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| ONR Assessment Report  Generic Design Assessment of the Rolls Royce SMR – Step 2 assessment of Fuel and Core |



ONR Assessment Report

**Project Name**: Generic Design Assessment of the Rolls-Royce SMR

**Report Title**: Step 2 assessment of Fuel and Core

**Authored by**: [Redacted]

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# Executive Summary

This report presents the outcomes of my fuel and core assessment of the Rolls-Royce Small Modular Reactor (SMR) as part of Step 2 of the Office for Nuclear Regulation (ONR) Generic Design Assessment (GDA). This assessment is based upon the information presented in version 2 of Rolls-Royce SMR Limited’s Environmental, Safety, Security and Safeguards (E3S) case chapters and supporting documentation.

ONR’s GDA process calls for a step-wise assessment, which increase in detail as the project progresses. The focus of my assessment in this step was towards the fundamental adequacy of the Rolls-Royce SMR design and safety case, and the suitability of the methodologies, approaches, codes, standards and philosophies which form the building blocks for the design and generic safety and security cases.

I targeted my assessment, in accordance with my assessment plan, at the content of most relevance to fuel and core against the expectations of ONR’s Safety Assessment Principles (SAPs), Technical Assessment Guides (TAGs) and other guidance which ONR regards as relevant good practice.

I targeted the following aspects in my assessment of the Rolls-Royce SMR E3S case:

* The nuclear design of the reactor core;
* The sub-channel thermal hydraulic design of the reactor core; and
* The thermo-mechanical design of the nuclear fuel system.

I specifically targeted those areas of the fuel and core design with novelty associated with them, when compared to a gigawatt scale reactor of a similar type. Through sampling against those areas I considered as having the highest developmental risk or with potentially the lowest safety margins, I was able to form a judgment on the fundamentals of the RP’s design and safety case; in line with the objectives of the step.

Based upon my assessment, I have concluded the following:

* The identification, location and sizing of key systems, structures and components in the fuel and core design have been set and in my judgement are suitable to underpin the reactor core’s defined Rated Thermal Power of 1,358 MWth.
* The RP's analysis has provided me with adequate confidence that the design will be capable of achieving compliance with its design criteria. Where non-compliances have been identified, options exist to return these to compliance.
* Appropriate optioneering has been performed by the RP and Relevant Good Practice including appropriate design codes, standards and operational experience have been identified; their relevance are understood, and arrangements are in place to address any shortfalls.
* There are areas of the design which were inside the scope of Step 2 which lacked maturity when compared to the rest of the safety case. They included:
  + Load following: Whilst the requirements for load following remain largely unset, there are no fundamental reasons why load following should be precluded by the RP’s design decisions made to date.
  + Power peaking: Power peaking performance is yet to be demonstrated for all times in life. However, the use of control rod sequencing was shown as being capable of controlling reactivity through cycle and reducing power peaking when sequencing patterns are optimised.
  + Fuel assembly crud and corrosion: A proof-of-concept is provided for crud and corrosion performance in potassium based chemistry. The justification needs to be strengthened including the identification of defence-in-depth measures to prevent, protect and mitigate against any deleterious effects and demonstrably reduce risks As Low As Reasonably Practicable.
  + Core monitoring during core load: The ability to monitor core load was demonstrated by making assumptions about ex-core neutronic detector performance. In-core detectors were not assumed. The RP will consider temporary in-core monitoring for inclusion in later design development.
* I did not identify any potential gaps, omissions or limitations which may result in a shortfall of sufficient significance that safety case claims may not be supported by an adequate future substantiation.

Overall, based on my assessment to date, and subject to the provision and assessment of suitable and sufficient supporting evidence, I have not identified any fundamental safety shortfalls that could prevent ONR permissioning the construction of a power station based on the generic Rolls-Royce SMR design.

