for high temperatures measured against 'total' radially averaged enthalpy (i.e., not rise) limits.

DNB, clad ballooning/rupture, and oxygen-induced embrittlement limits cross over with RAPFE limits to assess cladding failures that could result from an RIA. Clad ballooning/rupture (due to creep rupture) and oxygen-induced embrittlement will be revisited in a future revision of this chapter; however, the combination of RAPFE and DNB limits are likely to provide a robust assessment in the interim.

Design Limit

{REDACTED}

The Maximum RAPFE limit (following the prompt rise) of {REDACTED}kJ/kg shall be applied to preclude clad failure during a RIA event at all temperatures.

Design Bases

The assessment approach for RAPFE is expected to use steady-state data from CMS5 to provide cycle specific data and initial reactor conditions. This data will be passed to SIMULATE5-K (S5K) which will analyse the transient neutronic behaviour of several rod ejection permutations covering time in cycle, power levels/plant states, rod positions and plant parameter sensitivities. The total core power, reactivity insertion, 3D power distributions and power peaking results are extracted.

The data from S5K is used as boundary conditions for a VIPRE-01 calculation, which will use radial and axial power distributions to run sub-channel analysis calculations. S5K will provide predictions of RAPFE to be compared against the above limits, whilst VIPRE-01 will cover the DNBR criterion.

4.2.2.2 Non-Loss of Coolant Accident (LOCA) Clad Embrittlement

Description

Despite minimum DNBR and rod internal pressure limits, at very high heat fluxes a thin and insulating film of steam is produced on the surface of the fuel pin. This results in a significant increase in clad surface temperature and a rise in clad oxidation and embrittlement that can lead to clad failure.

Design Limit

A limit of {REDACTED}°C on Peak Clad Temperature (PCT) shall be applied for transients with a boiling duration {REDACTED} seconds or less [DBC-3, DBC-4].

For boiling durations longer than {REDACTED} seconds, a logarithmically reducing time versus temperature relationship, starting at {REDACTED} °C, shall be used to determine the PCT limit to protect against clad embrittlement [DBC3, DBC-4].

Design Bases

PCT analysis for non-LOCA scenarios shall be performed with RELAP5-3D and, where appropriate, S5K and VIPRE-01.

4.2.2.3 LOCA Clad Embrittlement

Description

LOCA clad embrittlement follows the same logic as non-LOCA clad embrittlement, except that a lower temperature limit is used because of the longer duration of the fault, meaning that more oxidation can occur on the clad.

Design Limit

A limit of {REDACTED} °C on PCT shall be applied for hydrogen concentrations below {REDACTED}ppm [DBC-3, DBC-4].

A limit of {REDACTED} °C on PCT shall be applied for hydrogen concentrations greater than or equal to {REDACTED}ppm [DBC-3, DBC-4].

A hydrogen concentration dependent limit on ECR shall be applied using the {REDACTED} [DBC-3, DBC-4].

Design Bases

PCT analysis for LOCA scenarios shall be performed with RELAP5-3D. The approach to ECR is subject to further methodology development. A lower temperature limit is used than in the non-

LOCA case because of the longer duration of the fault, meaning that more oxidation can occur on the clad.

The current expectation is that high performing clad materials offer significantly improved oxidation performance, thus the accepted limits noted here are appropriate. This will be reviewed with the fuel vendor.

4.2.2.4 Blowdown/Seismic/Transportation Loads

Description

Blowdown during a LOCA event can cause significant forces on the RPV and components held inside the RPV. Fuelled assemblies and control rod housing columns are particularly vulnerable to the high loads that occur during such an event. Similarly, some seismic events could cause significant loads on the Reactor Core that lead to assemblies coming into contact with one another and/or the vessel wall in extreme cases.

In both the blowdown and seismic load cases, stress on the fuel rods and spacer grids can lead to a loss of coolable geometry and/or an inability to fully insert control rods. Transportation of fuelled assemblies may also cause fuel rod damage that later impacts operation in a similar manner.

Design Limit

Fuel rod fragmentation shall not occur.

Control rod insertion shall not be impaired or an appropriate alternative shutdown method shall be incorporated into the design.

Spacer grids shall retain their geometry such that rod cooling is not impaired.

Design Bases

The assessment approach is currently immature for seismic events and for transportation.

LOCA assessments are being carried out using data generated by RELAP5-3D for a range of scenarios including hot leg breaks. This data is passed to a structural integrity code that will examine lateral and axial loads. This will determine if the required geometry for control rod insertion can be maintained. If not, alternative shutdown methods will be employed.

Verification of Seismic and LOCA event effects on fuelled assemblies is largely based on assessments and mechanical testing that will be performed in concert with the fuel vendor. Below is an outline of the *expected* design bases, as was applied for AP1000 [15].

To demonstrate that the fuel assemblies will maintain a geometry capable of being cooled under the worst-case accident infrequent fault, a plant-specific or bounding seismic analysis is performed. The fuel assembly response resulting from safe shutdown earthquake condition is analysed using time-history numerical techniques, considering the fuel assembly deflections and impact forces.

The maximum grid impact force obtained from seismic analyses must be shown to be less than the allowable grid strength. The stresses induced in the various fuel assembly non-grid components are assessed based on the most limiting seismic condition. The fuel assembly axial forces resulting from the hold-down spring load together with its own weight distribution are the primary sources of the stresses in the guide thimbles and fuel assembly nozzles. The fuel rod accident-induced stresses, which are generally very small, are caused by bending due to the fuel assembly deflections during a seismic event. The seismic-induced stresses are compared with the allowable stress limits for the fuel assembly major components.

The Rolls-Royce SMR has an aseismic bearing which is expected to give good performance in seismic analyses.

Localised yielding and slight deformation in some fuel assembly components are allowed to occur during an infrequent fault, so long as the maximum permanent deflection or deformation does not result in any violation of the functional requirements of the fuel assembly.

The nominal cold grid-to-grid clearance in core shall not be so large that it unduly increases LOCA blowdown impact loads, while remaining sufficient to meet fuel handling requirements.

4.2.2.5 Clad Stress/Strain/Collapse/Fatigue

Description

Design criteria are specified to prevent fuel cladding failure that can occur via a variety of mechanisms.

Design Limits

Clad Stress shall be limited by the maximum allowable stress according to ASME III NB-3200 at Beginning of Life (BOL) conditions [DBC-1, DBC-2, DBC-3i].

Clad Strain (maximum, uniform, permanent, end-of-life) shall be limited to {REDACTED}% [DBC-1, DBC-2].

Clad Collapse (creep) shall be prevented by limiting clad ovality to no more than {REDACTED} mm. [DBC-1, DBC-2].

Clad Fatigue due to power cycling shall be limited by ensuring the fraction of fatigue life exhausted is less than unity, based on a conservative assessment [DBC-1, DBC-2].

Clad Fatigue due to flow induced vibration shall be limited by ensuring that the maximum shear stress in the cladding is less than or equal to {REDACTED} N/mm^2 [DBC-1].

Design Bases

Clad stress will be calculated at BOL conditions, in line with the ASME III NB-3200 code. Stress will be evaluated using a Finite Element Analysis tool, using relationships derived in 2010 ASME III Subsection NB-3222. The maximum shear stress theory shall be applied when calculating the effective stresses. Allowable primary stress during frequent faults, is 10 % higher than specified for DBC-1 conditions, as per ASME III NB-3223.

Clad strain will be evaluated using the STAV7 fuel performance code.

Clad collapse will be evaluated using a code developed specifically for this analysis.

Clad Fatigue will be evaluated using the STAV7 fuel performance code.

4.2.2.6 Cladding Oxidation and Hydriding

Description

Oxidation is a result of a reaction between the coolant and cladding, leading to corrosion of the clad and thus wall thinning. Part of the hydrogen generated is incorporated into the cladding's metallic matrix, migrating under the effect of the thermal gradient to accumulate in the less hot regions, forming hydrides that may cause loss of ductility and brittleness in the cladding when cooled.

Cladding corrosion or oxidation degrades material properties, most importantly the effective cladding-to-coolant heat transfer coefficient.

Design Limit

A {REDACTED} micrometre ({REDACTED} %) limit shall be applied for clad oxidation [DBC-1].

A hydride concentration of {REDACTED} ppm shall be applied to protect against cladding failure by loss of ductility [DBC-1].

Design Bases

The assessment approach for cladding oxidation and hydriding requires further development during the detailed design phase. Maximum local cladding hydrogen content is normally calculated from the maximum local cladding oxide thickness, with such relationships embedded in the STAV7 fuel performance code.

4.2.2.7 Fuel Rod Internal Gas Pressure

Description

A design criterion on fuel rod pressurisation or rod internal gas pressure is employed to protect against cladding failure by creep rupture, or (indirectly) by fuel melting caused by excessive clad creep out. Both phenomena are accelerated by thermal feedback.

Increasing fuel-clad gap sizes increase fuel temperatures, which increases fission gas release, leading to an increase in fuel rod pressure. This criterion is typically applied in normal operations only; however, it will also be applied in frequent design basis faults to be consistent with other GDA submissions.

Design Limit

Fuel Rod Internal Gas Pressure shall be limited below the pressure that causes the outward cladding creep rate to exceed the fuel effective swelling rate [DBC-1, DBC-2, DBC-3i].

Design Bases

The STAV7 fuel performance code will be used to perform this assessment on a conservative design basis. Fission gas release uncertainties are particularly important in this analysis and will be accounted for appropriately.

4.2.2.8 Pellet-Clad Interaction and Stress Corrosion Cracking

Description

PCI-SCC is dependent on power ramps during startup, manoeuvring (e.g. load follow, rod adjustments and rod swaps), and normal operation transients (i.e. DBC-1 and DBC-2). This is distinct from PCMI that is typically caused by rapid transients (DBC-3 and DBC-4).

RR SMR will use vendor specific ramp rates based on their operational experience and analytical modelling. The below limit is based on European Utility Requirements (EUR) performance requirements for unconditioned fuel (Chapter 2 of EUR) for start-up only. This limit is subject to further adjustments/relaxations after conversations with the fuel vendor and further limits on load following ramp rates shall be added in future revisions of this chapter.

Design Limit

Start-up ramp rates for fresh fuel shall be limited to a rate less than or equal to 3 % per hour [DBC-1, DBC-2i].

Manoeuvring ramp rates (covering load following, frequency changes and other anticipated load changes) shall be limited; however, an appropriate limit needs to be discussed with the fuel vendor [DBC-1, DBC-2i].

Design Bases

SIMULATE5 will be used to calculate ramp rates and compare against vendor specific limits. Vendor methodologies exist for the evaluation of PCI fuel protection during frequent fault scenarios.

4.2.2.9 Fuel Melting

Description

The risk of fuel melting can be minimised by applying a rod internal pressure and a linear heat generation rate (LHGR) limit. Note that fuel thermal conductivity and melting temperature will be adversely affected by the presence of gadolinia. Reducing enrichment ensures adequate margin for this fuel type.

A limit on LHGR directly limits the maximum centreline temperature attained.

Design Limit

Fuel centreline temperature shall remain below {REDACTED} ^{0}C for gadolinia doped fuel and {REDACTED} ^{0}C for undoped fuel respectively. The same limit shall be conservatively applied regardless of local burnup [DBC-1 to 4].

Design Bases

The TRANSURANUS fuel performance code and RELAP5-3D can evaluate peak fuel temperatures in LOCA and Non-LOCA analysis. STAV7, S5K and VIPRE-01 can all be employed on a case-by-case basis for design basis analysis depending on the context.

4.2.3 Design Evaluation

The fuel assemblies, fuel rods, and in-core control components are designed to satisfy the performance and safety criteria and the mechanical design bases. The Westinghouse 'STAV7' code is used to confirm the performance of the fuel rods. All assessments have been performed using bounding power histories representative of the equilibrium cycle operations. Further details of the modelling approach and assessment results can be found in Fuel Performance Assessment [16].

4.2.3.1 RIA Cladding Failure and PCMI

Analysis of clad failure due to the PCMI following a reactivity addition accident is presented within E3S Case Version 2, Tier 1, Chapter 15: Safety Analysis [17].

4.2.3.2 Non-LOCA Clad Embrittlement

Performance of the fuel clad against the non-LOCA clad embrittlement limits shall be reported in the Fault Schedule [18].

4.2.3.3 LOCA Clad Embrittlement

Performance of the fuel clad against the LOCA clad embrittlement limits shall be reported in the Fault Schedule [18].

4.2.3.4 Blowdown/Seismic/Transportation Loads

Analysis of the mechanical integrity of the fuel assembly as a result of blowdown, seismic or transportation loads shall be conducted in conjunction with the fuel vendor and reported during the detailed design phase.

4.2.3.5 Clad Stress/Strain/Collapse/Fatigue

Generic cladding stress and instability calculations for fresh optimized ZIRLO™ fuel rods during normal operation (Level A) and frequent faults (Level B) have been done with finite element simulation in the ANSYS program and then evaluated in accordance with ASME III Subsection NB.

Stress assessments have shown that the maximum allowable stress across the clad is in excess of that imparted from the system pressure. As such the design criteria for clad stress are fulfilled.

Clad strain predictions have been made and have been shown to be within the design limits for normal operations and for frequent faults.

Calculations of clad ovality, which can lead to clad collapse, have been conducted and shown to be well within the design limits.

Conservative calculations of the clad fatigue as a result of load following and frequent faults have shown that the criterion for clad fatigue can be preserved following a total of 200 load follow cycles per year in conjunction with a limited number of high-power frequent faults.

4.2.3.6 Cladding Oxidation and Hydriding

Predictions of the limiting oxide thickness and hydriding have been made using bounding power histories. Using conservative correlations at a 95 % confidence interval, both the oxide thickness and hydriding have been shown to be significantly less than the design limits.

Due to the lack of comprehensive empirical data on clad performance in KOH chemistry and in the lack of soluble boron, autoclave testing shall be conducted in the detailed design phase to demonstrate the anticipated benefits of operating in a more benign chemistry environment.

4.2.3.7 Fuel Rod Internal Gas Pressure

Assessments of the rod internal pressure following conservative power histories have been calculated. Fuel pins without the gadolinia were found to be more limiting due to their higher operational powers and extended burnups.

Limiting rod internal pressures were found to be approximately {REDACTED}, well within the design limit {REDACTED}.

4.2.3.8 Pellet-Clad Interaction and Stress Corrosion Cracking

Assessments of PCI and SCC during startup and power transients have not yet been performed due to the lack of Rolls-Royce SMR Limited specific PCI mitigation guidelines which shall be developed in conjunction with the fuel vendor during the detailed design phase.

PCI risk is directly related to the ramp rates being applied to the fuel. If PCI risk is considered to be unduly high when operating in accordance with the target ramp rates suggested in the EUR, operational restrictions shall be put in place to eliminate this risk. As such, the safety implications of PCI during normal operations are considered to be low.

4.2.3.9 Fuel Melting

Fuel centreline temperatures have been calculated for normal operational conditions and during frequent faults.

Fuel temperatures during normal operations were found to be most limiting in fuel pins with no gadolinia i.e. the uranium enrichment ensures doped fuel pins are non-limiting. Maximum fuel temperatures were found to be {REDACTED} with a design limit of {REDACTED} °C.

To assess the response to frequent faults, the power in each pin was raised until fuel melt was initiated. Limiting powers for each pin type and burnup step have bene calculated and can be used in the assessment of the fuel in frequent faults. Performance against these limits shall be reported in the Fault Schedule [18] .

4.2.4 Design Description

4.2.4.1 Fuel Assembly

The standard fuel assembly design for the RR SMR is based on generic design information available on a Westinghouse 17x17 RFA design. A difference between the two designs is that the RR SMR fuel assembly has been designed to interface with upper-mounted in-core instrumentation, versus bottom-mounted in-core instrumentation. Another key difference is that the standard RFA fuel length has a 365.76 cm (12 ft) active core, whereas the RR SMR fuel will be shorter, with a 280.0 cm (9.18 ft) active core length.

Each standard fuel assembly consists of 264 fuel rods, 24 guide thimbles, and one instrumentation tube in a 17x17 array arranged within a supporting structure. The instrumentation thimble is located in the centre position and provides a channel for insertion of an in-core neutron detector if the fuel assembly is located in an instrumented core position. The guide thimbles provide channels for insertion of a RCCA, a GRCA or a neutron source assembly, depending on the position of the particular fuel assembly in the core. If control rods or source assemblies are not required, thimble plugs can be inserted to limit the fuel bypass flow. The guide thimbles are joined to the top and bottom nozzles of the fuel assembly and provide the supporting structure for the fuel grids. Figure 4.2-1 shows a full-length view of the generic fuel assembly.