# List of Abbreviations

AIC Silver Indium Cadmium

ALARP As Low As Reasonably Practicable

ASF Alternative Shutdown Function

BDF Boron Dilution Fault

BU Burn Up

BWR Boiling Water Reactor

CAE Claim, Argument and Evidence

CH Closure Head

CILC Crud Induced Localised Corrosion

CoFT Control of Fuel Temperature

CoR Control of Reactivity

CoRM Control of Radioactive Material

CR Control Rod

CRE Control Rod Ejection

Crud Chalk River Unidentified Deposits

CRWA Control Rod Withdrawal Accident

CZP Cold Zero Power

DAC Design Acceptance Confirmation

DBC Design Basis Condition

DiD Defence-in-Depth

DNB Departure from Nucleate Boiling

DNBR Departure from Nucleate Boiling Ratio

DPC Doppler Power Coefficient

ECC Emergency Core Cooling

EIMT Examination, Inspection, Maintenance and Testing

EPRI Electric Power Research Institute

EUR European Utility Requirements

E3S Environment, Safety, Security and Safeguards

FA Fuel Assembly

FQ Heat flux hot channel factor

FV Fuel Vendor

F&C Fuel and Core

FΔH Nuclear enthalpy rise hot channel factor

GDA Generic Design Assessment

HPUF Hydrogen Pick Up Factor

HTS Heat Transfer Surface

IAEA International Atomic Energy Agency

It.# Core design iteration number #

Keff Effective (neutron) multiplication factor

Kinf Infinite (neutron) multiplication factor

ONR Office for Nuclear Regulation

OpEx Operational Experience

PCT Peak Clad Temperature

LOCA Loss of Cooling Accident

MTC Moderator Temperature Coefficient

MWe Mega Watts electrical

MWth Mega Watts thermal

NEA Nuclear Energy Agency

NRW Natural Resources Wales

NS Neutron Source

OECD Organisation for Economic Co-operation and Development

PDHR Passive Decay Heat Removal

PIE Post Irradiation Examination

PWR Pressurised Water Reactor

RCCA Rod Cluster Control Assembly

RDS-PP Reference Designation System for Power Plants

RGP Relevant Good Practice

RP Requesting Party

RPV Reactor Pressure Vessel

RQ Regulatory Query

SAP Safety Assessment Principle(s)

SDD System Design Description

SDM Shut Down Margin

SFAIRP So Far As Is Reasonably Practicable

SFP Spent Fuel Pool

SMDD Safety Measure Design Description document

SMR Small Modular Reactor

SSC Structure, System and Component

TAG Technical Assessment Guide(s) (ONR)

TSC Technical Support Contractor

UFC Ultrasonic Fuel Cleaning

V&V Verification and Validation

WENRA Western European Nuclear Regulators’ Association

WSR Worst Stuck Rod

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# Introduction

1. This report presents the outcomes of my Fuel and Core (F&C) assessment of the Rolls-Royce Small Modular Reactor (SMR) as part of Step 2 of the Office for Nuclear Regulation (ONR) Generic Design Assessment (GDA). This assessment is based upon the information presented in Version 2 of Rolls-Royce SMR Limited’s Environmental, Safety, Security and Safeguards (E3S) case chapters (refs [1], [2], [3], [4], [5], [6], [7], [8], [9], [10], [11], [12], [13], [14], [15], [16], [17] and supporting documentation).
2. Assessment was undertaken in accordance with the requirements of the ONR Management System and follows ONR’s Guidance on the Mechanics of Assessment, NS-TAST-GD-096 (ref. [18]). The ONR Safety Assessment Principles (SAPs) (ref. [19]) together with supporting Technical Assessment Guides (TAGs) (ref. [20]) have been used as the basis for this assessment.
3. This is a Major report (refer to NS-TAST-GD-108 (ref. [21])).

## Background

1. The ONR’s GDA process (ref. [22]) calls for a step-wise assessment of the Requesting Party's (RP) submissions with the assessments increasing in detail as the project progresses. Rolls-Royce SMR Limited is the RP for the GDA of the Rolls-Royce SMR design.
2. In April 2022 ONR, together with the Environment Agency and Natural Resources Wales (NRW), began Step 1 of the GDA for the generic Rolls-Royce SMR design. Step 1, which is the preparatory part of the design assessment process and mainly associated with initiation of the project and preparation for technical assessment in later steps, was successfully completed in 12 months.
3. Step 2 commenced in April 2023. This is the first substantive technical assessment step. The focus of ONR’s assessments in this step is towards the fundamental adequacy of the design and safety and security cases, and the suitability of the methodologies, approaches, codes, standards and philosophies which form the building blocks for the design and generic safety and security cases. The objective is to undertake an assessment of the design against regulatory expectations to identify any fundamental safety or security shortfalls that could prevent ONR permissioning the construction of a power station based on the design.
4. Prior to the start of Step 2 I prepared a detailed Assessment Plan for F&C (ref. [23]). This has formed the basis of this assessment and was also shared with the RP to maximise openness and transparency.
5. This report is one of a series of Assessments which support ONR’s overall judgements at the end of Step 2 which are recorded in the Step 2 Summary Report (ref. [24]).

## Scope

1. The assessment documented in this report is based upon the E3S case for the Rolls-Royce SMR as summarised in the E3S case chapters and supporting documentation.
2. The overall scope of the Rolls-Royce SMR GDA is described in (ref. [25]). Rolls-Royce SMR Limited has indicated that it intends to complete a three step GDA, with the objective of receiving a Design Acceptance Confirmation (DAC) from ONR and have aligned their GDA scope with this objective. The GDA scope defines the generic plant and layout and includes all systems, structures and components (SSC) that are identified as being important to safety, security and safeguards, all modes of operation, and all stages of the plant lifecycle.
3. However, given the step-wise assessment during GDA, information has not been submitted for all aspects within the GDA Scope during Step 2. The following aspects of the E3S case are therefore out of scope of this assessment.

* Uncertainty quantification for manufacturing tolerances and parameter distributions.
* Procedures and operating rules, including for degraded modes.

1. The absence of data related to these areas is not significant for GDA Step 2 but will need to be addressed for Step 3, as is intended by the RP.
2. My assessment has considered the following aspects:

* The nuclear design of the reactor core, the sub-channel thermal hydraulic design of the reactor core, and the thermo-mechanical design of the nuclear fuel system.
* Those SSCs identified within Ch.4 (ref. [3]) and in addition, the Control Rod Assemblies whose design and functional requirements are inherently linked to those of the Reactor System. In summary:
  + Core Assembly (covered by Reference Designation System for Power Plants (RDS-PP) identifier [JAC]), consisting:
    - Reactor Vessel Internals [PT110];
    - Fuel Assemblies (FA) [PT164]; and
    - Neutron Sources (NS) [PT165].
  + Control Rod Assemblies (CR) [PT190].
* And, the justification of those components for:
  + All cores – Initial, transitional and equilibrium cores (based on a single design concept of an 18 month, three batch cycle).
  + All modes of operation – from Mode 1 (power operation, including load following) through to Mode 6b (refuelling).
  + All lifecycle phases – Design (Ch.3 (ref. [2])), Plant Construction and Commission (Ch.14 (ref. [10])), Conduct of Operations (Ch.13 (ref. [9])), Operational Limits and Conditions (Ch.16 (ref. [12])) to Decommissioning and End of Life Aspects (Ch.21 (ref. [15])).

# Assessment standards and interfaces

1. For ONR, the primary goal of the GDA Step 2 assessment is to reach an independent and informed judgment on the adequacy of a safety, security and safeguards case for the reactor technology being assessed.
2. ONR has a range of internal guidance to enable Inspectors to undertake a proportionate and consistent assessment of such cases. This section identifies the standards which have been considered in this assessment.
3. This section also identifies the key interfaces with other technical topic areas.

## Standards

1. The ONR SAPs (ref. [19]) constitute the regulatory principles against which the RP’s case is judged. Consequently, the SAPs are the basis for ONR’s assessment and have therefore been used for the Step 2 assessment of the Rolls-Royce SMR.
2. The International Atomic Energy Agency (IAEA) safety standards (ref. [26]) and nuclear security series (ref. [27]) are a cornerstone of the global nuclear safety and security regime. They provide a framework of fundamental principles, requirements and guidance. They are applicable, as relevant, throughout the entire lifetime of facilities and activities.
3. Furthermore, ONR is a member of the Western European Nuclear Regulators Association (WENRA). WENRA has developed Reference Levels (ref. [28]), which represent good practices for existing nuclear power plants, and Safety Objectives for new reactors (ref. [29]).
4. The relevant SAPs, IAEA standards and WENRA reference levels are embodied and expanded on in the TAGs (ref. [20]). The TAGs provide the principal means for assessing the F&C aspects in practice.

### Safety Assessment Principles

1. The key SAPs applied within my assessment are EKP, ERC, EAD, FA and the AV series.
2. The EKP and ERC SAPs set engineering principles that are fundamental to the design of the F&C SSCs.
3. The EAD SAPs are important because of the significant way in which F&C performance evolves during operation as fuel is depleted, as well as the degradation mechanisms that affect FAs through irradiation.
4. The FA SAPs are important because they set an expectation that fault consequences be identified and analysed such that risk can be demonstrated as low as reasonably practicable (ALARP), which has implications for all F&C data and methods used in fault analyses.
5. The AV SAPs are important because of the highly complex nature of neutronic, thermal hydraulic and thermo-mechanical phenomena occurring in the core mean that the design and safety case are reliant on outputs from analyses using computer codes. The AV SAPs also set expectations for assuring the validity of data and models used within, or outputted by, such computer codes.
6. A list of the SAPs used in this assessment is recorded in Appendix 1 – Relevant SAPs considered during the assessment.