The fuel rods are loaded into the fuel assembly structure so that there is clearance between the fuel rod ends and the top and bottom nozzles. The fuel rods are supported within the fuel assembly structure by either 7 or 8 structural grids (top grid (one), bottom grid (one), intermediate grids (five or six), and possibly two IFM grids (if it is determined that they are needed), and one protective grid. Top, bottom, and intermediate grids provide axial and lateral support to the fuel rods. In addition, the two IFM grids located near the centre of the fuel assembly and between the intermediate grids provide additional coolant mixing, if it is determined that it is necessary/needed. Debris failure mitigation is provided by a combination of the protective grid with a debris filter bottom nozzle; the fuel cladding with oxide coating near the bottom; and the long, solid fuel rod bottom end plug.

Fuel assemblies are installed vertically in the reactor vessel and stand upright on the lower core plate, which is fitted with alignment pins to locate and orient each assembly. After the fuel assemblies are set in place, the upper support structure is installed. Alignment pins, built into the upper core plate, engage and locate the upper ends of the fuel assemblies. The upper core plate then bears down against the holddown springs on the top nozzle of each fuel assembly to hold the fuel assemblies in place.

Improper orientation of fuel assemblies within the core is prevented by the use of an indexing hole in one corner of the top nozzle top plate. The assembly is oriented with respect to the handling tool and the core by means of a pin inserted into this indexing hole. Visual confirmation of proper orientation is also provided by an engraved identification number on the opposite corner of the top plate.

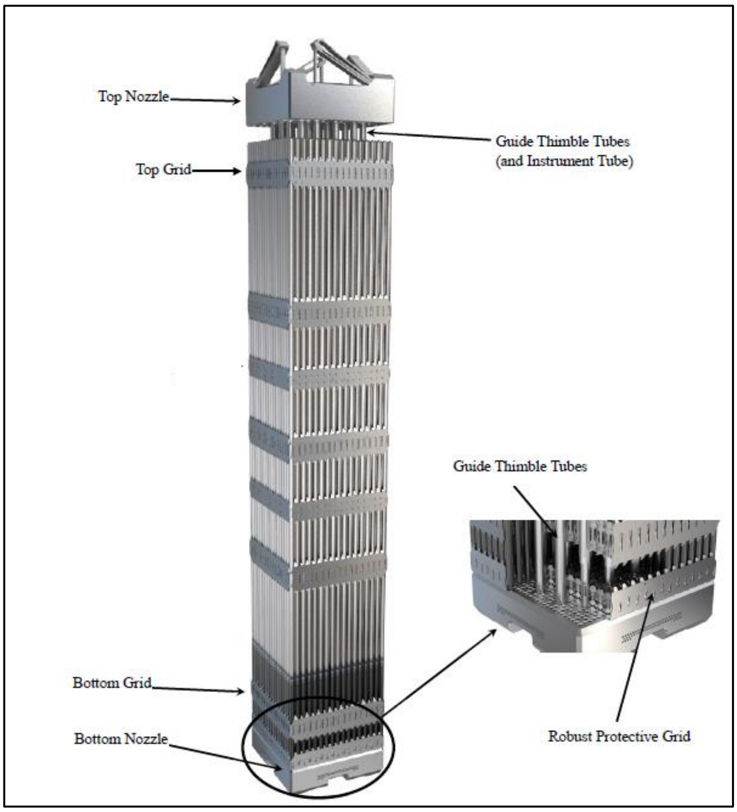

Figure 4.2-1: Generic Fuel Assembly Schematic

4.2.4.2 Fuel Rods

The fuel rods consist of uranium dioxide (UO_2) ceramic pellets contained in partially recrystallized annealed (PRXA) tubing constructed from a zirconium alloy (Optimized ZIRLO™ High Performance Fuel Cladding Material), which is plugged and seal welded at the ends to encapsulate the fuel. Optimized ZIRLO™ is selected for its corrosion resistance, mechanical properties and low neutron absorption cross-section. Figure 4.2-2 shows a generic schematic of the fuel rod.

The fuel pellets are right circular cylinders consisting of slightly enriched uranium dioxide powder that has been compacted by cold pressing and then sintered to the required density. {REDACTED}.

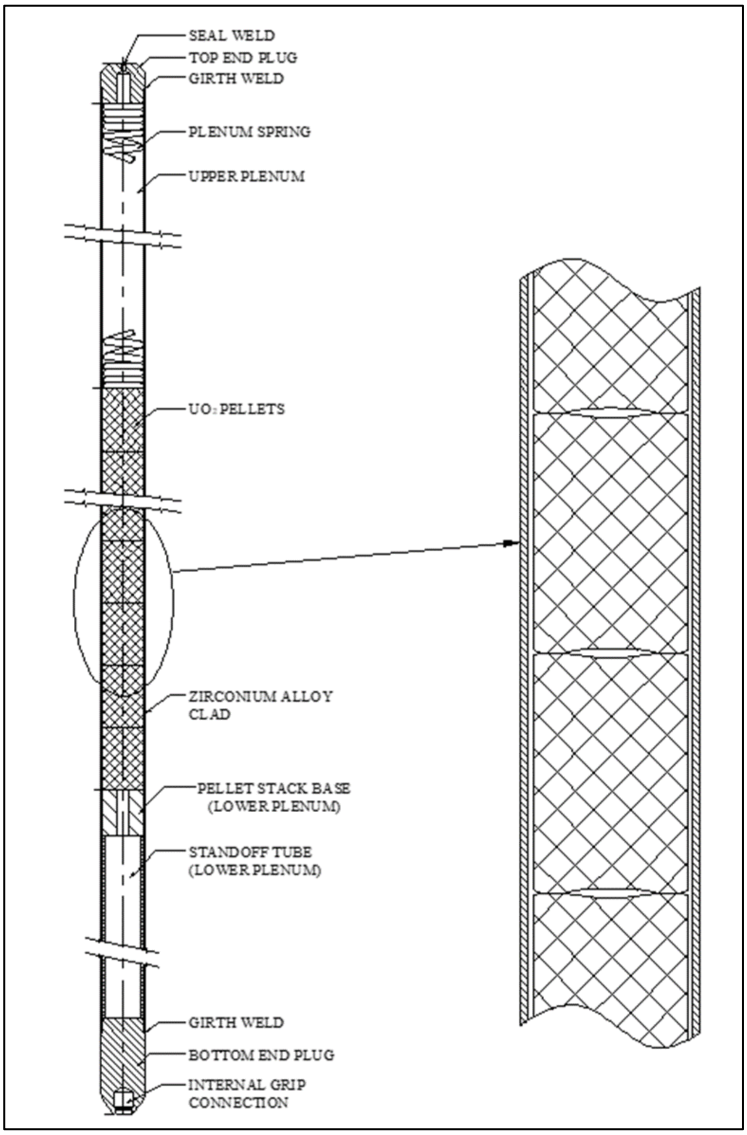

Figure 4.2-2: Generic Fuel Rod Schematic

Void volume and clearances are provided within the rods to accommodate fission gases released from the fuel, differential thermal expansion between the clad and the fuel, and fuel density changes during irradiation. To facilitate the extended burnup capability necessitated by longer operating cycles, the fuel rod is designed with two plenums (upper and lower) to accommodate the additional fission gas release. The upper plenum volume is maintained by a fuel pellet holddown spring. The lower plenum volume is maintained by a standoff assembly.

Shifting of the fuel within the clad during handling or shipping prior to core loading is prevented by the fuel pellet holddown spring: a stainless-steel helical spring that bears on top of the fuel pellet stack. The spring also prevents gaps from opening in the fuel stack as pellets densify and provides support to the cladding due to creepdown during steady-state operation. Assembly consists of plugging and welding the bottom of the cladding; installing the bottom plenum spacer assembly, fuel pellets, and top plenum spring; and then plugging and welding the top of the rod.

{REDACTED}

The fuel rods are internally pressurised with helium during the welding process to minimise compressive clad stresses and prevent clad flattening under reactor coolant operating pressures.

The RR SMR reactor fuel rod design includes axial blankets. The axial blankets consist of fuel pellets of a reduced enrichment at each end of the fuel rod pellet stack. Axial blankets reduce neutron leakage axially and improve fuel utilization. {REDACTED}.

A fraction of the fuel rods will contain pellets which incorporate gadolinia (Gd_2O_3) in the uranium dioxide (UO_2) fuel matrix as a burnable absorber for reactivity control and core power distribution control. Axial zoning of gadolinia content may also be employed. The number and placement of burnable absorber (BA) rods within an assembly may vary depending on core loading pattern.

4.2.4.3 Fuel Assembly Structure

As shown in Figure 4.2-1, the fuel assembly structure consists of a bottom nozzle, top nozzle, fuel rods, guide thimbles, and grids.

Bottom Nozzle

The bottom nozzle serves as the bottom structural element of the fuel assembly and directs the coolant flow distribution to the assembly. The nozzle is fabricated from {REDACTED} stainless steel and consists of a perforated plate and casting that incorporates a skirt and four angle legs with bearing pads. The legs and skirt form a plenum to direct the inlet coolant flow to the fuel assembly. The perforated plate also prevents accidental downward ejection of the fuel rods from the fuel assembly. The bottom nozzle is fastened to the fuel {REDACTED}, which penetrate through the nozzle and engage with a threaded plug in each guide thimble. The flow hole pattern, together with top nozzle ligaments, limits the fuel rod movement within the cavity between the two nozzles.

Coolant flows from the plenum in the bottom nozzle upward through the penetrations in the plate to the channels between the fuel rods. The penetrations in the plate are positioned between the rows of the fuel rods.

In addition to serving as the bottom structural element of the fuel assembly, the bottom nozzle also functions as a debris filter. The bottom nozzle perforated plate contains a multiplicity of flow holes sized to minimise passage of detrimental debris particles into the active fuel region of the core while maintaining sufficient hydraulic and structural margins. Furthermore, the skirt provides improved bottom nozzle structural stability and increased design margins to reduce damage from abnormal handling. Small flow {REDACTED} and help ensure that debris doesn’t bypass the bottom nozzle and travel into the fuel bundle region where it could possibly cause debris-induced fretting failures.

Axial loads (from top nozzle holddown springs) imposed on the fuel assembly and the weight of the fuel assembly are transmitted through the bottom nozzle to the lower core plate. Indexing and positioning of the fuel assembly is controlled by alignment holes in two diagonally opposite bearing pads that mate with locating pins in the lower core plate. Lateral loads on the fuel assembly are transmitted to the lower core plate through the locating pins.

{REDACTED}

Top Nozzle

The reconstitutable top nozzle functions as the upper structural component of the fuel assembly and, in addition, provides a partial protective housing for the RCCA, discrete BA, or other core components. The top nozzle assembly includes four sets of holddown springs, which are secured to the top nozzle top plate. {REDACTED}.

The adapter plate contains various-shaped holes to permit the flow of coolant upward through the top nozzle. Round holes are provided in the adapter plate to accept (guide thimble) inserts that are mechanically locked to the adapter plate using a lock tube. The unique design of the insert joint and lock tube are the key design features of the reconstitutable top nozzle which allows the top nozzle to be removed during a fuel outage for fuel rod examination or in case there is a need to reconstitute the fuel assembly because of any leaking fuel rods.

The ligaments in the adapter plate cover the top of the fuel rods, precluding any upward ejection of the fuel rods from the fuel assembly. The enclosure is a boxlike structure that establishes the distance between the adapter plate and the top plate. The top plate has a large square hole in the centre to permit access for the RCCA or other components. Holddown springs are mounted on the top plate and are retained by retaining pins located at diagonally opposite corners of the top plate.

The top plate also contains integral pads located on the two remaining top nozzle corners. The pads include alignment holes which, when fully engaged with the reactor internals upper core plate guide pins, provide proper alignment to the fuel assembly, reactor internals, and RCCA.

To remove the top nozzle assembly, a tool is first inserted through a lock tube and expanded radially to engage the bottom edge of the tube. An axial force is then exerted on the tool in the upward direction, which overrides local lock tube deformations and withdraws the lock tubes from the inserts. After the lock tubes have been removed, the nozzle assembly is removed by raising it off the upper slotted ends of the nozzle inserts, which deflect inwardly under the axial lift load.

With the top nozzle assembly removed, direct access is provided for fuel rod examination or replacement. Reconstitution is completed by the remounting of the nozzle assembly and the insertion of lock tubes.

The RR SMR fuel design has an instrumentation hole which allows for the insertion of upper-mounted instrumentation.

Guide Thimbles and Instrument Tube

The guide thimbles are structural members that provide channels for the neutron absorber rods, neutron source rods, or other assemblies. Each guide thimble is fabricated from ZIRLO™ alloy with constant outer and inner diameter over the entire length. Separate dashpot tubes, made from ZIRLO™ tubing, are inserted into the bottom portion of the guide thimble tubes. The larger tube diameter at the top section provides a relatively large annular area necessary to permit rapid control rod insertion during a reactor trip, as well as to accommodate the flow of coolant during normal operation. Holes provided on the guide thimble above the dashpot reduce the rod drop time and also provide sufficient cooling to the RCCAs and GRCAs when they are inserted into the core without unduly reducing the flow past the fuel rods.

The lower portion of the guide thimble with the dashpot tube results in a dashpot action near the end of the control rod travel during normal trip operation. The dashpot is closed at the bottom by means of an end plug, which is provided with a small flow port to avoid fluid stagnation in the dashpot volume during normal operation.

{REDACTED} An expansion tool is inserted inside the nozzle insert and guide thimble to the proper elevation. The four lobes on the expansion tool force the guide thimble and insert outward locally to a predetermined diameter, therefore joining the two components.

Upon installation of the top nozzle assembly, the bulge near the top of the nozzle insert is captured in a corresponding groove in the thimble hole of the top nozzle adapter plate. The mechanical connection between the nozzle insert guide thimble and top nozzle is made by insertion of a lock tube into the insert. The design of the top grid sleeve-guide thimble and top nozzle insert guide thimble bulge joint connections have been mechanically tested and found to meet applicable design criteria.

The fuel rod support grids, {REDACTED}, are secured to the guide thimbles using a similar bulge joint connection to create an integral structure.

The intermediate mixing vane and IFM grids employ a single-tier bulge connection between the grid sleeve and guide thimble as compared with the two-tier bulge connection used for the top grid. The design of the single tier bulge joint connection has also been mechanically tested and meets the design requirements.

The Alloy-718 bottom grid is secured to the guide thimble assembly by a double-tier bulge connection between the grid sleeve and guide thimble. The design of the double-tier bulge joint connection has also been mechanically tested and meets the design requirements.

The lower end of the guide thimble is fitted with a welded end plug. {REDACTED}. The spacer is captured between the guide thimble end plug and the bottom nozzle by means of a (thimble) locking screw.

The described methods of grid fastening are standard and have been used successfully since the introduction of guide thimbles in 1969 on thousands of Westinghouse designed and built fuel assemblies.

The central instrumentation tube in each fuel assembly is constrained by seating in counterbores located in both top and bottom nozzles. The instrumentation tube has a constant diameter and provides an unrestricted passageway for the in-core neutron detector which enters the fuel assembly from the top nozzle. Furthermore, the instrumentation tube is secured to the top, bottom, IFM, and mid-grids with bulge joint connections like those previously discussed for securing the grids to the guide thimbles.

Grid Assemblies

The fuel rods are supported at intervals along their lengths by grid assemblies that maintain the lateral spacing between the rods throughout the design life of the assembly. Each fuel rod is given support at six contact points within each structural grid by the combination of support dimples and springs. The grid assembly consists of individual slotted straps assembled and interlocked into an egg-crate-type arrangement with the straps permanently joined at their points of intersection. The straps may contain springs, support dimples, and mixing vanes, or any such combination.

Two types of structural grid assemblies are used on the fuel assembly of the RR SMR fuel design. One type, with mixing vanes projecting from the edges of the straps into the coolant stream, is used in the high heat flux region of the fuel assemblies to promote mixing of the coolant. This mid-grid is illustrated in Figure 4.2-3.