### Technical Assessment Guides

1. The following TAGs have been used as part of this assessment:

* NS-TAST-GD-005 (TAG-005) – Regulating Duties to Reduce Risks to ALARP (ref. [30]).
* NS-TAST-GD-075 (TAG-075) – Safety of Nuclear Fuel in Power Reactors (ref. [31]). Specific advice is given to inspectors in TAG-075 on the interpretation of the EKP, ERC and EAD SAPs.
* NS-TAST-GD-096 (TAG-096) – Guidance on Mechanics of Assessment (ref. [18]).

### National and international standards and guidance

1. The following international standards and guidance have been used as part of this assessment:

* IAEA, Safety of Nuclear Power Plants: Design, SSR-2/1 (ref. [32]).
* IAEA, Deterministic Safety Analysis for Nuclear Power Plants, SSG-2 (ref. [33]).
* IAEA, Storage of Spent Nuclear Fuel, SSG-15 (ref. [34]).
* IAEA, Design of the Reactor Core for Nuclear Power Plants. SSG-52 (ref. [35]).
* IAEA, Format and Content of the Safety Analsyis Report for Nuclear Power Plants, SSG-61 (ref. [36]).
* IAEA, Core Management and Fuel Handling for Nuclear Power Plants, SSG-73 (ref. [37]).
* IAEA, Review of Fuel Failures in Water Cooled Reactors, NF-T-2.1 (ref. [38]).
* OECD NEA, Nuclear Fuel Safety Criteria Technical Review, Second Edition (ref. [39]).

1. These references have been used alongside the identified ONR SAPs and TAGs as a more detailed means of establishing whether relevant standards have been applied to the RP’s design definition.

## Integration with other assessment topics

1. I have worked closely with other topics as part of my F&C assessment. Similarly, other assessors sought input from my assessment. These interactions are key to the success of GDA to prevent or mitigate any gaps, duplications or inconsistencies in ONR’s assessment.
2. The key interactions with other topic areas were:

* Chemistry: To understand the implications of the primary circuit chemistry on the fuel pins’ heat transfer surfaces (ref. [40]).
* Fault Studies: To agree the suitablity of the F&C design criteria being used as part of fault analysis to demonstrate adequate fault tolerance (ref. [41]).
* Mechanical Engineering: To provide a view on the RP’s assumptions relating to the numbers and as-irradiated condition of FAs, to inform any future spent fuel pool (SFP) thermal safety assessments. This is expected during Step 3 (ref. [42]).
* Radiological Protection and Criticality: To provide a view on the RP’s assumptions relating to the numbers and as-irradiated condition of FAs, to inform spent fuel pool (SFP) criticality safety assessment (ref. [43]).
* Structural Integrity: To provided my view on whether the numbers of Reactor Pressure Vessel (RPV) Closure Head (CH) penetrations seemed justifiable; and in providing my view on assumptions originating from the RP’s F&C area with potential consequence to RPV irradiation embrittlement (ref. [44]).

## Use of technical support contractors

1. During Step 2 I engaged Technical Support Contractors (TSCs) to support the following specific aspects of my assessment of F&C for the Rolls-Royce SMR:

* An independent technical review of the RP’s approach to code validation, including the adequacy of the validation basis for the codes employed for GDA, and the overall strategy for ‘full’ validation of the codes in the future.

1. They provided me with technical advice and supported my assessment, working under my close direction and supervision. It should be noted that all regulatory judgements have been made exclusively by ONR.

# Requesting Party’s submission

1. Rolls-Royce SMR Limited submitted a series of E3S chapters, or summary reports, and other supporting references, which outline the E3S case for the generic Rolls-Royce SMR design. This section presents a summary of the RP’s safety case for F&C. It also identifies the documents submitted by the RP which have formed the basis of my F&C assessment of the Rolls-Royce SMR.

## Summary of the Rolls-Royce SMR design

1. The generic Rolls-Royce SMR design is a three loop Pressurised Water Reactor (PWR) with a target electrical power output of 470 MWe (from a thermal power of 1,358 MWth) and a design life of 60 years for non-replaceable components.
2. The Rolls-Royce SMR design has been developed by the RP based upon well-established PWR technology, in use all over the world. Innovation comes in the form of its modular approach to construction which would see the majority of the power station built in factory conditions and assembled on site.
3. The reactor itself is of a typical PWR design, including a steel RPV holding FAs, Steam Generators, Reactor Coolant Pumps and piping, all held within a steel containment vessel. The reactor is equipped with a number of supporting systems for normal operations and a range of safety measures are present in the design to provide cooling, control criticality and contain radioactivity under fault conditions. Passive safety features are preferred to active components, reflecting the RP’s design philosophy.
4. The reactor core consists of 121-off 2.8m 17x17 fuel pin FAs, to be burnt-up over an 18 month, three batch refuelling cycle; with 44 FAs replaced per cycle.
5. There are two FA types used, with the lower reactivity FAs generally loaded towards the centre of the core. Both FA types use UO2 fuel and are axially zoned. The equilibrium core’s FAs are each constructed from three unique fuel pin designs, six pin types in total. Whilst axial zoning is more commonly associated with Boiling Water Reactor (BWR) designs, the remainder of the FAs construction is of a ‘standard’ PWR type.
6. The reactor has been designed to operate without soluble boron; instead using its CRs alone for the purpose of providing reactivity control. There are 89 Rod Cluster Control Assemblies (RCCA) made up of boron carbide (B4C) CRs, silver-indium-cadmium (AIC) CRs, and stainless-steel ‘grey rods’. The RCCAs (latterly referred to as CRs) are operated in banks to manage through life reactivity control and core depletion.
7. Potassium hydroxide (KOH) is used as the alkalising agent for the primary circuit, instead of lithium hydroxide (LiOH); as in the absence of soluble boron the RP predicts better corrosion performance.

## E3S case approach and structure

1. Rolls-Royce SMR Limited has chosen to develop its cases in a holistic manner, as an E3S case. The overall objective for the E3S case is to demonstrate that the design will ‘protect people and the environment from harm’.
2. This means that, although the case made for each of the E3S purposes (i.e. environment, safety, security and safeguards) will inevitably be different at the top level, it will draw upon common evidence outputs (as well as other non-common outputs) to substantiate each of the purposes. This is claimed to offer benefits in terms of clarity, integration and understanding impacts from any changes to the case.
3. The E3S case is being developed using a three tier hierarchy and incorporating a Claim, Argument and Evidence (CAE) structure with the highest-level claims being derived from the RP’s own E3S principles. The highest level of the three tiers is the RP’s Tier 1 E3S chapters, with the lower tiers providing more detailed arguments and evidence. This is illustrated in Figure 1.

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**Figure 1: Claim, Argument and Evidence (CAE) structure within the E3S hierarchy (ref.** [1]**)**

1. The structure of the E3S case largely aligns with the IAEA guidance for safety cases, SSG-61 (ref. [36]), supplemented to include UK specific expectations and expanded to include the other E3S purposes.

## Summary of the Requesting Party’s E3S case for fuel and core

1. The aspects covered by the Rolls-Royce SMR safety case in the area of F&C can be broadly grouped under five headings, summarised as followed:

### Generate heat

1. The reactor core is required to generate 1,358 MWth for a cycle length of 18 months (ref. [3]).

### Control of Fuel Temperature (CoFT)

1. A safe margin to the Departure from Nucleate Boiling Ratio (DNBR) is to be achieved for all operational modes, including during transients and frequent faults (ref. [3]).

### Maintain negative fuel and coolant temperature coefficients of reactivity – Control of Reactivity (CoR)

1. 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 (ref. [3]).

### Control (the release) of Radioactive Material (CoRM)

1. The reactor core shall maintain barriers for limiting the release of radioactive material from the fuel pellets, in normal and faulted conditions (ref. [3]).

### ALARP

1. The design of the RR SMR will reduce safety risks to ALARP (ref. [17]).

## Basis of assessment: Requesting Party’s documentation

1. The principal documents that have formed the basis of my F&C assessment of the E3S case at Tier 2 are:

* SMR Reactor System Design Description (SDD) (ref. [45]) – details the description of the system design and its development from the requirements through to the definition and the supporting assessments to underpin the definition.
* Fuel and Core Design Basis Summary (ref. [46]) – covers the key safety criteria and their associated design bases for F&C. This report follows the general structure of the Justification of Design Limits document, which provides the key safety limits and their related criteria that the design basis is aiming to meet.
* Fuel and Core Performance Analysis Summary (ref. [47]) – presents a summary of the key performance assessments that have been performed to demonstrate that the reactor fuel and core will meet the key functional requirements from operability and safety perspectives.
* Fuel and Core Validation Summary (ref. [48]) – presents a summary of the validation status of the methods used to support the design and justification of the Reactor Core.
* Reactor System Examination, Maintenance, Inspection and Testing Strategy (ref. [49]) – outlines relevant EIMT activities for inspection of fuel, monitoring reactivity through core load, monitoring a safe approach to criticality, physics testing, through cycle monitoring at power and Post Irradiation Examination (PIE).
* Justification of Design Limits (ref. [50]) – covers the key safety criteria and their associated design limits for fuel and core.
* Primary Water Chemistry, Minimisation of Fuel Cladding Corrosion (ref. [51]) – describes the expected impacts of the primary circuit chmeistry on the fuel cladding.