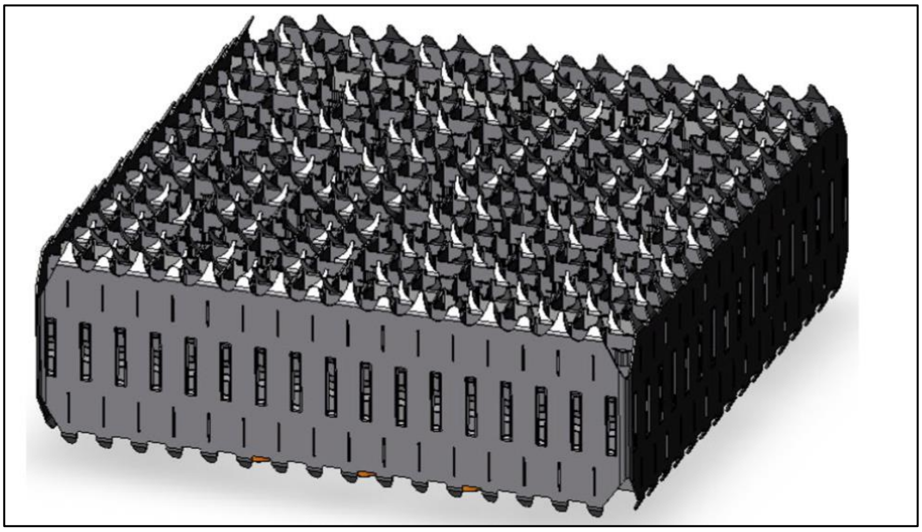

Figure 4.2-3: Schematic of a Mid-Grid

The other type, located at the top and bottom of the assembly, does not contain mixing vanes on the internal straps. The outside straps on the grids contain mixing vanes that, in addition to their mixing function, aid in guiding the grids and fuel assemblies past projecting surfaces during handling or loading and unloading of the core.

Because of its corrosion resistance and high-strength properties, the bottom grid material chosen for the fuel assembly design is {REDACTED}. The top grid is also fabricated from {REDACTED}. The magnitude of the grid restraining force on the fuel rod is set high enough to minimise possible fretting without overstressing the cladding at the points of contact between the grids and fuel rods. The grid assemblies are designed to allow axial thermal expansion of the fuel rods without imposing restraint sufficient to develop buckling or distortion of the fuel rods.

The intermediate (mixing vane) or structural grids on the fuel assembly are made of Low Tin ZIRLO™. This material was selected to take advantage of its inherent low neutron capture cross section. Low Tin ZIRLO™, like other zirconium alloys used in the nuclear industry, contains a high percentage of zirconium and therefore inherits a low capture cross section for thermal neutrons from zirconium.

The percent of other elements (nickel, tin, iron, niobium, etc.) in the zirconium alloy are limited to the content necessary for good mechanical properties and corrosion resistance.

The mid grids have thicker straps and incorporate the same grid cell support configuration as the {REDACTED}. The interlocking strap joints for the protective grid are also fabricated by laser welding.

The mixing vanes incorporated in the intermediate grids induce additional flow mixing among the various flow channels in a fuel assembly as well as between adjacent fuel assemblies. This additional flow mixing enhances thermal performance.

The presence of IFM grids is being evaluated based upon desired thermal margins. The IFM grids are typically located at selected spans between the mixing vane structural grids and incorporate a similar mixing vane array. Their prime function is midspan flow mixing in the hotter fuel assembly spans where thermal margins are lowest. Each IFM grid cell contains four dimples that are designed to prevent midspan channel closure in the spans containing intermediate flow mixers and fuel rod contact with the mixing vanes. This simplified cell arrangement allows for relatively short grid cells so that the intermediate flow mixer grid can accomplish its flow mixing objective with minimal impact on the overall fuel assembly pressure drop.

The IFM grids, like the structural grid assemblies, are fabricated from Low Tin ZIRLO™. The IFM grids are manufactured using the same basic techniques as the structural grid assemblies and are bulged to the guide thimble tubes and the instrumentation tube via sleeves welded at the bottom of applicable grid cells.

Grid impact testing has been performed on the structural grids and the IFM grids. The purpose of the testing was to determine the dynamic buckling, or crush strength of the grids. The grid impact testing was performed at an elevated temperature of {REDACTED}. This temperature is a conservative value representing the core average temperature at the mid grid locations.

The IFM grids are not intended to be structural members but they do share the loads of the structural grids during faulted loading and, as such, enhance the load-carrying capability of the fuel assembly.

4.2.4.4 Rod Cluster Control Assemblies

The rod cluster control assemblies are divided into two categories: control and shutdown. The shutdown rods shall be removed from the core during high power operations (although they may be inserted slightly during startups) and inserted to provide the majority of shutdown reactivity. The shutdown rods shall be moved in a number of groups to minimise any single reactivity increase during startup. The control rods shall compensate for reactivity changes due to variations in operating conditions of the reactor, that is, power and temperature variations, and will be the primary mechanism for controlling reactivity during plant transients. The control rods shall be separated into a number of groups, of which, only {REDACTED} will ever be inserted into the core whilst at power. During operations, the control groups available to be inserted into the core will be changed to minimise control rod depletion, promote an even core burnup and to maintain power peaking within the design limits.

Two nuclear design criteria have been employed for selection of the control group. First, the total reactivity worth must be adequate to meet the nuclear requirements of the reactor. Second, since these rods may be partially inserted at power operation, the total power peaking factor should be low enough to confirm that the power capability is met. The control and shutdown (along with the grey rods) groups provide adequate shutdown margin (SDM) for all modes of operation.

An RCCA is comprised of 24 individual neutron absorber rods fastened at the top end to a common spider assembly, illustrated in Figure 4.2-4. The absorber material used in the control rods is SINCAD, which is essentially “black” to thermal neutrons and has sufficient additional resonance absorption to significantly increase worth. The absorber material in the shutdown rods is boron carbide. The absorber material is in the form of bars sealed in coldworked stainless steel tubes. Sufficient diametral and end clearance is provided to accommodate relative thermal expansions. The control rods have bottom plugs with bulletlike tips to reduce the hydraulic drag during reactor trip and to guide smoothly into the dashpot section of the fuel assembly guide thimble assemblies. The material used in the absorber rod end plugs is {REDACTED}.

The spider assembly is in the form of a central hub with radial vanes with cylindrical fingers from which the absorber rods are suspended. Internal groovelike profiles to facilitate handling tool and drive rod assembly connection are machined into the upper end of the hub. Coil springs inside the spider body absorb the impact energy at the end of a RCCA insertion (following reactor trip). The radial vanes may either be joined to the hub by welding and brazing, and the fingers are joined to the vanes by brazing, or the vanes and fingers may be integral with the spider body. A bolt that holds the springs and retainer is threaded into the hub within the skirt and welded to prevent loosening while in service. The components of the spider assembly are made from {REDACTED}.

The absorber rods are fastened securely to the spider. The rods are first threaded into the spider fingers and then secured with a locking device. The end plug below the pin position is designed with a reduced section to permit flexing of the rods to correct for small operating or assembly misalignments.

The overall length of the RCCA is such that, when the assembly is withdrawn through its full travel, the tips of the absorber rods remain engaged in the fuel assembly guide thimbles so that alignment between rods and thimbles is always maintained. Since the rods are long and slender, they are relatively free to conform to any misalignments with the guide thimble. Any such misalignments are small and the RCCA rods remain positioned within the core so that flow-induced wear on the RCCA absorber rods as they sit at the top of the guide thimbles is evenly spread and within acceptable limits.

{REDACTED}

Figure 4.2-4: Schematic of a Generic RCCA

4.2.4.5 Gray Rod Cluster Assemblies

The purpose of GRCAs is to provide powershape control and reactivity control capability during operation. GRCAs contain a smaller quantity of neutron absorber relative to RCCAs. The GRCAs are used in load follow manoeuvring and base load operation to control core temperature and power; they provide a mechanical shim capability that replaces the need for soluble boron, that is, chemical shim, normally used for this purpose.

Three concepts of grey rods are under consideration. The first concept is a solid bar of stainless steel. The second concept is a “tube-in-tube” design where a tube of stainless steel is contained within a conventional RCCA cladding. The third concept is a design that contains tungsten inside of a conventional RCCA cladding. As the core design and fuel design progress then the need for a grey rod and the design of that rod will be finalized. The baseline concept at RD7/DRP1 is stainless steel bars.

4.2.4.6 Neutron Source Assemblies

The purpose of a neutron source assembly is to provide a base neutron level to give confidence that the detectors are operational and responding to core multiplication neutrons. For the first core, a neutron source is placed in the reactor to provide a positive neutron count of at least two counts per second on the source range detectors attributable to core neutrons. The detectors, called source range detectors, are used primarily during subcritical modes of core operation.

The source assembly also permits detection of changes in the core multiplication factor during core loading, refuelling, and approach to criticality. This can be done since the multiplication factor is related to an inverse function of the detector count rate. Changes in the multiplication factor can be detected during addition of fuel assemblies while loading the core or changes in control rod positions.

Two primary source assemblies are installed in the initial load of the Reactor Core. Each primary source assembly contains one primary source rod and a number of thimble plugs. Neutron source assemblies are employed at opposite sides of the core.

The primary source rods both use the same cladding material as the absorber rods. The primary source rods contain capsules of californium source material and spacers to position the source material within the cladding. The rods in each assembly are fastened at the top end to a hold-down assembly.

{REDACTED}.

4.2.4.7 Thimble Plugs

Thimble Plugs are not currently utilised in the Reactor Core to reduce coolant flow in the assembly guide thimbles. A decision whether to include Thimble Plugs shall be made during the detailed design phase based on the estimated benefit in DNBR.

4.2.5 Examination, Maintenance, Inspection and Testing

Examination, Maintenance, Inspection and Testing (EMIT activities) shall be conducted throughout the fuel cycle to ensure each fuel assembly is built as intended and continues to operate within safe limits. A summary of the EMIT activities which shall be conducted on the Fuel and Core, and details of how these meet the associated safety claims are listed in Reference [19].

4.2.5.1 Quality Assurance and Quality Control

Appropriate quality assurance (QA) process shall be put in place by the fuel vendor to ensure the manufactured product is compliant with the design intent, and to limit any manufacturing defects.

Details of the fuel vendor’s QA processes shall be reported in the detailed design phase.

4.2.5.2 New Fuel Receipt

Upon arrival at the RR SMR site, fuel shall be inspected to ensure fuel damage has not occurred during transport. Details of the fuel receipt processes shall be developed in the detailed design phase.

4.2.5.3 Post Irradiation Examination

Post irradiation examination (PIE) shall be conducted on fuel assemblies during refuelling periods and after final discharge from the core. Focus shall be placed on the first of a fleet to ensure the margins to fuel performance limits are monitored and well understood. Details of the PIE strategy can be found in [20].

4.3 Nuclear Design

4.3.1 Design Bases

The design basis for the fuel and core has been summarised in the fuel and core design basis summary report [21] and relevant design criteria have been collated in the justification of design limits report [22]. A summary of the main arguments and evidence is provided in subsequent sections.

4.3.1.1 Fuel Burnup

Description

Burnup is a measure of the energy density obtained from nuclear fuel with a higher value representing greater uranium utilisation and improved fuel cycle economics. However, higher burnups also lead to damage to the pellet and fuel pin cladding, with in the fuel assembly.

Fuel burnup limits are set to improve performance during normal operation, frequent faults, and infrequent faults. Limits have steadily increased following ongoing advanced fuel material research, improvements in understanding fuel behaviour during accident conditions and changes in fuel morphology at higher burnups.

Violation of a burnup limit does not necessarily lead to fuel failure during normal operation as performance depends on the power history. Applying a burnup limit is useful when designing a core reload, although cycle-specific fuel performance assessments would need to be completed to ensure adequate steady state and accident behaviour.

Design Limit

A peak pellet burnup limit of 65 GWd/tHM is applied. This value is within the range of RGP limits described in [22].

Design Bases

The assessment approach uses steady-state data from the core management system (CMS) 5 to provide cycle specific data. Predicted peak pellet burnups will be compared to the limit and changes to the core design made to ensure compliance.

Analysis is conducted to account for all uncertainties in the operating conditions of the core including temperature, power, and the preceding cycle length. The burnup assessments shall account for initial and early cycles as well as the equilibrium cycle.

A conservative load factor is applied when determining peak burnup values; this essentially assumes full power operation for the entire cycle (and preceding cycles), neglecting load following and reduced power stretch-out operations which would reduce cycle burnup. A less conservative load factor is determined for future analysis to ensure that the burnup assessments are bounding without being unduly pessimistic.

4.3.1.2 Reactivity Coefficients

Description

A negative fuel temperature and moderator temperature coefficient (MTC) are the principal means by which reactor power can be passively controlled. The fuel temperature responds almost immediately to power changes and therefore the fuel temperature (Doppler) coefficient is particularly important in rapid reactivity insertion accidents such as a rod ejection. It is important the minimum Doppler coefficient remains suitably negative to minimise the amount of energy deposition in the fuel during such faults.

The temperature rise in the coolant is much more gradual following a rise in core power, with a time constant of several seconds (i.e. the time required for coolant to flow through the primary circuit). Thus, moderator temperature feedback is much slower compared with fuel temperature feedback.

Compared with a typical borated PWR, MTC is consistently negative, as in a borated plant a decreasing density (as a result of an increase in temperature) not only reduces scattering cross section but also capture cross section.

Bulk voidage of the coolant is not anticipated under nominal operating conditions; however, given the strongly negative moderator temperature coefficient, the void coefficient is also expected to be negative.

Design Limit

Doppler and moderator temperature coefficients shall remain negative [DBC-1 to DBC-4].

In general, Doppler coefficient should not be overly negative as this will increase power defect and adversely impact SDM.

Unlike for a typical borated PWR design, a minimum limit (i.e. most negative) is judged to be less important for SDM calculations for the RR SMR, given SDMs are already calculated for the most reactive core state (i.e. fully dense and cold coolant with no xenon). However, a strong negative MTC would impact faults that involve cold water addition.

Doppler and moderator temperature coefficient design limits will be updated and justified following plant performance assessments. The limits ultimately chosen will ensure adequate fuel performance by minimising the amount of energy deposition to acceptable levels (during a rod ejection accident) and to ensure acceptable margins to fuel melt (during heat-up faults).

Design Bases

The MTC and Doppler coefficient calculations in SIMULATE5.

Analysis of MTC and Doppler coefficients is conducted across for all possible operational statepoints and take into account historic burnup conditions, to ensure that these reactivity coefficients cover the full range of steady state operating conditions that a frequent fault can initiate from. Additional analysis of reactivity coefficients may be required to cover transient conditions; the statepoints table can be extended as necessary, though all such data will be for steady state conditions rather than modelling the progression of the transient.

4.3.1.3 Control of Power Distribution

Description

The distribution of power within the core is a key criterion for the safe and efficient operation of the reactor. The power distribution must facilitate adequate thermal margins such that fuel integrity can be demonstrated under DBC-1 and DBC-2 conditions.

Several parameters are used to describe the complex nature of the power distribution in concise limits that can be analysed and optimised. The key parameters are the power peaking factors F_Q and $F_{\Delta H}$.

The heat flux hot channel factor (F_Q) is also referred to as the total peaking factor or peak pin relative linear power density. This parameter is the ratio of the maximum LHGR anywhere in the core to the core average LHGR.

The nuclear enthalpy rise hot channel factor ($F_{\Delta H}$) is the ratio of the maximum integrated pin power within the core to the average pin power. Therefore, $F_{\Delta H}$ is a measure of the maximum total power produced in any fuel pin in the core. The peak $F_{\Delta H}$ that is acceptable is linked to the capability of the core and plant systems to provide adequate cooling under all normal and frequent fault conditions.

Design Limit

Power distributions are not ascribed a specific set of design limits. However, analysis limits on F_Q must be defined to prevent clad failure and fuel melt, and analysis limits on $F_{\Delta H}$ must be defined to prevent DNBR. Any analysis limits developed for normal conditions shall need to ensure sufficient additional margin for frequent faults.

An $F_{\Delta H}$ of {REDACTED} is assumed by the DNBR analysis, prior to the application of uncertainties, therefore this represents the current analysis limit for the power distribution assessments. An analysis limit on F_Q shall require further development of the design basis for prevention of clad failure and fuel melt. Improved forecasting and risk profiling for thermal and transient analysis margins will inform updated F_Q and $F_{\Delta H}$ analysis limits for RD8 and presented in Version 3 of the generic E3S Case.

Design Bases

Maximum F_Q and $F_{\Delta H}$ values as well as the range of axial offset (AO) values are monitored and optimised against during design optimisation.

For RD7/DRP1, optimisation focused on full power equilibrium cycle operation, with more limited optimisation for early cycles and load following operations. For RD8/DRP2, power distributions will be analysed, and the design optimised for all state points, to ensure that power distributions are bounding for all steady state conditions and all conditions a frequent fault can initiate from. This will be presented in Version 3 of the generic E3S Case.

4.3.1.4 Fuel Enrichment

Description

The RR SMR is fuelled with UO_2. For fuel pellets containing gadolinia (integral burnable poison), it is important to reduce enrichment to ensure these pellets and pins are non-limiting. This is due to the gadolinium adversely affecting thermal conductivity and melting temperature.

Design Limit

For undoped fuel, a maximum fuel enrichment of 4.95 w% has been applied [DBC-1].