1. And at Tier 3, a range of documents (Design basis, Modelling guidelines, Analysis, and Validation documents) covering: Reactor physics; Thermal hydraulics; Criticality; Fuel performance, and Neutron source topic areas. The associated Analysis documents (a subset of the Tier 3 documents) are:

* Reactor Physics Performance Assessment report (ref. [52]) – Analysis of equilibrium cycle for nominal full power depletion.
* Steady State Thermal Hydraulics Performance Assessment report (ref. [53]) – Evaluation of steady state thermal performance.
* Criticality Assessments for In-Core and Storage (ref. [54]) – Calculation of the shutdown margin for a number of core loading sequences.
* Fuel Performance Assessment (ref. [55]) – Summary of the work conducted by the Fuel Vendor (FV) using STAV7.
* Neutron Source Assessment report (ref. [56]) – Provides an overview of the strategy for NSs.
* Also at Tier 3, and essential to underpin the basis of the whole plant design, is the decision record for boron-free operation (Ref. [57]).

# ONR assessment

## Assessment strategy

1. To establish whether the Rolls-Royce SMR F&C design had been developed adequately at GDA Step 2 and against the scope as described within Section ‎1.2, I used the following assessment strategy.

### Targeting

1. I considered whether limitations or constraints originating from the fundamental basis of the design might preclude the RP from delivering its stated safety and performance objectives, including reducing risks ALARP.
2. I identified areas of the design notionally different from a gigawatt scale PWR and considered how the differences might affect the performance of the reactor fuel and in-core SSCs. For example:

* Smaller cores have higher levels of neutron leakage – Is power peaking adequately controlled?
* A boron-free design necessitates more fixed control materials – Can the CRs provide adequate shutdown and holddown?
* KOH and boron-free potentially reduces primary circuit corrosion – Might the changes negatively impact fuel cladding corrosion performance?

1. I sought to understand the RP’s monitoring and protection approaches and whether the F&C system could perform its safety functions under normal and anticipated operating conditions in compliance with the E3S case.
2. Finally, I mapped my target areas to the lifecycle phases of reactor start-up, operation, and shutdown. I used more general headings to cover the fundamentals related to the basis-of-design, and for comment on the overall adequacy of the E3S case and for the adequacy of the computer codes’ validation basis, resulting in:

* Basis-of-design (‎4.2.1)
  + Architecture and sizing of key structures, systems and components (‎‎‎‎4.2.1.1);
  + Relevant Good Practice and ALARP (‎4.2.1.2‎); and
  + Identification and compliance with fuel safety criteria (‎4.2.1.3).
* Reactor start-up (‎4.2.2)
  + Safe approach to criticality (‎‎4.2.2.1).
* Reactor operation (‎4.2.3)
  + Coefficients of reactivity (‎‎4.2.3.1);
  + Load following (‎4.2.3.2);
  + Reactivity control (‎‎‎4.2.3.3); and
  + Impacts of core chemistry on fuel condition (‎‎4.2.3.4).
* Reactor shutdown and post-shutdown operations (‎‎4.2.4)
  + Shutdown margin and reactivity holddown (‎4.2.4.1);
  + Criticality safety during core load/refuel (‎4.2.4.2‎); and
  + On-site post-irradiation fuel management (‎4.2.4.3).
* E3S case (‎4.2.5)
* Adequacy of the computer codes’ validation basis (‎4.2.6)

### Sampling

1. My sampling focussed on Tier 2 documents looking at the high-level claims, arguments and entry-point evidence; and Tier 3 documents, consisting detailed evidence. Although it was at Tier 2 where the majority of the value resided in terms of:

* The RP reporting compliance with their own CAE; and
* Where the RP’s interpretation of their own evidence was being made, allowing for a more direct comparison with my own safety expectations.

1. This approach allowed me to form a judgement on the fundamentals of the RP’s design and E3S case in line with the Step 2 objectives.
2. Whilst my judgements in this assessment are often described in terms of design rather than specifically safety requirements, they can be considered interchangeable. This is because any GDA Statement provided, will be against the design, which is underpinned by the corresponding safety case.
3. My assessment strategy was consistent with my Assessment Plan (ref. [23]).

## Assessment

### Basis-of-design

1. SAP EKP.1 (ref. [19]) states that the underpinning safety aim for any nuclear facility should be an inherently safe design, consistent with the operational purposes of the facility. Similarly, IAEA SSR 2/1 (ref. [32]) Requirement 6 requires that, the plant can be operated safely within the operational limits and conditions for the full duration of its design life.
2. For the RR-SMR, the important aspects which may affect such a demonstration are the physical constraints placed on core and the decision and implications of adopting boron-free operation; and whether these may be incongruous with safely delivering the reactor core’s defined Rated Thermal Power (RTP) of 1,358 MWth.

#### Architecture and sizing of key structures and components

1. The key performance and design parameters are reported upon with the Reactor SDD report (ref. [58]). From my sample, all parameters seem in keeping with the RP’s stated aspirations including core thermal power and are recognisable for a PWR-type design including typical system pressures and core ΔTs. Of particular note is the core size.

**Core size**

1. Core size is ultimately constrained by the diameter of the RPV, which has reportedly been maximised for UK road transport by the RP.
2. Working to this, the RP has decided that a core made up of 121-off 17x17 (289 fuel pin) FAs (ref. [57]) will allow for appropriate core design limits to be met, including shutdown margin (SDM).
3. Achieving adequate SDM under boron-free conditions requires large concentrations of fixed control materials (CRs and other solid burnable absorbers) to be used in the reactor core. This can result in potentially larger CRs and hence larger FAs in which to house them. However, the larger the FAs the less ‘circular’ the core and overall less fuel contained within. For a fixed power, the fuel that is present would need to achieve higher Burn Ups (BU) and may be nearer thermal limits. An analysis of FA numbers, including a reduced number of larger FAs, is reported upon within the RPs boron-free decision record (ref. [57]).
4. From my review of the document (ref. [57]) it is clear that a suitable range of FA options have been considered; working away from the potentially control-material constrained 121-off 17x17 design towards increasingly off-circular larger FA options, including a 69-off 22x22 design.
5. The document provides both qualitative, and some quantitative analysis to underpin the RP’s decision that 17x17 FAs should be used – albeit with options exercised to increase control materials. I found the RP’s arguments and the sensitivity studies exploring the robustness of the conclusions compelling and expect that these will be valuable in supporting a demonstration of having reduced risks ALARP later in GDA and meet my expectations against SAP SC.4.
6. I sampled the RP’s analysis of core design Iteration 5 (It.5) (ref. [47]), developing the arguments on the 17x17 design’s viability. I noted two areas of non-compliance including SDM for the equilibrium core, and a peak pellet BU limit. The former has since been addressed within It.6, and the later may reasonably be subject to a review of the limit being used, subject to FV input; and is in any case pessimistic as 100% continuous through-cycle power operation has been assumed.
7. Considering the contrasting requirements between SDM and BU for this core, I consider the RP’s choice of a 17x17 core design balanced. However, an adequate deterministic demonstration of its performance against appropriately specified and conservative design limits remains outstanding at this stage. This will need to be provided in Step 3.
8. I also expect that the decision of retaining an industry ‘standard’ 17x17 FA mechanical design will reduce a variety of future risks in so far as is reasonably practicable (SFAIRP). For example: qualifying additional FA manufacturing processes; having to provide additional justifications for safe transport and fuel handing; the applicability of OpEx for underpinning core performance, fuel storage and disposal solutions all having to be bespoke, and so on.
9. The implications of the RP’s FA sizing decision on other areas of the SMR design are addressed later in this assessment, including F&C related SSCs such as the CRs and the CH (‎4.2.1.2), and the SFP (‎4.2.4.3).