For gadolinia doped fuel, a sufficient cut-back factor (to be confirmed with the fuel vendor) will be required and will be applied to ensure sufficient margin to fuel melt [DBC-1].

Design Bases

Fuel enrichment and burnable poison loading are important design parameters that can vary both radially and axially within a fuel assembly to optimise core performance and to maintain adequate safety margins. For initial transition and equilibrium cycles, all fresh fuel assembles constitute fuel with fuel enrichments no greater than 4.95 w%. In the core designs proposed so far, cut-back factors of between {REDACTED} w% have been used in gadolinia doped fuel. Gadolinia concentrations are no greater than 10 w%.

4.3.1.5 Linear Heat Generation Rate Limits

Description

LHGR limits are applied to ensure plant performance during normal, frequent faults and infrequent faults are bounded by the safety analyses performed. Furthermore, rates are applied to minimise peak centreline temperature and improve margin to fuel melt.

Design Limit

The LHGR at which the fuel melts will be derived within the fuel rod design scope as a function of burnup for both UO_2 and $(U,Gd)O_2$ rods using the fuel temperature limits defined in Section 4.2.2.9 [DBC-1 to DBC-3i].

Design Bases

In normal operation, fuel rods should operate well below specified LHGR limits, to allow for load following and xenon transients as well as faulted conditions. In the case of RR SMR, reliance on control rods is higher than a typical borated plant which can consequently give greater local LHGR changes when rods are moved.

In the nuclear design basis, an analysis limit on F_Q shall be employed which will ensure that LHGRs are maintained within an acceptable range.

The limits will ultimately need to account for several important uncertainties including those resulting from the neutronics methods used, manufacturing tolerances, assembly/pin bowing, flow maldistribution and core calorimetry.

4.3.1.6 Shutdown Margins

Description

Attaining subcriticality must be assured upon shutdown from all operating modes and conditions. RR SMR uses control rods as the primary means of reactivity control and for shutdown. The subcriticality requirement is assessed using the SDM metric. A secondary means of shutdown is provided by the alternative shutdown function (ASF) [JD02], which injects boron in scenarios where the control rods do not actuate on a scram signal (or other scenarios where scram does not occur). The design basis for shutdown covers Mode 3 (hot standby) down to Mode 5 (cold shutdown).

After a power cycle the integrated head package (IHP) and then the RPV Upper internals are lifted, before all the fuel assemblies are unloaded from the core barrel and transferred to the spent fuel pool for storage. Prior to the next power cycle, the core barrel is initially empty and then loaded with fuel assemblies. The design basis for refuelling covers all Mode 6 within-RPV activities.

During all modes of operation, the core must be able to be continuously monitored via neutron detectors. For startup (Mode 2), particularly at the beginning of the first cycle when there are no fission products, installed neutron sources are being used to ensure a detectable signal even when the intrinsic neutron source is relatively low.

Design Limit

With all control rods fully inserted, SDM (with uncertainties) for the most reactive core state are greater than {REDACTED} pcm [DBC-1]. This applies for shutdown and refuelling modes of operation (Modes 3 to 6).

With one stuck rod (assumed to be fully removed from the core), SDM (with uncertainties) for the most reactive core state is greater than {REDACTED} pcm [DBC-2 to DBC-4]. This applies for all shutdown modes of operation (Modes 3 to 5). The same margin applies for refuelling (Mode 6) when there is a misload or misbuild fault.

During all modes, including startup and refuelling, the flux at the neutron detectors must be sufficient to enable continuous monitoring. The required minimum signal strength is to be quantified, as are the usage of in-core and ex-core detectors.

Design Bases – Shutdown and Hold Down

Given the RR SMR is boron free during normal operation, it is important to ensure sufficient SDM for the most reactive core state (cold-zero-power with no xenon). As a consequence, the entire power defect is already accounted for in the calculation so SDM predictions will be insensitive to core over-power and uncertainties relating to fuel power inlet temperature. SDM can therefore be treated in the same way as hold down for borated designs, and Mode 5 (cold shutdown) bounds Modes 3 and 4 (hot standby and hot shutdown). Therefore, in the nuclear design basis SDM is used as general term to cover both SDM and hold down margin.

The most reactive core state is taken to be cold-zero-power with no xenon. Cold is defined to be 4 °C as this maximises the water density and therefore its efficiency as a moderator. To ensure the results are demonstrably conservative, the analysis accounts for uncertainties in the k-eff prediction based on code validation, as well as an additional allowance for control rod depletion.

SDM is dependent on whether there is a stuck RCCA. The RCCA that is assumed stuck is determined analytically, each RCCA within the quarter core is withdrawn with all others remaining inserted, and the one assumed to be stuck is based on the maximum reduction in SDM. This determination is done for each statepoint being assessed. The worst stuck rod locations typically neighbour unrodded locations as this results in a larger contiguous uncontrolled fuelled region.

SDMs are calculated using SIMULATE5 at all points during the fuel cycle, for initial, transition and equilibrium cycles. Short and long preceding cycles are also assessed to ensure calculations bound variability in core burnup history.

A methods uncertainty of {REDACTED} pcm applies to SIMULATE5 k-effective calculations, with an additional {REDACTED} pcm to bound the effect of control rod depletion. Future work shall refine both aspects of this uncertainty [23] [24].

4.3.1.7 Core Stability

Stability to xenon oscillations has not currently been studied within the nuclear design basis. Due to the comparatively small size of the RR SMR core, axial and radial oscillations are not expected to be a concern. Further work shall be presented to demonstrate core stability in later phases.

4.3.2 Description

4.3.2.1 Nuclear Design Description

A detailed description of the nuclear design of the Reactor Core is provided in the Core Design Optimisation report [25], which describes the core design at Iteration 7. Details of the nuclear design assessments and criticality design assessments can be found in the Fuel and Core Performance Analysis Summary report [24]; however, given the iterative nature of the analysis this is based on Iteration 6 of the core design.

This section describes the 15 cycles modelled taking the RR SMR core from cycle 1 through to equilibrium. Bespoke assembly designs and loading patterns were developed for cycles 1, 2 and 3. The equilibrium core design was then applied to cycle 4 onwards until the core reached equilibrium.

As such, cycle 4 (and possibly 5) should be considered sub-optimal given no specific effort was spent in optimising either fresh fuel design, assembly placement or control rod sequencing. However, steady state performance was still satisfactory given the lack of optimisation performed.

General core parameters are shown in Table 4.3-1.

Table 4.3-1: General Core Operating Parameters

Parameter	Value
Thermal power	1358 MW(th)
Core mass	{REDACTED}
Active core height	2.8 m
Total control rods	{REDACTED}
Nominal power density	{REDACTED}
Average LHGR (incl. gamma heat)	{REDACTED}
Total thermal hydraulic flow (incl. bypass)	{REDACTED}
T_{inlet} at 100% power	{REDACTED}
Un-controlled fuel region height at with all rods inserted	{REDACTED}

4.3.2.2 Cycle 1 Core Design

A total of {REDACTED} unique assembly designs are utilised, see Figure 4.3-1. This is significantly more than later cycles where only {REDACTED} unique assembly designs are needed. However, it is important to note that the relatively steep change in fuel reactivity with time makes it very difficult to balance power across the core as the assemblies deplete whilst ensuring sufficient reactivity mid-cycle. This is due to:

- The lack of plutonium in the entire core
- The lack of any fission product at beginning of cycle (BOC).

{REDACTED}

Figure 4.3-1: Unique Assembly Designs Loaded in Cycle 1

The {REDACTED} fuel assembly types are arranged in the core as shown in Figure 4.3-1 (the serial ID (first line) is a unique assembly ID, the fuel type is given in parenthesis), with the aim to:

- Minimise power peaking throughout the cycle
- Ensure sufficient SDM with the highest worth rod fully removed (SDM-1SR).

In general, the locations that are nominally used for fresh fuel in subsequent cycles utilise fuel types {REDACTED}. These fuel types have higher gadolinia contents and are intended to mimic the behaviour of the fresh fuel assembly designs used in later cycles. Fuel in other locations have less gadolinia and lower enrichments; the intention is to mimic the behaviour of once and twice burnt fuel.

{REDACTED}

Figure 4.3-2: Cycle 1 Loading Pattern

In total, there are {REDACTED} unique segments used and 6 unique lattice arrangements. This results in {REDACTED} unique pin designs as shown in Figure 4.3-3 and

Figure 4.3-4. In Figure 4.3-3 the colours and the first number in each region indicate the fuel enrichment {REDACTED}. The pin numbers at the bottom of Figure 4.3-3 are used in

Figure 4.3-4 to indicate the {REDACTED} lattice designs for the cycle 1 core.

{REDACTED}

Figure 4.3-3: Unique Pin Designs Used in Fuel Loaded in Cycle 1

{REDACTED}

Figure 4.3-4: Location of the Unique Pins in each of the Unique Assemblies Loaded in Cycle 1

{REDACTED}.

It is important to note that the proposed cycle 1 core is stopped early at {REDACTED} GWd/tHM to improve the performance in subsequent cycles. Premature termination results in higher reactive fuel loadings for cycle 2, allowing higher enrichments for fresh fuel loaded fresh.

A total of {REDACTED}, This is done to optimise power peaking in the core due to gadolinia burn-up at different rates across the core. Rod sequencing for cycle 1 is shown in Figure 4.3-5 (the dashed dark lines indicate the range over which the banks are used during hot full power equilibrium xenon operation).

{REDACTED}

Figure 4.3-5: Cycle 1 Control Rod Sequencing Patterns.

The dashed vertical lines indicate the range control rods are expected to be inserted at hot full power with equilibrium xenon. Further optimisation is possible outside these ranges without affecting the full power base depletion calculation results summarised in this chapter (i.e. optimising control rod insertion to improve Worst Stuck Rod SDM and PCI concerns during startup).

It is apparent that before reaching a cycle burnup of {REDACTED} banks are utilised concurrently. However, it is believed this could be further refined (e.g. by moving each bank {REDACTED} at a time) such that only one bank is moved at a time. Beyond a cycle burnup of {REDACTED}, only one bank is ever utilised concurrently.

{REDACTED}

In all cases, only the SINCAD rods are used to manage excess reactivity during full power operation. At no point are the grey rods used given the intention was to use these during load following events. {REDACTED}. The rodded locations and RCCA types are shown in Figure 4.3-7 and Figure 4.3-8 and are valid for all cycles. In Figure 4.3-7, blue is unrodded locations, with red (1), green (2) and grey (3) being boron carbide, SINCAD and grey (control rod steel) RCCA designs respectively.

{REDACTED}

Figure 4.3-6: Axial Zoning for the Control Rod Types

{REDACTED}

Figure 4.3-7: Control Rod Types within the Core

{REDACTED}

Figure 4.3-8: Control Rod Groups

4.3.2.3 Cycle 2 Core Design

{REDACTED}.

Table 4.3-2 summarises the assemblies reloaded in the bottom right core quadrant. In general, the assemblies reloaded from cycle 1 have higher assembly reactivities as is evident by the 'End of cycle (EOC) k-infinity' values.

Table 4.3-2: List of Assemblies Reloaded in The Bottom Right Quadrant for Cycle 2

{REDACTED}

{REDACTED}.

Assemblies {REDACTED} (and their equivalent assemblies in the other three core quadrants) have relatively high reactivities given they are only utilised for a single cycle.

A total of {REDACTED} fresh fuel assemblies are loaded. This is significantly higher than later cycles and is due to the relatively low enriched fuel used in the preceding cycle.

{REDACTED}

{REDACTED}

Figure 4.3-9: Assembly Designs for Cycle 2

{REDACTED}

Figure 4.3-10: Loading Pattern for Cycle 2

The {REDACTED} assembly designs utilise {REDACTED} unique pin designs as shown in Figure 4.3-11. Like the previous cycle, the fuel pins {REDACTED}.

{REDACTED}

Figure 4.3-11: Fuel Assembly (left) and Pin Designs (right) for the Assemblies Loaded Fresh in Cycle 2 (the numbers shown in the pin designs (x.xx (y.y)))

{REDACTED} rod sequencing patterns have been used over the course of the {REDACTED} cycle.

4.3.2.4 Cycle 3 Core Design

{REDACTED} unique assembly types are loaded fresh in cycle 2 as illustrated in

Figure 4.3-12. A total of {REDACTED} fuel assemblies are loaded fresh.

{REDACTED}

{REDACTED}

Figure 4.3-12: Loading Pattern (Left) and {REDACTED}

{REDACTED}.

{REDACTED}. These are coloured light-grey in

Figure 4.3-12. The fuel assemblies chosen had relatively high reactivity and helped promote power to regions that would otherwise have been low. This flattened the radial power profile resulting in improved $F_{\Delta H}$ and F_q peaking factors.

A total of {REDACTED} unique fuel pin designs are used across the {REDACTED} assembly types loaded fresh in cycle 3 (see Figure 4.3-13). Note that the gadolinia doped fuel pins {REDACTED} (as for cycle 1). This was done to increase BOC core reactivity. As a result of this layout, additional lower enriched fuel pins were required to control within-assembly peaking {REDACTED}. Locations particularly susceptible include:

- Pin locations surrounded by a large number of gadolinia doped fuel pins
- The corner pin (possibly due to increased moderation from the assembly gap)
- Locations directly neighbouring a guide tube or instrumentation tube

Similar to previous cycles, the fuel pins neighbouring the central instrumentation tube contain no gadolinia.

{REDACTED}

Figure 4.3-13: Assembly (Left) and Pin Designs (Right) for Fuel Loaded Fresh in Cycle 3

Throughout cycle 3, {REDACTED} of {REDACTED}.

4.3.2.5 {REDACTED} (and equilibrium) Core Design

{REDACTED}

{REDACTED}

Figure 4.3-14: Loading Pattern (Left) and {REDACTED}

The {REDACTED} fuel assembly designs loaded fresh in {REDACTED} onwards are shown in Figure 4.3-15. Across these {REDACTED} assembly designs, {REDACTED} unique fuel pins are used. For these fuel designs, fuel pins directly neighbouring guide tubes contain no gadolinia to:

- Improve control rod-worth
- Slightly prolong gadolinia burnout

Prolonging gadolinia burnout improves BOC reactivity whilst also reducing the amount of control rod insertion required mid-cycle.

Similar to previous cycles, fuel pins neighbouring the central instrumentation tube contain no gadolinia to help improve in-core instrumentation response.

{REDACTED} control rod sequencing patterns are used throughout the cycle, with changes occurring after cycle-burnups of {REDACTED} full power months (EFPM)).

{REDACTED}

Figure 4.3-15: Assembly (Left) and Pin Designs (Right) for Fuel Loaded Fresh in Cycles 4 Onwards

4.3.3 Nuclear Performance

This section provides information on the performance of the Reactor Core against the nuclear design basis. The performance shown is representative of the core design at iteration 6. Further improvements have been made to the nuclear performance in Iteration 7; however, only limited parts of this analysis have been formally issued as part of the GDA process.

4.3.3.1 Peak Discharge Burnups

For a nominal cycle the peak pin and peak pellet burnups are {REDACTED} GWd/MT respectively. With a long preceding cycle, the peak pin and peak pellet burnups are {REDACTED} GWd/MT respectively.

{REDACTED}.

{REDACTED}.

Currently burnup uncertainty is only captured as part of the long and short preceding cycles approach. This shall be investigated further in conjunction with the fuel vendor and shall be the subject of future core design optimisation.

4.3.3.2 Reactivity Coefficients

Reactivity Coefficients – Fuel Temperature (Doppler)

The fuel temperature (Doppler) coefficient represents the reactivity change for each degree change in the fuel temperature.

Doppler feedback involves specific capture resonances (primarily in ^{238}U and plutonium nuclides) that broaden with increasing fuel temperature. Since an increase in fission rate leads to an almost immediate increase in fuel temperature, negative feedback is extremely rapid and important in rapid reactivity insertion accidents such as rod ejection. It is essential the minimum Doppler coefficient remains suitably negative to minimise the amount of energy deposition in the fuel during such infrequent faults. Figure 4.3-16 demonstrates the change of the fuel temperature coefficient with the cycle burnup for a number of sensitivity studies representing Lower Study Limits (LSL) and Upper Study Limits (USL), which account for changes in core reactivity from preceding cycle lengths and core power.

{REDACTED}

Figure 4.3-16: Fuel Temperature (Doppler) Coefficient for All Defined Initial Conditions

The Doppler coefficient shows a minor dependence on burnup, due to changes to fuel temperature and the increasing importance of the ^{240}Pu resonance.

With uncertainties included, the least negative Doppler coefficient observed is {REDACTED}. This together with the low sensitivity to burnup and operating conditions demonstrates that the requirement for a negative fuel temperature coefficient of reactivity is met across all DBC-1 conditions.

Reactivity Coefficients – Moderator Temperature

Figure 4.3-17 shows the change of the moderator temperature coefficient with the cycle burnup for a range of study limits, before uncertainties are applied. The results show that the MTC is always negative and in general gradually reduces (i.e. becomes more negative) as the cycle progresses, as a result of plutonium production hardening the neutron spectrum.

With uncertainties included, the least negative MTC observed is {REDACTED}. This demonstrates that the requirement for a negative MTC is met across all DBC-1 conditions.

{REDACTED}

Figure 4.3-17: Moderator Temperature Coefficient (MTC) for All Defined Initial Conditions

4.3.3.3 Power Distribution

The distribution of power within the core is a key criterion for the safe and efficient operation of the reactor. The power distribution must facilitate adequate thermal margins such that fuel integrity can be demonstrated under DBC-1 and DBC-2 conditions.

The current reactor physics design basis bounds all DBC-1 conditions; further analysis with an updated design basis shall be required to ensure that power distributions under DBC-2 conditions are adequate.

Several parameters are used to describe the complex nature of the power distribution in concise limits that can be analysed and optimised. The key parameters are the power peaking factors Fq and $F_{\Delta H}$.

The heat flux hot channel factor (F_Q) is also referred to as the total peaking factor or peak pin relative linear power density. This parameter is the ratio of the maximum local linear power density to the core average local linear power density. Therefore, F_Q is a measure of the maximum Linear Heat Generation Rate (LHGR) anywhere in the core. The peak F_Q is linked to the maximum fuel pin cladding temperature, and therefore limits on F_Q must be defined (in consultation with the fuel vendor) to prevent clad failure.

The nuclear enthalpy rise hot channel factor ($F_{\Delta H}$) is the ratio of the maximum integrated pin power within the core to the average pin power. Therefore, $F_{\Delta H}$ is a measure of the maximum total power produced in any fuel pin in the core. The peak F_{Δ} is linked to the capability of the core and plant systems to provide sufficient cooling, and therefore limits on $F_{\Delta H}$ must be defined to prevent DNBR.

Maximum F_Q for the nominal full power equilibrium xenon depletion is {REDACTED}, while $F_{\Delta H}$ peaks at {REDACTED} (without uncertainties).

Across all cases, F_Q and $F_{\Delta H}$ peak at {REDACTED} respectively (without uncertainties). It is important to note that the control rod group sequences across the equilibrium cycle have only been optimised for the nominal full power equilibrium xenon depletion, therefore further optimisation for off-nominal conditions such as short and long cycles would be expected to yield improved power peaking results.

Applying total combined uncertainties to these values gives the worst-case values as summarised in Table 4.3-3. Note that the thermal hydraulic analysis described in Section 4.4 currently assumes more conservative power peaking which will be made consistent in future issues of these analyses.

Table 4.3-3: Power Peaking Factors

Power Peaking Factor	Peaking Factor no Uncertainty		Peaking Factor + Combined Uncertainty	
	Nominal	Study Limits	Nominal	Study Limits
F_Q	{REDACTED}	{REDACTED}	{REDACTED}	{REDACTED}
$F_{\Delta H}$	{REDACTED}	{REDACTED}	{REDACTED}	{REDACTED}

4.3.3.4 Shutdown Margin

Shutdown margins are calculated for "full" shutdown where all control rods are inserted, and "worst stuck rod" shutdown where the highest worth rod is assumed to be stuck fully withdrawn.

Figure 4.3-18 shows the evolution of the full shutdown and worst stuck rod shutdown margins as a function of burnup from the initial cycle through to an equilibrium. Each dashed vertical line in the figure represents a refuel cycle. In this analysis only Cycles 1 to 3 have been optimised along with an equilibrium cycle which is introduced from Cycle 4 onwards. As Cycle 4 has not been optimised at this time, the results from Cycle 4 until an equilibrium is reached (approximately Cycle 7) are not considered representative.

When including uncertainties of {REDACTED} pcm, Figure 4.3-18 shows that the {REDACTED} pcm limit for full shutdown and the {REDACTED} pcm limit for worst stuck rod are met for all optimised cycles, at all times through cycle and for the most limiting conditions.

{REDACTED}

Figure 4.3-18: Through-cycle Shutdown Margins

4.3.4 Analytical Methods

4.3.4.1 Code Selection

The primary analysis codes in the nuclear design and analysis for the RR SMR are the CMS suite, which are state of the art neutron transport codes created by Studsvik. The latest edition, CMS5, primarily consists of the CASMO5 two-dimensional lattice physics code and the SIMULATE5 three-dimensional analytical nodal code. The suite of codes is used to perform neutronic analysis with coupled thermal hydraulic feedback needed for the design, optimisation and safety analysis of nuclear Reactor Cores. The codes are well validated and used across the industry for many operational PWRs.

The core modelling in CMS5 and all key modelling assumptions are described in [27].

For criticality modelling, Monte-Carlo N-Particle (MCNP) Version 6.2 is used. MCNP is widely used in the nuclear industry in the United Kingdom and around the world for both criticality and radiation physics calculations. As many of the design features of the fuel assembly and storage rack can be explicitly represented, the sensitivity of the k-effective to uncertainties in parameters due to mis-manufacture, such as fissile loading and fuel rod pitch, can be modelled in an MCNP model.

The criticality modelling performed in MCNP and all key modelling assumptions are described in [28].

Both CMS5 and MCNP methods use an ENDF/B Version VII.1 based nuclear data library.

4.3.4.2 Validation Status

The validation status of the reactor physics methods and criticality methods are summarised in [29]. Appropriate margins for uncertainties in the analytical methods have been quantified and are included in supporting analyses.

4.4 Thermohydraulic Design

4.4.1 Design Bases

The thermal hydraulic design basis covers DNB sub-channel methods for steady state as well as faulted conditions. The design basis also covers corrosion and hydraulic aspects such as assembly hold-down force, fuel rod bow and assembly bow.

4.4.1.1 Code

The VIPRE-01 analysis software was developed primarily based on the COBRA family of codes by Battelle Pacific Northwest Laboratories for the Electric Power Research Institute (EPRI). The code is used to evaluate nuclear reactor parameters, including minimum DNBR, critical power ratio (CPR), fuel and cladding temperatures, and reactor coolant state, in normal and off-normal conditions. Further overview on the codes used is given in section 2.2 of the Fuel and Core Design Basis Summary [21].

4.4.1.2 Departure from Nucleate Boiling

Description

DNB is reached at critical heat flux (CHF) or boiling crisis which describes the thermal limit where the bubbly density from nucleate boiling in the boundary layer of a fuel rod is so great that adjacent bubbles coalesce and form a vapour film on the surface of the rod. Heat transfer across this vapour is relatively low compared to the coolant resulting in a marked increase of the cladding surface temperature leading to rapid oxidation or even melting of the cladding. This can result in fuel failure.

In PWRs, departure from nucleate boiling is reflected in DNBR, the ratio of CHF to the local heat flux of a fuel rod. These ratios incorporate margin into the phenomena.

A key safety requirement is that DNB will not occur during steady state operation, normal operational transients, and anticipated operational occurrences (AOO's; DBC-1, DBC-2; Design Basis Condition). EUR safety requirements have been used to define the approach towards an acceptable number of rod failures during DBC-3 and DBC-4 events.

Design Limit

The design limits for DNBR are based on the CHF correlations used in various assessments. Two CHF correlations are employed during Step 2 for the RR SMR design; Westinghouse W-3S and EPRI-1.

- Where the W-3S correlation is employed, the minimum DNBR limit shall be {REDACTED}. [DBC-1 to DBC-3i].

- Where the EPRI-1 correlation is employed, the minimum DNBR limit shall be {REDACTED} [DBC-2 to DBC-3i].

- DBC-3ii events shall be assessed to the same DNBR limits as above with up to 5 % of rods allowed to exceed the limits, with an appropriate ALARP case [DBC-3ii].

- DBC-4 events shall be assessed to the same DNBR limits as above with up to 10 % of rods allowed to exceed the limits, with an appropriate ALARP case [DBC-4].

- The design shall prevent hydrodynamic instabilities (static or dynamic), that degrade DNBR performance, from occurring [DBC-1 to DBC-4].

- A DNB propagation limit on peak clad temperature that prevents clad ballooning shall be explored with the fuel vendor.

Design Bases

Computational Fluid Dynamics

Computational Fluid Dynamics (CFD) analysis is performed for the reactor plant, including the RPV and RCS. The RPV analysis is key to understanding the flow distribution at the bottom of the core. It is intended that CFD analysis provides input data for the VIPRE-01 sub-channel analysis, rather than directly assessing any of the design limits.

CFD is also used to assess design options for components such as flow inlet devices; this is excluded from the scope of this document.

Steady State

VIPRE-01 is used for carrying out DNB assessments based on boundary conditions derived from RELAP5-3D analysis (DBC-2 to DBC-4) or using directly inputted steady-state plant assumptions (DBC-1).

VIPRE-01 assessments are governed by a design basis methodology that covers uncertainties in plant parameters and phenomena such as flow maldistribution. Care is taken not to double count uncertainties across VIPRE-01 and RELAP5-3D analyses.

A summary of the selection of inputs in section 2.3.13 of the Fuel and Core Design Basis Summary [21].

Transient/Faulted Conditions

All transient/faulted mass flux data is penalised by {REDACTED} % to bound the potential for flow maldistribution effects. For each transient/fault, sensitivity tests of VIPRE-01 minimum DNBR predictions are performed. The sensitivity tests involved the examination of the impact of all possible parameter permutations for those parameters outlined in section 2.3.15 of the Fuel and Core Design Basis Summary [21].

A full justification of all parameter choices for the above transient models is contained within reference [30], as well as an extended description of how the design basis is applied to complete loss of flow (CLOF) faults.

4.4.1.3 CRUD Deposition, Nucleate Boiling and CRUD Induced Localised Corrosion

Description

CRUD build-up on nuclear fuel rods is common in light water reactors due to the typical operating conditions. CRUD build-up can lead to the CRUD induced localised corrosion (CILC) phenomena occurring. CILC is an accelerated corrosion of the zirconium cladding caused by CRUD-induced cladding temperature increases, CRUD thickness, and enhanced corrosion due to water chemistry effects.

RR SMR plans to use Potassium as a pH raiser in the primary coolant. Potassium has previously been studied in the presence of Boron; further work is planned to better understand the corrosive effects of Potassium chemistry and allowable clad concentrations.

Design Limit

A specific design limit for CRUD deposition is not thought to be necessary so long as fuel performance calculations use a bounding CRUD thickness in their calculations. This will be confirmed with the fuel vendor.

A nucleate boiling limit of {REDACTED} % (maximum void fraction) during normal operations for the nominal fuel channel shall be used to reduce CILC risk [DBC-1, DBC-2].

Design Bases

To assess the CILC risk, the VIPRE-01 sub-channel analysis solver is used to generate so-called steaming rates (effectively the mass flux of steam in the core) for each assembly, through several representative cycles. The steaming rates will be assessed using a code such as BOA, which can assess the CILC risk for assemblies that experience higher steaming rates.

Steaming rates have been calculated based on Iteration 5 physics data for a single cycle, noting that steaming rates are not expected to significantly change for further core iterations. The steaming rates show that most assemblies do not see any appreciable steaming, with peak steaming rates ~30-50 % below those of a typical PWR average steaming rate. RELAP5-3D shall be used to determine the nominal channel maximum void fraction.

The void fraction in the bulk boiling region is predicted by using homogeneous flow theory and assuming no slip. The void fraction in this region is therefore a function only of the thermodynamic quality [15].

Studies have been performed to determine the sensitivity of the minimum DNBR to the void fraction correlation and the inlet flow distributions. The results of these studies show that the minimum DNBR is relatively insensitive to variation in these parameters. Furthermore, the VIPRE-01 flow field model for predicting conditions in the hot channels is consistent with that used in the derivation of the DNB correlation limits, including void and quality modelling, turbulent mixing, crossflow, and two-phase flow [15].

It is noted that the term 'quality' (liquid, vapour, void, true, etc.) used within the thermal hydraulic design sections of this chapter relate to the physics and engineering (non-nuclear) thermodynamic quantity.

4.4.1.4 Assembly Hold-down Force

Description

The fuel assembly is designed to be installed vertically in the reactor pressure vessel and stand upright on the lower core support plate, fixed in place via alignment pins. The process is repeated at the top of the core, where the upper core support plate bears downward force against the top nozzle through hold-down springs. The spring force balances prevention of assembly lift-off (too little compression) against fuel rod bow and guide tube effects (too much compression).

Design Limit

The assembly hold-down force shall be sufficiently high as to prevent enough lift-off force that could unseat the lower fuel assembly tie-plate from the fuel support structure [DBC-1, DBC-2]. The assembly hold-down force shall be limited such that assembly bow is prevented [DBC-1, DBC-2]. This range shall be quantified in conjunction with the fuel vendor and reported in a future update.

Assembly guide tube material selection shall consider compressive forces, corrosion, and hydrogen pickup effects [DBC-1, DBC-2].

Design Bases

The assessment approach will be defined in detail with the fuel vendor. This is expected to include detailed optimisation of key fuel assembly components such as hold-down springs and guide tubes, to meet the above design limits over the course of the life of an assembly. The assessments will also consider the effects of reactor coolant pump overspeed conditions (DBC-2). For the RR SMR, the mechanical flow rate is currently assumed to be {REDACTED} percent greater than the best estimate value [31].

{REDACTED} [15].

{REDACTED} [15].

REDACTED} [15].

4.4.1.5 Fretting Wear

Description

Grid to rod fretting arises due to the high coolant velocity through the spacing grid causing it to vibrate and rub against the fuel element. This can result in significant wear to the rod and can cause fuel failure. A secondary mode of fretting wear can occur due to rod growth where individual rods can grow faster than the assembly average.

Design Limit

A design limit of {REDACTED} due to fretting wear will be applied [DBC-1, DBC2].

Design bases

{REDACTED} [15]:

- {REDACTED}
- {REDACTED}
- {REDACTED}
- {REDACTED}
- {REDACTED}
- {REDACTED}

{REDACTED} [15].

4.4.1.6 Rod Bow

Description

Fuel rod dimensional changes and distortion (summarised as "Rod Bow") occur during the normal operation of PWRs. Fuel rod length increases because of irradiation growth of the cladding and axial strain produced by mechanical interaction of the fuel pellets with the cladding.

{REDACTED}.

Rod bow has a direct impact on DNBR (as well as rod powers) since channel closure is reduced, leading to lower available volume for heat transfer. Since rod bow can occur anywhere in the core, the effect must be accounted for explicitly in DNBR calculations. Rod bow will be pessimistically applied at this stage in the project as a non-burn-up dependent limit. This will be covered in future reports on Reactor Core thermal hydraulics.

Design Limit

A design limit for rod bow will be decided in concert with a fuel vendor during detailed design. This limit will take the form of an allowable limit on deflection (and growth, if appropriate), which will ensure that the rod bow penalty on DNBR is minimised [DBC-1, DBC-2].

Design Bases

Rod bow assessments are partially based on operational experience and analytical modelling. Modelling of the phenomena can be performed using creep models that examine the behaviour of fuel rods through cycle. Fuel vendors maintain assessment tools and methodologies that can be used to carry out such assessments as required. Since the fuelled length of the RR SMR core is shorter than a standard core, a bespoke assessment will be required.

Fuel vendor analyses and operational experience shall be used to select optimal design parameters and tolerances in order to minimise the potential for rod bow. For instance, grid parameters shall be chosen to minimise rod bow, ensure limits on channel closure are not exceeded, and the inclusion of any additional non-structural grids shall not increase rod bowing [15].

{REDACTED}.

Rod bow is covered in DNB analysis using VIPRE-01 through a rod bow DNBR penalty based on the allowable deflection.

4.4.1.7 Assembly Bows

Description

Assembly bow is the global distortion of an entire fuel assembly such that it does not resemble its original, ideal form, causing a reduction in inter assembly gaps.

Assembly bow, particularly in extreme cases, can cause significant shifts in reactor power due to water gap changes. Such perturbations lead to uncertainty factors being required for key nuclear parameters, e.g. $F_{\Delta H}$ and F_Q (Nuclear Enthalpy Rise Hot Channel Factor and Heat Flux Hot Channel Factor respectively).

Another potential impact of extreme cases of assembly bow on the safety case is incomplete control rod insertion upon reactor trip, which would result in a reduction in available shutdown margin.

Design Limit

A design limit for assembly bow will be decided in concert with the fuel vendor. This, like rod bow, will likely take the form of an allowable limit on deflection to minimise the uncertainty on inputs to the CMS5 suite (CASMO5 and SIMULATE5) that calculate $F_{\Delta H}$ and F_Q, and to ensure control rod insertion upon reactor trip [DBC-1 to DBC-4].

Design Bases

The relative resistance of the design to assembly bow is evaluated through comparison with existing fuel designs with known assembly bow characteristics.

The RR SMR reactor fuel assemblies shall be built with a high degree of resistance to assembly bowing, thus minimising or eliminating the potential for any impact on safety margins. It is expected that the RR SMR design can benefit from fuel assembly design features such as the following used on AP1000 [15] that provide resistance to fuel assembly bow:

- Increased thickness of the guide tube wall
- Changing the dashpot region of the guide tube to a tube-in-tube design, effectively further increasing the guide tube wall thickness in this critical region
- Implementing ZIRLO™ as the grid and guide tube material to reduce neutron fluence induced growth.
- Optimised top nozzle hold-down spring forces

Given the short length of the RR SMR fuel, it is anticipated that the fuel assembly bow performance will be even better than has been experienced for traditional fuel.

In the event that the RR SMR plant fuel assembly bow measurements are inconsistent with existing performance expectations, detailed mechanical and nuclear methods are available for modelling the effects of assembly bow on the core power distribution. Any potential peaking factor penalties would be expected to be small and available margins could be reallocated to accommodate these small penalties without affecting the conclusions of the RR SMR plant safety case.

To mitigate against the potential for incomplete control rod insertion upon reactor trip, a surveillance programme will be conducted to confirm the dimensional stability of the fuel assembly. This programme will include pre- and post-irradiation measurements of fuel assembly length (to determine growth) and fuel assembly bow and twist. These measurements will be conducted on assemblies expected to have the highest burnups. In addition, Rod Cluster Control Assembly (RCCA) drag force and RCCA drop time will be measured. Any unusual results that might indicate fuel assembly distortion beyond the expected limits will be evaluated as necessary.

4.4.2 Description of the Thermal and Hydraulic Design of the Reactor Core

4.4.2.1 Critical Heat Flux Ratio or Departure from Nucleate Boiling Ratio and Mixing Technology

The original W-3 CHF correlation is one of the CHF correlations contained in the US NRC approved generic version of VIPRE-01 [32] summarises the applicability and the ranges of validity for VIPRE-01/W-3 CHF correlations when applied to the RR SMR core. For the single assembly model, the specific W-3S (correlation with simple grid factor) CHF correlation was selected from the various options available with the general W-3 correlation because it accounts for mixing grids, but treats them as a generic type (unlike W-3C which does not include grids or non-uniform axial power shape adjustments, or W-3L that is used for a specific grid type). The standard W-3 correlation limit of 1.3 has been applied for the range shown in Table 4.4-1.

Table 4.4-1:Summary of W-3 Correlation Range and RR SMR Nominal Conditions

	Pressure (MPa)	**Mass Flux ($kg/m^2.s$)**	**Equilibrium Quality**	**Heated Length (m)**	**Inlet Enthalpy (KJ/kg)**	**Hydraulic Diameter (cm)**	**Grid Factors**
W-3	{REDACTED}	{REDACTED}	{REDACTED}	{REDACTED}	{REDACTED}	{REDACTED}	{REDACTED}
EPRI-1	{REDACTED}	{REDACTED}	{REDACTED}	{REDACTED}	{REDACTED}	{REDACTED}	{REDACTED}
RR SMR	{REDACTED}	{REDACTED}	{REDACTED}	{REDACTED}	{REDACTED}	{REDACTED}	{REDACTED}

All the nominal fuel design and core conditions are within the range of W-3S CHF correlation applicability, as are the edges of the normal operating band based on the most up to date understanding of uncertainties. In some faulted situations, e.g. loss of flow faults, EPRI-1 is employed to calculate the DNBR. EPRI-1 has a calculated correlation limit of {REDACTED} based on the test data used to derive the correlation and the Owen criterion.

4.4.2.2 Power Distribution

The 17x17 Assembly Model, described Steady-State Thermal Hydraulic Assessments Report [33] is developed in a conservative manner so as not to represent a cycle specific assembly. The application of physics power shapes is done in line with a limiting assembly across the equilibrium core cycle for the iteration 5 core. This will be updated for future core iterations, however it is expected to bound data from iterations 7 and 8 due to power shape optimisation. The radial power distribution is taken from the limiting assembly and there are several sub-channels in the model bounded by fully heated rods that have {REDACTED} higher powers than the average for the assembly.

The radial power peaking parameter is defined to be the enthalpy rise hot channel factor ($F_{\Delta H}$). This parameter is the ratio of the maximum integrated rod power within the core to the average rod power. Therefore, $F_{\Delta H}$ is a measure of the maximum total power produced in a fuel rod. The $F_{\Delta H}$ limit is representative of the hottest fuel rod and the sub-channel connected to this rod is required by the thermal margin acceptance criteria to have the limiting DNBR.

During operations, $F_{\Delta H}$ will be limited to a specific value that is bounded by design calculations. In the case of the iteration 6 core, the baseline design value for $F_{\Delta H}$ is {REDACTED}; however, this is increased with uncertainties to {REDACTED} during analyses for conservatism in line with the Deterministic analysis approach.

The total peaking factor (F_Q) is also referred to as the heat flux hot channel factor. This parameter is the ratio of maximum local heat flux on the surface of a fuel rod to the average fuel rod heat flux. The maximum F_Q value is used to calculate the peak linear heat generation rate. The design F_Q acceptance criterion has not yet been determined.

Operationally, there will be no limit on axial peaking because (i) the other limits of F_Q and $F_{\Delta H}$ enforce a sufficiently flat power distribution, and (ii) axial peaking is treated in a multi-step approach involving operational restrictions and analysis that will be subsequently described. A rod (not core average) axial peaking factor (F_z) is defined as the maximum relative power at any axial point for a unique rod divided by the average power throughout the fuel rod.

The axial peaking factors used more often than the rod peak in sub-channel analysis are that of the core average axial peaking factor (F_z^{Core}) and an assembly average axial peaking factor (F_z^{Assm}), because when used in combination with the $F_{\Delta H}$ value it is conservative. Similarly, another key axial power parameter is the axial offset (AO), which is a measure of the difference in power generated in the top and bottom halves of the core.

Axial offset alone is not enough of an indicator for the axial power shape. Two axial power shapes with different peak F_z^{Core} values and peak locations can have the same AO percentage if each half of the core produces the same power. Therefore, the DNBR for two axial power profiles with the same AO can be quite different.

At this stage in the design, an F_z value of {REDACTED} has been selected to bound all physics conditions; however, this is expected to be relaxed to a more appropriate value following the conclusion of uncertainty quantification work in the physics area. The combination of an F_z and $F_{\Delta H}$ of {REDACTED} and {REDACTED} respectively {REDACTED} with the modelling approach described above will provide a conservative estimate of the DNBR. A more realistic estimate of the maximum F_z (core average, to be consistent with a core average $F_{\Delta H}$) value is {REDACTED} based on iteration 6, which would improve margins by approximately 10 %. The highest F_Q value is highly unlikely to coincide with the highest F_z value, which ensures additional conservatism in the calculation.

Typical F_Q values for other PWRs are of the order of {REDACTED} for so-called 'limiting condition of operation' assessments (Reference [34]), which typically use a LOCA-based F_Q limit; however, normal operations are expected to be well below this. Some reactors appear to use higher F_Q limits for rapid faults (e.g. Evolutionary Power Reactor) so this will be investigated further as the design programme progresses.

VIPRE-01 allows for axial power shapes to be implemented as middle peaked (chopped cosine), top peaked (μ sin μ) or a custom shape based on user inputs. Comparisons between the two power shapes showed that, for the same F_z, top peaked power shapes are approximately 4 % more limiting than middle-peaked shapes. Examination of CHF test data has shown that test sections with top peaked power shapes are typically run with lower F_z values than the {REDACTED} selected here [35]. This suggests that the combination of high F_z and top peaked is unlikely to be found in practice, therefore a middle-peaked assumption is used as the baseline for the design basis. Detailed power shape analysis will be carried out in further design work to confirm this assumption.

Future revisions of this chapter will provide significantly more detail towards the approach of selecting appropriate F_Q, $F_{\Delta H}$ and F_z values and their application to more complex VIPRE-01 input files.

Pellet Diameter, Density, and Enrichment

Variations in pellet diameter, density, and enrichment will be described in Version 3 of the generic E3S Case.

Inlet Flow Maldistribution

A DB of {REDACTED} percent reduction in coolant flow to the hot assembly is used in the VIPRE-01 analyses. Previous studies have shown that flow distributions significantly more non-uniform than {REDACTED} percent have a very small effect on DNBR.

Flow Redistribution

The flow redistribution accounts for the reduction in flow in the hot channel resulting from the high-flow resistance in it due to the local or bulk boiling. The effect of the non-uniform power distribution is inherently considered in the VIPRE-01 analyses for every operating condition evaluated.

Flow Mixing

The subchannel mixing model incorporated in the VIPRE-01 code and used in reactor design is based on experimental data. The mixing vanes incorporated in the spacer grid design induce additional flow mixing between the various flow channels in a fuel assembly as well as between adjacent assemblies. This mixing reduces the enthalpy rise in the hot channel resulting from local power peaking or unfavourable mechanical tolerances.

Effects of Rod Bow on DNBR

Effects of Rod Bow on DNBR are covered in section 4.4.1.6.

4.4.2.3 Core Pressure Drops and Hydraulic Loads

The core pressure drop includes those in the fuel assembly, lower core plate, and upper core plate. The full power operation pressure drop values are the unrecoverable pressure drops across the vessel, including the inlet and outlet nozzles, and across the core. Since the best estimate flow is that flow most likely to exist in an operating plant, the calculated core pressure drops are based on this best estimate flow rather than the thermal design flow.

The fuel assembly hold-down springs are designed to keep the fuel assemblies in contact with the lower core plate under normal operation and frequent faults. The hold-down springs are designed to tolerate the possibility of an over-deflection associated with fuel assembly lift-off for this case and to provide contact between the fuel assembly and the lower core plate following this transient. More adverse flow conditions occur during a LOCA.

Hydraulic loads at normal operating conditions are calculated considering the best estimate flow, accounting for the minimum core bypass flow based on manufacturing tolerances. Core hydraulic loads at cold plant startup conditions are based on the cold best estimate flow but are adjusted to account for the coolant density difference. Conservative core hydraulic loads for a pump overspeed transient are considered.

4.4.2.4 Correlation and Physical Data

Forced convection heat transfer coefficients are obtained from the Dittus-Boelter correlation with the properties evaluated at bulk fluid conditions. This correlation has been shown to be conservative for rod bundle geometries with pitch-to-diameter ratios in the range used by PWRs. The onset of nucleate boiling occurs when the clad wall temperature reaches the amount of superheat predicted by Thom’s correlation.

The analytical model used in the VIPRE-01 code and the experimental data used to calculate the pressure drops are described below. Unrecoverable pressure losses occur as a result of viscous drag (friction) and/or geometry changes (form) in the fluid flow path. The flow field is assumed to be incompressible, turbulent, single-phase water. Those assumptions apply to the core and vessel pressure drop calculations for the purpose of establishing the primary loop flow rate. Two-phase considerations are neglected in the vessel pressure drop evaluation because the core average void is negligible. Two-phase flow considerations in the core thermal subchannel analysis are considered in the calculation of the core and vessel pressure losses.

Fluid density is assumed to be constant at the appropriate value for each component in the core and vessel. Because of the complex core and vessel flow geometry, precise analytical values for the form and friction loss coefficients are not available; therefore, experimental values for these coefficients are obtained from geometrically similar models.

Tests of the primary coolant loop flow rates are made prior to initial criticality to verify that the flow rates used in the design, which are determined in part from the pressure losses calculated by the method described here, are conservative.

VIPRE-01 considers two-phase flow in two steps: first, a quality model is used to compute the flowing vapour mass fraction (true quality), including the effects of subcooled boiling; then given the true void quality, a bulk void model is applied to compute the vapour volume fraction (void fraction).

VIPRE-01 uses a profile fit model for determining subcooled quality. It calculates the local vapour volumetric fraction in forced convection boiling by: 1) predicting the point of bubble departure from the heated surface and 2) postulating a relationship between the true local vapour fraction and the corresponding thermal equilibrium value.

The void fraction in the bulk boiling region is predicted by using homogeneous flow theory and assuming no slip. The void fraction in this region is therefore a function only of the thermodynamic quality.

4.4.2.5 Uncertainties in Estimates

Key parameter uncertainties have been defined in Table 4.4-2, which represent estimates of 2σ or the 95th percentile in each case. Using the 2σ approach is consistent with other PWR design bases and the approach to correlation limits.

Table 4.4-2: Key Plant Parameter and Physics Uncertainties

Parameter	Nominal Value	Uncertainty	Conservative Direction
Pressure [MPa]	{REDACTED}	{REDACTED}	Negative
Inlet Temperature [K]	{REDACTED}	{REDACTED}	Positive
Power [MWth]	{REDACTED}	{REDACTED}	Positive
Mass Flux [kg/m^2.s]	{REDACTED}	{REDACTED}	Negative
$F_{\Delta H}$	{REDACTED}	{REDACTED}	Positive
Inlet Flow Distribution	-	{REDACTED}	Negative

Pressure, temperature, power, and flow uncertainties have been based on Reference [36], which discusses the safety justification of a PWR in the US (North Anna Units 1 and 2). Consultation with other industry experts concluded that those values are typical of operating PWRs and are appropriate for performing bounding steady-state assessments. Pressure, temperature, and flow uncertainties have been increased slightly over those in [36] to ensure that adequate conservatism has been used whilst instrumentation and control equipment is finalised.

The uncertainty on $F_{\Delta H}$ has been selected based on a bounding {REDACTED} % power uncertainty calculated in a the Assessment of the Impact of Manufacturing Uncertainties on Power Shapes Report [37] on the impact of manufacturing uncertainties on power shapes, (FEΔH, engineering uncertainty), and the {REDACTED} % value discussed in [36] (FNΔH, nuclear uncertainty). {REDACTED}. This uncertainty will be re-examined following further physics analysis and adjusted accordingly to account for physics modelling, manufacturing uncertainties and power imbalance effects.

The flow distribution at the inlet of the core is expected to be non-uniform, with some assemblies seeing slightly higher inlet flows than others. The flow distribution device will be optimised to minimise the degree of asymmetry in inlet flows, with a target (including uncertainty) maximum reduction from the average flow of {REDACTED} %.

4.4.2.6 Flux Tilt Considerations

Significant quadrant power tilts are not anticipated during normal operation since this phenomenon is caused by some asymmetric perturbation. A dropped or misaligned RCCA could cause changes in hot channel factors, this is described in E3S Case Version 2, Tier 1, Chapter 9A: Auxiliary Systems [3].

Other possible causes for quadrant power tilts include X-Y xenon transients, inlet temperature mismatches, enrichment variations within tolerances, and so forth.

In addition to unanticipated quadrant power tilts as described above, other readily explainable asymmetries may be observed during calibration of the ex-core detector quadrant power tilt alarm. During operation, in-core maps are taken at least one per month and additional maps are obtained periodically for calibration purposes. Each of these maps is reviewed for deviations from the expected power distributions.

Asymmetry in the core, from quadrant to quadrant, is frequently a consequence of the design when assembly and/or component shuffling and rotation requirements do not allow exact symmetry preservation. In each case, the acceptability of an observed asymmetry, planned or otherwise, depends solely on meeting the required accident analyses assumptions. In practice, once acceptability has been established by review of the in-core maps, the quadrant power tilt alarms and related instrumentation are adjusted to indicate zero quadrant power tilt ratio as the final step in the calibration process. This action confirms that the instrumentation is correctly calibrated to alarm if an unexplained or unanticipated change occurs in the quadrant-to-quadrant relationships between calibration intervals.

Proper functioning of the quadrant power tilt alarm is significant. No allowances are made in the design for increased hot channel factors due to unexpected developing flux tilts, since likely causes are presented by design or procedures or are specifically analysed.

Finally, if unexplained flux tilts do occur, the operating technical specifications [38] will provide appropriate corrective actions to provide continued safe operation of the reactor.

4.4.2.7 Fuel and Cladding Temperatures

Consistent with the thermal hydraulic design bases described in section 4.4.1, the following discussion pertains mainly to fuel pellet temperature evaluation.

The thermal hydraulic design provides that the maximum fuel temperature is below the melting point of uranium dioxide. To preclude centre melting and to serve as a basis for overpower protection system setpoints, a calculated centreline fuel temperature of {REDACTED} °C is selected as the overpower limit. This provides sufficient margin for uncertainties in the thermal evaluations.

The temperature distribution within the fuel pellet is predominantly a function of the local power density and the uranium dioxide thermal conductivity, but the computation of radial fuel temperature distributions combines CRUD, oxide, clad, gap, and pellet conductivity.

Fuel rod thermal evaluations (fuel centreline and average and surface temperatures) are performed at several times in the fuel rod lifetime (with consideration of time-dependent densification and thermal conductivity degradation) to determine the maximum fuel temperatures.

4.4.3 Description of the Thermal and Hydraulic Design of the Reactor Cooling System

Details of the reactor coolant system are provided in E3S Case Version 2, Tier 1, Chapter 5: Reactor Coolant System and Associated Systems [2].

4.4.4 Evaluation of the Validity of Thermal and Hydraulic Design Techniques

4.4.4.1 Critical Heat Flux

The CHF correlations planned to be used in the core thermal analysis are explained in Section 4.4.2.

4.4.4.2 Core Hydraulics

The following flow paths for core bypass are considered:

1. Flow through the spray nozzles into the upper head for head cooling purposes
2. Flow entering the rod cluster control and grey rod cluster guide thimbles
3. Leakage flow from the vessel inlet nozzle directly to the vessel outlet nozzle through the gap between the vessel and the barrel
4. Flow between the core barrel and the core shroud for the purpose of cooling and not considered available for core cooling

The above contributions are evaluated to confirm that the design value of the core bypass flow is met. Of the total allowance, one part is associated with the core (item 2 above) and the rest is associated with the internals (items 1, 3 and 4 above).

The friction factor for VIPRE-01 in the axial direction, parallel to the fuel rod axis, is evaluated using a correlation for a smooth tube. The effect of two-phase flow on the friction loss is expressed in terms of the single-phase friction pressure drop and a two-phase friction multiplier. The multiplier is calculated using the homogenous equilibrium flow model.

The flow in the lateral directions, normal to the fuel rod axis, views the Reactor Core as a large tube bank. This correlation is of the form:

$$FL = A\,Re_L^{-0.2}$$

where,

A	=	Function of the rod pitch and diameter a
Re_L	=	Lateral Reynolds number based on the rod diameter

4.4.4.3 Influence of Power Distribution

Influence on power distribution is covered in 4.3.3.3.

4.4.4.4 Core Thermal Response

The design bases of the application are to prevent DNB and to prevent fuel melting for frequent faults.

4.4.4.5 Fuel/Clad Temperatures

Fuel and clad temperatures are extracted from VIPRE-01 for comparison against key safety limits, noting this section of the report will be updated following confirmatory analysis by the fuel vendor and the calibration of the VIPRE-01 fuel model.

4.4.4.6 Thermal Margins

At the time of writing, margins to fuel limits are not being considered but are instead monitored to look for grossly high values.

A normal operations DNBR target has not been set at the time of writing; however, this is expected to be confirmed by the next issue of this chapter.

A DNB Analysis Limit of {REDACTED} has been selected based on the {REDACTED} correlation limits respectively, which are conservative in relation to fuel vendor correlations. The additional {REDACTED} additive factor applied to the correlation limits is discussed in section 3.15 of [39]. This additive factor is subject to change based on further developments with the fuel vendor. Margins are therefore quoted in relation to the DNB Analysis Limit, rather than to an operational limit.

4.4.4.7 Fuelled Region Design Inputs

The geometry for the radial and axial dimensions must be defined to develop the inputs for the single assembly model. Geometry may be input for ‘cold’ conditions, meaning the dimensions are the measured values at room temperature, or for ‘hot’ conditions, which are traditionally the dimensions that are thermally expanded using a material-specific equation evaluated at the core average temperature.

The RR SMR sub-channel analyses uses ‘cold’ geometry conditions. This assumption allows the use of dimensions directly from the reference fuel design document and maintains consistency with the pressure drop information. The grid spacer form loss and bare rod friction losses are evaluated with flow areas consistent with 'cold' conditions. To remove any conversions required for the inputs based on the bare rod flow areas, the inputs are simplified to assume 'cold' conditions. The thermal expansion for fuel and spacer materials are nearly identical; thus the change in flow area, and wetted and heated perimeters for the 'hot' conditions, is negligible. Additionally, axial geometry changes in the active fuel occur with exposure; however, hot channel factors discussed later in this chapter account for pellet densification among other variations.

Currently, the fuelled region inputs are based on the proprietary version of [40]. This chapter does not provide detailed descriptions here to avoid US Export Control restrictions.

4.4.4.8 17x17 Assembly Model

The 17x17 Assembly Model has been developed in conjunction with Constellation Energy Group. {REDACTED}.

1/8th Core Model

The 1/8th core model has been developed in conjunction with Constellation Energy Group [42]. Note that this model has not yet been used in RR SMR safety analysis for E3S Case Tier 1 Chapters.

4.4.4.9 Quarter Core Model

A quarter core model has been developed to support chemistry analysis. This will be discussed in more detail in a future revision of this chapter.

4.4.4.10 Turbulent Mixing

The turbulent mixing model within VIPRE-01 accounts for the exchange of enthalpy and momentum between adjacent sub-channels due to turbulent flow. The coefficient for turbulent mixing (ABETA) and the turbulent momentum factor are the two inputs needed for this model. This mixing model is incorporated into the energy and momentum equation, which is dependent on the amount of turbulent crossflow per unit length.

During sensitivity testing (Reference [43]), ABETA was found to have a reasonable impact on DNBR calculations for transient cases, with a recommended value of {REDACTED} performing better than the minimum (most conservative) value that provided reasonable convergence of {REDACTED}. The performance difference at steady-state, nominal conditions were negligible, therefore the recommended value of {REDACTED} is used in all steady-state assessments at nominal conditions.

Based on the sensitivity testing, this parameter is varied for fault studies analysis and the lowest value of ABETA that did not result in any stability issues was selected for forward analysis in each category.

The turbulent momentum factor was found to have a much smaller impact in transient and steady-state sensitivities than ABETA, thus the factor was set to its default value of {REDACTED}.

4.4.4.11 Flow Instability

Hydrodynamic stability (both static and dynamic) is not expected to be an issue in the RR SMR design due to its basis as a dispersed PWR. Future revisions of this chapter will show the results of the standard stability assessments performed to confirm this assumption.

4.4.4.12 Fuel Rod Behaviour Effects from Coolant Flow Blockage

Fuel rod behaviour effects from coolant flow blockage will be covered in Version 3 of the generic E3S Case.

4.4.5 Testing and Verification

4.4.5.1 Tests prior to Initial Criticality

A reactor coolant flow test will be performed following fuel loading but prior to initial criticality. Coolant loop pressure data is obtained in this test. This data allows determination of the coolant flow rates at reactor operating conditions. This test verifies that proper coolant flow rates have been used in the core thermal and hydraulic analysis.

4.4.5.2 Initial Power and Plant Operation

Core power distribution measurements are made at several core power levels. These tests are used to confirm that conservative peaking factors are used in the core thermal and hydraulic analysis.

4.4.5.3 Components and Fuel Inspections

Inspections performed on the manufactured fuel are described in Version 3 of the generic E3S Case. Fabrication measurements critical to thermal and hydraulic analysis are obtained to verify that the engineering hot channel factors in the design analyses are met.

4.4.6 Instrumentation Requirements

Instrumentation requirements will be covered in Version 3 of the generic E3S Case.

4.5 Core Components

4.5.1 System and Equipment Functions

The primary purpose of the Reactor System [JA] is to generate nuclear heat and transfer it to pressurised water flowing through the core, for onward heat transfer to the secondary systems so that electrical power can be generated. The core must both generate and transfer the heat in a safe, controlled manner throughout the plant lifetime.

The Reactor System comprises:

- RPV [JAA]
- IHP
- Control rod drive mechanisms (CRDMs)
- Reactor Core, comprising:
    - Reactor vessel internals (including in-core instrumentation)
    - Fuel assemblies
    - Neutron sources.

4.5.2 Design Bases

4.5.2.1 Functional Requirements

Safety categorised functional requirements are specified for the Reactor Core [JAC] based on the high-level safety functions (HLSFs) they deliver, including the applicable plant states and operating modes. These are presented in Table 4.5-1.

Table 4.5-1: Reactor Core [JAC] Safety Categorised Functional Requirements

Requirement ID	Requirement	Safety Category	Discussion
JA-R-1252	While in Modes 1, 2, 3, 4a, 4b, 5a or 5b, the Reactor System [JA] shall contain and confine coolant.	A	This requirement covers the need to maintain a pressurised coolant inventory in the core. Failure to meet this safety function could lead to fuel damage, the loss of the primary circuit boundary and a release of radioactive material.
JA-R-1260	While in all normal modes of operation, the reactor system [JA] shall transfer heat to reactor coolant.	C	This requirement covers the need to maintain a coolant inventory, in contact with the fuel assemblies in a state conductive to heat transfer.

Requirement ID	Requirement	Safety Category	Discussion
JA-R-1265	While in All modes of operation, the reactor system [JA] shall control the composition and configuration of fixed reactivity contributions.	A	This requirement covers the need to maintain a fixed coolable geometry in the Reactor Core. Any component which can affect reactivity should be controlled and restrained. Failure to meet this safety function could lead to a loss of reactivity control and result in fuel damage.
JA-R-1291	Whilst the emergency core cooling [JN01] function is in operation, the reactor system [JA] shall transfer residual heat from fuel to reactor coolant.	A	This requirement covers the need to ensure a continued flow of coolant is available to cool the core whilst in emergency core cooling (ECC) [JN01]. ECC is a safety category A function.
JA-R-1314	While in All modes of operation, the reactor system [JA] shall contain fuel.	A	This requirement covers the need to maintain fuel in a coolable geometry and ensure no migration of radioactive material from the fuel into the primary coolant.
JA-R-1366	While in Modes 1 and 2, the reactor system [JA] shall shutdown the reactor on demand.	C	This requirement covers the need be able to manually shutdown the reactor in all critical modes of operation during duty control. Shutdown via Scram is covered is a safety category A function.
JA-R-1367	While in Modes 3, 4a, 4b, 5a, 5b, 6a and 6b, the Reactor System [JA] shall hold down reactivity.	A	This requirement covers the need to maintain adequate SDM in all shutdown modes and under all possible conditions. This requirement supports the successful delivery of Scram which is a safety category A function.
JA-R-1368	While in all modes of operation, the reactor system [JA] shall prevent reactivity increase by inadvertent rod withdrawal.	A	This requirement covers the need to protect the fuel from inadvertent rod withdrawal (through rod withdrawal or ejection).
JA-R-1373	While in Modes 1 and 2, the reactor system [JA] shall control variable reactivity adjustments.	C	This requirement covers the need to control reactivity and thus fuel temperature in normal duty operations and thus has been categorised as safety category C.

Requirement ID	Requirement	Safety Category	Discussion
JA-R-1274	While in alternative shutdown function [JD02] is in operation, the Reactor System [JA] shall mix soluble boron with reactor coolant.	B	This requirement covers the need to supply boron to the fuelled region when the emergency boron injection function is initiated. Emergency boron injection is a safety category B function.
JA-R-1283	While the passive decay heat removal (PDHR) [JN02] function is in operation, the Reactor System [JA] shall transfer residual heat from fuel to reactor coolant.	B	This requirement covers the need to ensure a continued flow of coolant is available to cool the core whilst in PDHR [JN02]. PDHR [JD02] is a safety category B function.

Detailed non-functional performance assigned to the safety categorised functional requirements are in the requirements management database.

4.5.2.2 Non-Functional System Requirements

The E3S design principles, described in section 3.1.7 of E3S Case Version 2, Tier 1, Chapter 3: E3S Objectives and Design Rules for SSCs [44], are considered as part of the design of core components. Non-functional system requirements derived from these principles will be specified for the design in Version 3 of the generic E3S Case.

4.5.2.3 E3S Classification

The safety classification of the core components is provided in Table 4.5-2.

Table 4.5-2: Safety Classification of Key Components

Component	Safety Class	Discussion
Reactor pressure vessel (including the closure head)	1 (VHR)	The RPV provides the primary pressure boundary for the reactor system and support the safety category A 'contain coolant' safety function. Failure of the RPV cannot be mitigated through any other functions and thus is considered a VHR component.
Reactor vessel internals	1	The RPV internals fulfil several safety functional requirements including directing coolant from the RCS to the core. The RPV Internals play a principal role in supporting both Scram [JD01] and emergency core cooling [JN01] which are safety category A functions; thus, they have a safety class of 1.

Component	Safety Class	Discussion
Integrated head package (excluding the closure head and CRDMs)	1	A safety class of 1 has been assumed at RD7/DRP1.
Fuel assemblies	1	The fuel assemblies provide support to several safety functional requirements. The fuel assemblies play a principal role in supporting Scram [JD01] which is a safety category A function; thus, they have a safety class of 1.
Control rods	1	The Control rods provide play a principal role in supporting Scram which is a safety category A function.
Neutron sources	1	The Neutron sources provide a principal role in supporting Scram in low power conditions which is a safety category A function.

No environment, security or safeguards functions are defined at RD7/DRP1.

4.5.3 Description

Key performance and design parameters for the system are presented in Table 4.5-3.

Table 4.5-3: Key Performance and Design Parameters for the Reactor System [JA]

Parameter	Value	Units
Core Thermal Power	1358	MW
Number of Fuel Assemblies	121	n/a
Active Fuelled Length	2.8	m
Nominal Cycle Length (including refuel)	18	Months
Fuel Assemblies per Reload (equilibrium cycle)	{REDACTED}	n/a
RPV Outer Diameter (at core height)	{REDACTED}	mm
RPV Height (excluding Closure Head)	{REDACTED}	mm
Closure Head Height	{REDACTED}	mm
IHP Height	{REDACTED}	mm
Best Estimate Coolant Inlet Temperature (at Full Power)	{REDACTED}	°C
Best Estimate Coolant Outlet Temperature (at Full Power)	{REDACTED}	°C
Best Estimate System Coolant Mass Flow Rate	{REDACTED}	kg/s
Maximum Core Bypass Rate	{REDACTED}	%
System Coolant Pressure	{REDACTED}	MPa

The reactor system [JA] is a subsystem of the reactor plant [J].

The reactor system [JA] comprises the RPV [PT108], the closure head [PT158] and integrated head package [PT159], the RPV internals [PT110], the fuel assemblies [PT164], the neutron sources [PT165], the control rod assemblies [PT190] and the CRDMs [PT101].

The reactor system [JA] interfaces with the reactor coolant system [JE] via three primary circuit loops. These loops direct coolant to and from the RPV via three equally distributed (azimuthally) nozzles.

The RPV internals are suspended within the RPV from the upper flange of the core barrel, which seats on a support ledge in the RPV upper shell. A number of core support lugs on the RPV also provide lateral support to the lower core support plate of the RPV internals in the event that the upper flange fails. The RPV internals comprise of the upper internals which house the control rod housing columns and provide the upper core support plate, and the lower internals which include the core barrel, neutron reflector and the flow distribution device.

The RPV internals [PT110] also provide support for, and position, the fuel assemblies. There are 121 fuel assemblies within the core, with the primary function of generating heat through the fission process. Each fuel assembly contains UO_2 fuel pellets, enriched up to 4.95 % ^{235}U. As the reactor plant [J] operates without soluble boron, reactivity control is achieved through the addition of gadolinia neutron poisons within the fuel, and through the insertion and withdrawal of control rods.

The control rods [PT190] are the primary means of normal duty reactivity control. The control rods are a mixture of SINCAD, boron carbide and stainless steel. There are {REDACTED} control rod assemblies within the reactor system, which provide duty reactivity control, reactor shutdown and reactivity hold down for all operating conditions. The boron carbide control rods are generally fully withdrawn for high powered operations and provide the majority of the core shutdown and hold down reactivity. The SINCAD control rods are grouped into {REDACTED} banks, some of which will be partially inserted into the core at all times to provide reactivity control through cycle and through reactivity transients. The stainless steel control rods are present to provide powershape control and are specifically used during reactivity transients.

A simplified schematic of the reactor system is illustrated in Figure 4.5-1. It should be noted that Figure 4.5-1 does not show all components with the reactor system such as the fuel, control rods and neutron sources.

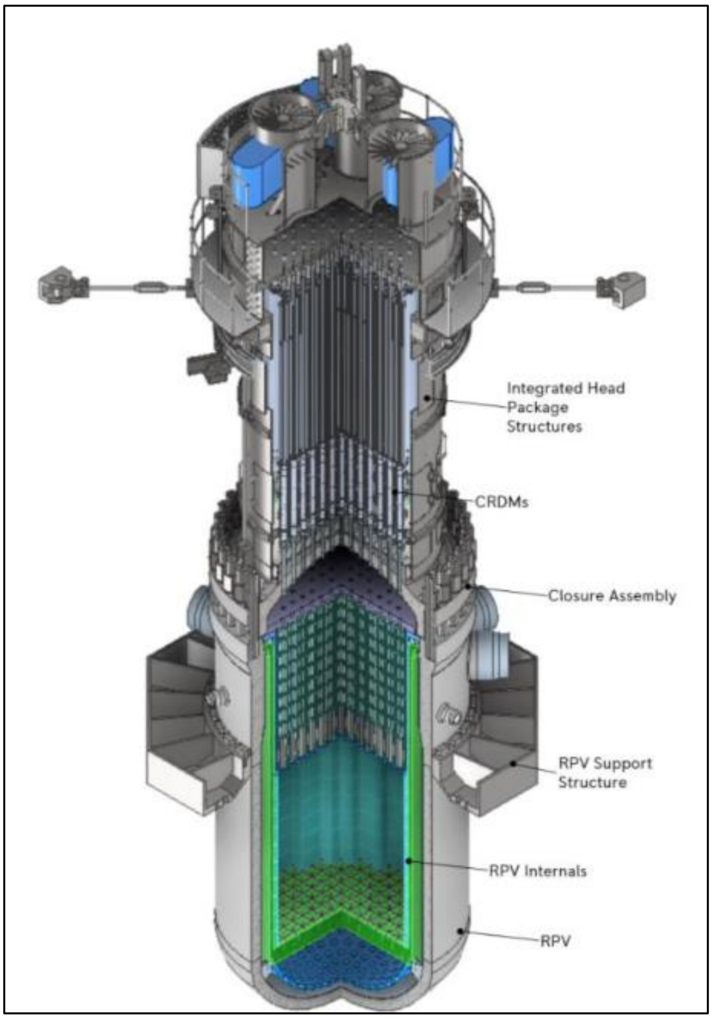

Figure 4.5-1: Simplified Schematic of the Reactor System [JA]

4.5.4 Materials

The major component forgings material for the reactor vessel internals [JAC10] (core barrel, neutron reflector and flow distribution device) is {REDACTED} stainless steel. The fuel pellets are made of UO_2 with a maximum enrichment of 4.95 %. Gadolinia poison is placed in a number of fuel pellets to control start of cycle reactivity. Fuel pellets have a maximum gadolinia loading of {REDACTED} %. The fuel clad is made of optimised ZIRLO™.

The description and justification of materials used for Class 1 SSCs are presented in E3S Case Version 2, Tier 1, Chapter 23: Structural Integrity [1].

4.5.5 Interfaces with other Equipment or Systems

Interfaces for the Reactor System [JA] are identified and managed within the RR SMR requirements management database.

4.5.6 System and Equipment Operation

None defined at RD7/DRP1.

4.5.7 Instrumentation and Control

4.5.7.1 Neutron Flux Monitoring

During powered operations the distribution of power within the core will be monitored using a series of self-powered neutron detectors (SPNDs). The SPNDs will be placed on a lance with {REDACTED} detectors in {REDACTED} axial locations and inserted into the instrumentation tube of all fuel assemblies which do not contain an RCCA or a primary neutron source.

4.5.7.2 Core Exit Thermocouples

Core exit thermocouples will be inserted into the same locations as the SPNDs and provide an indication of the primary coolant temperature distribution at the core exit. This provides an additional mechanism to understand the power distribution within the core as well as the ability to measure the margin to saturation on a finer spatial fidelity than with the hot loop thermocouples.

4.5.8 Monitoring, Inspection, Testing and Maintenance

At RD7/DRP1, the reactor system [JA] examination, maintenance, testing and inspection (EMIT) activities are to be defined, specific to the system environment and the operating context. The EMIT activities to be considered include:

- Safety derived tasks
- Design derived tasks (Supplier provided)
- Reliability derived tasks
- RGP/OPEX.

Specifically, for the Reactor Core [JAC], EMIT activities will include:

- Periodic physics testing
- Active core monitoring through the in-core monitoring system
- Post irradiation examination.

4.5.9 Radiological Aspects

Radiological aspects of the design of core components are considered low risk as standard materials and designs have been utilised. The removal of secondary neutron sources from the core design is considered a move to eliminate radiation hazards wherever possible.

4.5.10 Performance and Safety Evaluation

4.5.10.1 RPV Thermal Hydraulics

CFD analysis of the RPV thermal hydraulics has been conducted to gain confidence in the flow behaviour in the core inlet and through the outlet plenum.

Analysis results show that for all fuel assemblies, the core inlet maldistribution is within {REDACTED} % of the average flow at the core inlet. It has also been demonstrated that no significant undesirable hydraulic phenomena are present within the lower plenum due to the latest FDD design, full details of the analysis performed on the FDD are contained within the FDD Decision Record [45].

Initial results also show the coolant flow is well distributed by the top of the fuelled region. Initial concerns were raised about the potential for cross flow in the fuelled region caused by crowding in the outlet plenum (due to the large number of control rods); {REDACTED}.

4.5.10.2 Reflector Heating

Initial calculations for the heating of coolant in the metal radial reflector have been conducted to demonstrate adequate flow rates in this region. Flow rates were initially set to {REDACTED} % of the total RPV inlet mass flow rate based on the requirement for reflector cooling flow ({REDACTED} of RPV inlet flow allocated to the cooling channels and {REDACTED} of RPV Inlet flow allocated to the annulus to the Core Barrel). CFD analysis has been conducted on a single channel with a predefined mass flow rate; this was then used to feed into an array design tool which utilises the CFD data with nuclear heating input data to provide an initial guide to generate the hole pattern for the cooling channels.

Whilst the design of the reflector is not yet fixed, this initial analysis has shown that reflector metal temperatures can be kept below the maximum design temperature requirement with a bypass rate that is slightly above the assumed percentage of {REDACTED}. However, this presents a low risk as there is future optimisation of the Neutron Reflector cooling channel pattern will look to reduce the required bypass flow rate whilst maintaining the required cooling rate.

4.5.10.3 RPV Head Cooling

The flow path for the RPV head cooling bypass flow was previously undefined; however, design development has been conducted in order to determine the geometry required to achieve the defined requirement for {REDACTED} bypass flow. The flow path has been designed such that the pressure drop matches the pressure drop across the main flow path. This utilises various holes in the Core Barrel upper flange including those allocated to surveillance capsule removal as well as {REDACTED} holes in the upper support barrel flange. Further analysis of this region will finalise the bypass flow rate requirement and demonstrate the required temperature has been achieved.

4.5.10.4 Clad Corrosion

Corrosion of the fuel clad is being investigated to demonstrate that the corrosion rates, as well as other related phenomena (CRUD deposition, CILC, Hydrogen Pickup etc.), are minimised as far as possible. Whilst standard materials are being used within the core (including the use of Optimized ZIRLO™ as a clad material), the absence of boron in the primary coolant and the use of potassium as a pH raiser represents a novelty. Initial reviews of corrosion data taken from autoclave testing shows that corrosion rates are improved with the use of potassium in comparison to lithium which is industry standard for western PWRs.

Thermal analysis has demonstrated that the RR SMR is a low boiling duty core which suggests that CRUD deposition rates will be reduced in comparison to other modern nuclear plant designs. As a result of this, it is believed that related phenomena such as CILC will also be minimised.

4.5.10.5 Mechanical Load

Analysis of the mechanical performance of reactor components including RCCAs during high load faults (LOCA, seismic etc) is to be conducted during the detailed design phase.

4.6 Conclusions

4.6.1 ALARP, BAT, Secure by Design, Safeguards by Design

The design of the reactor system [JA] components is developed in accordance with the systems engineering design process. This includes alignment to RGP and OPEX, design to codes and standards according to the safety classification, and a systematic optioneering process with down-selection of design options based on assessment against relevant safety criteria that ensure risks are reduced to ALARP, apply BAT, and are secure by design and safeguards by design, as described in E3S Case Version 2, Tier 1, Chapter 3: E3S Objectives and Design Rules [44]).

The suite of analysis presented in this chapter provide confidence at RD7/DRP1 that all acceptance criteria in the design basis can be met by the fuel and core design. This provides confidence that claims can be met when the full suite of arguments and evidence is developed.

The overall demonstration of ALARP, BAT, secure by design and safeguards by design at RD7/DRP1 is presented in E3S Case Version 2, Tier 1, Chapters 24, 27, 32 and 33 respectively.

4.6.2 Assumptions and Commitments on Future Dutyholder / Licensee / Permit Holder

Assumptions and commitments raised on the future Dutyholder/Licensee/Permit Holder are summarised in Table 4.6-1.

Table 4.6-1: Assumptions and Commitments on Future Dutyholder/Licensee/ Permit Holder

Assumption/ Commitment	ID	Description
Commitment	C4.1	The future Duty Holder/Licensee shall conduct suitable inspection of new fuel.
Commitment	C4.2	The future Duty Holder/Licensee shall conduct suitable checks to ensure fuel and control rods are loaded into the reactor assembly as intended.
Commitment	C4.3	The future Duty Holder/Licensee shall monitor the reactivity of the reactor during core load.
Commitment	C4.4	The future Duty Holder/Licensee shall conduct physics testing on every new fuel arrangement before a return to power operations.
Commitment	C4.5	The future Duty Holder/Licensee shall conduct suitable Post Irradiation Examination on burned fuel to monitor fuel integrity.

4.6.3 Conclusions and Forward Look

The generic E3S Case objective at Version 2 is ‘to provide confidence that the RR SMR design will be capable of delivering the E3S fundamental objective as it developed from a concept design into a detailed design’ [5]. This confidence is built through development and underpinning of top-level claims across each chapter of the E3S Case, through supporting arguments and evidence. The top-level claim for chapter 4 is ‘the reactor (fuel and core) is conservatively designed and verified to deliver E3S functions through-life, in accordance with the E3S design principles, to reduce risks to ALARP with application of BAT, secure by design and safeguards by design’.

The arguments and evidence presented to meet the generic E3S Case objective at Version 2 include an overview of the fuel and core design at ‘Iteration 7’, representing a mature design where the total core layout and design limits are defined. A suite of nuclear, fuel, and thermohydraulic analyses is undertaken to demonstrate that acceptance criteria and design limits can be achieved at this stage.

Further arguments and evidence to underpin the claim will be developed in line with the E3S Case Route Map [6] and reported in future revisions of the generic E3S Case, which will further build confidence that the RR SMR can deliver its fundamental E3S objective. This broadly includes development of the design basis and optimisation of the core design particularly surrounding the operational philosophy during load follow and lower power states, further iterations of the suite of analyses for the fuel and core for all modes of operation, development of a complete set of non-functional system requirements for the core components from the E3S design principles, and verification and validation of all E3S requirements.

4.7 References

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4.8 Appendix A: Claims, Arguments, Evidence

Table 4.8-1 provides a mapping of the claims to the corresponding sections of the chapter that summarise the arguments and/or evidence. The full decomposition of claims and link to underpinning Tier 2 and Tier 3 information containing the detailed arguments and evidence is presented in the E3S Case Route Map [6]. The route map includes the trajectory of Tier 2 and Tier 3 information as the generic E3S Case develops, which will be incorporated into Tier 1 chapters as it becomes available and in line with generic E3S Case issues described in [5].

Table 4.8-1: Mapping of Claims to Chapter Sections

Claim	Section of Chapter 4 containing Arguments / Evidence summary
Reactor Core [JAC] non-functional system requirements are complete	4.5.2.2
Reactor Core [JAC] non-functional system requirements are correctly assigned	4.5.2.2
Reactor Core [JAC] codes and standards are correctly assigned	4.0.4
Safety Requirements for the Reactor Core [JAC] are complete	4.5.2.1
Environmental functional requirements for the Reactor Core [JAC] are complete	None at this revision
Security functional requirements for the Reactor Core [JAC] are complete	None at this revision
Safeguards functional requirements for the Reactor Core [JAC] are complete	None at this revision
The Reactor Core [JAC] is classified correctly	4.5.2.3
The Reactor Core [JAC] design achieves its E3S functional requirements	4.5.3 – 4.5.10
The Reactor Core [JAC] design achieves its E3S non-functional system requirements	4.5.3 – 4.5.10
The Nuclear Design Basis (DBC 1, 2i, 2ii, 3i and 3ii) is clearly defined and justified	4.3.1
The Thermal Hydraulics Design Basis (DBC 1, 2i, 2ii, 3i and 3ii) is clearly defined and justified	4.4.1
The Fuel Performance Design Basis (DBC 1, 2i, 2ii, 3i and 3ii) is clearly defined and justified	4.2.2
The Criticality Design Basis (DBC 1, 2i, 2ii, 3i and 3ii) is clearly defined and justified	4.3.1
Analysis of the Nuclear Design verifies that all acceptance criteria defined in the design basis have been met	4.3.3

Claim	**Section of Chapter 4 containing Arguments / Evidence summary**
Analysis of the Thermal Hydraulics verifies that all acceptance criteria defined in the design basis have been met	4.4.2 – 4.4.6
Analysis of the Fuel Performance verifies that all acceptance criteria defined in the design basis have been met	4.2.3
Analysis of the Neutron Sources verifies that all acceptance criteria defined in the design basis have been met	Not covered in this revision
Criticality analysis verifies that all acceptance criteria defined in the design basis have been met	4.3.3
The Reactor Core [JAC] analysis methods are validated	4.3.4
E3S requirements for the Reactor Core [JAC] are verified and validated through manufacture, installation, and assembly	4.5.3 – 4.5.10
E3S requirements for the Reactor Core [JAC] are verified and validated through commissioning	Not covered in this revision
E3S Requirements for the Reactor Core [JAC] are verified and validated through its operational life	Not covered in this revision

4.9 Abbreviations

ALARP	As Low As Reasonably Practicable
AO	Axial Offset
ASF	Alternative Shutdown Function
ASME	American Society of Mechanical Engineers
BA	Burnable Asborber
BAT	Best Available Techniques
BOC	Beginning of Cycle
BOL	Beginning of Life
BS	British Standard
BWR	Boiling Water Reactor
CAE	Claims, Arguments, Evidence
CFD	Computational Fluid Dynamics
CHF	Critical Heat Flux
CILC	CRUD Induced Localised Corrosion
CMS	Core Management System
CRD	Control Rod Drive
CRDM	Control Rod Drive Mechanism
DB	Design Basis
DBC	Design Basis Condition
DFBN	Debris Filter Bottom Nozzle
DNB	Departure from Nucleate Boiling
DNBR	Departure from Nucleate Boiling Ratio
DRP	Design Reference Point
E3S	Environment, Safety, Security and Safeguards
ECC	Emergency Core Cooling
EFPM	Effective Full Power Month
EMIT	Examination, Maintenance, Testing and Inspection
EOC	End of Cycle
EUR	European Utility Requirements

FDD	Flow Distribution Device
FGR	Fission Gas Release
FQ	Heat Flux Hot Channel Factor
GDA	Generic Design Assessment
GRCA	Gray Rod Cluster Assembly
HFP	Hot Fuel Power
HLSF	High-Level Safety Function
HMI	Human-Machine Interface
HWR	High Rod Worth
I/L2	Fuel Rod Clad Moment of Inertia/Grid Span
IAEA	International Atomic Energy Agency
IFM	Intermediate Flow Mixer
IHP	Integrated Head Package
LHGR	Linear Heat Generation Rate
LOCA	Loss of Coolant Accident
LWR	Light Water Reactor
MCNP	Monte-Carlo N-Particle
MCR	Main Control Room
MES	Manufacturing Execution System
MOC	Middle of Cycle
MTC	Moderator Temperature Coefficient
NDE	Non-Destructive Evaluation
OECD/NEA	Organisation for Economic Co-operation and Development Nuclear Energy Agency
OPEX	Operating Experience
PCI	Pellet-Clad Interaction
PCMI	Pellet-Cladding Mechanical Interaction

PDHR	Passive Decay Heat Removal
PIE	Post-Irradiation Examination
PRXA	Partially Recrystallized Annealed
PWR	Pressurised Water Reactor
QMS	Quality Management System
RAPFE	Radially-Averaged Peak Fuel Enthalpy
RCCA	Rod Control Cluster Assembly
RCS	Reactor Coolant System
RD	Reference Design
RFA	Robust Fuel Assembly
RGP	Relevant Good Practice
RPV	Reactor Pressure Vessel
RR SMR	Rolls-Royce SMR
RTN	Removal Top Nozzle
RTP	Rated Thermal Power
SCC	Stress Corrosion Cracking
SDM	Shutdown Margin
SINCAD	Silver-Indium-Cadmium Alloy
SPND	Self-Powered Neutron Detectors
SSG	Safety Specific Guide
SSR	Safety Specific Requirements
TDC	Thermal Diffusion Coefficient
UO_2	Uranium Dioxide
WIN	Westinghouse Integral Nozzle
WSR	Worst Stuck Rod