#### Relevant Good Practice and ALARP

1. TAG-005 (ref. [30]) states that “the determination of control measures to reduce risks to ALARP is part of a dutyholder’s risk assessment. For those risks that are well-known and well-understood, there are often well-established standards for control measures in the form of relevant good practice (RGP) […]. For less well-understood and novel hazards/risks […] the determination of control measures should include more thorough and careful considerations by the dutyholder to ensure a balanced approach to reducing risks to ALARP is achieved”. SAP SC.4 states that “a safety case should explicitly set out the argument for why risks are ALARP”.
2. From my sampling of the E3S case, I am able to observe that much of the SMR’s F&C design is taken from well-established standards from either PWR, or on occasion BWR technologies where applicable. Evidence exists across all areas of the design, from the use of 17x17 FAs (Section ‎4.2.1.1) through to familiar fuel safety criteria and associated design limits (Section ‎4.2.1.3). From my sample, I am content that the RP has identified and is using RGP appropriately, including appropriate design codes/methods (Section ‎4.2.6), standards, and through the identification of Operational Experience (OpEx).
3. Where the identification of directly relevant RGP is potentially less evident is within the decision to adopt boron-free operation. Whilst the RP cites that globally there is a “wealth of reactor experience without soluble boron”, the RP’s own conclusions are restricted to having demonstrated “viability of the boron-free option in principle” (ref. [57]). As this area appears less well-understood, I sought to understand if F&C specific risks resulting from boron-free operation have been identified and whether control measures have been taken to reduce risks, or plans are in place to sentence the risks later.

**Boron-free operation**

1. The RP’s boron-free operational philosophy identifies the use of CRs alone as the means to maintain nominally constant reactivity levels over the cycle. Arguments discounting the use of duty-boron, and a boron-free sub-option of CR’s coupled with reducing cold-leg temperatures (to assist with controlling reactivity through cycle) are made (ref. [57]). These arguments seem justifiable and as an appropriate basis for the RP to develop its design.
2. Removing duty-boron minimises system complexity and provides numerous plant wide benefits, including the removal of Boron Dilution Faults (BDF). However, the F&C design has had to accommodate changes to facilitate these. Not least of which is the need to house larger amounts of fixed control materials.
3. The RP assumes up to 10 wt% Gadolinia (Gd) in the fuel matrix (ref. [59]); and originally identified 109 CRs (ref. [57]) – approximately double that of a similarly sized borated core (the number of CRs has been reduced since then, see Paragraph ‎84. The inclusion of these control materials has consequences:

* High fuel pin Gd concentrations reduce fuel melting temperatures – The RP has identified enrichment cut-back as a means to address this, the extent to be informed by the FV (ref. [50]). Whilst 10% Gd is above what is typically seen in PWRs, the RP’s approach is comparible to BWRs and based upon RGP;
* CR-only control can lead to difficulties in maintaining acceptable through cycle power peaking – The RP identifies an approach of CR sequencing to address this (ref. [60]), again mimicking BWR operation and hence RGP; and
* Excessive numbers of CRs can cause hydraulic congestion in the upper plenum and physical congestion of the CH. My expectation is that CR numbers are optimised (a surrogate for SAP EMC.9 (ref. [19])), where success is measured by a design having adequate but not excessive SDM.

1. CR optimisations have been performed by the RP over several design iterations. The outcome of which is the down selection of a CR mechanical design (exact dimensions outstanding), specification of the control materials to be used, and CR locations having been identified.
2. I view the RP’s stated preference of CRs based on a ‘standard’ PWR mechanical design (ref. [57]) as having been clearly justified and another example of the RP identifying and using RGP to inform their design. The decision also reduces similar future risks SFAIRP as per the use of ‘standard’ FAs (Paragraph ‎73).
3. The RP’s move away from the Hafnium (Hf) (ref. [57]) in favour of B4C, AIC and steel (ref. [60]) also takes the CR design closer to other PWRs.
4. Core design It.5 onward uses these replacement materials and seeks to optimise the CR pattern, reducing CR numbers to 89. A minor non-compliance reported for It.5’s SDM (Paragraph ‎71) (resolved for It.6 – Paragraph‎ ‎186) demonstrates the RP’s optimisation process at work and provides me confidence that a wholesale reduction in CR numbers or changes to CR locations is unlikely to provide material benefit.
5. For un-Borated cores, CRs are generally inserted into the core further and for longer. B4C in particular can suffer from irradiation induced swelling. The RP recognises this risk and identifies annular AIC sections on the tips of B4C and AIC CRs (ref. [60]). I consider provision of this free space a good practice measure that will delay the onset of radial swelling. The RP’s design includes CR sequencing and the implementation of insertion limits, which will also provide benefit to CR lifetimes. The consideration of these approaches alongside the effects of material ageing and degradation will be of value when demonstrating performance against my SAP EAD.2 expectations.
6. In addition to the 89 CRs, each requiring a CRDM and associated CH penetration, there will be 16 in-core instrumentation penetrations (ref. [58]). The implications on CH structural integrity (ref. [44]) are not in scope of my assessment. However, I note that the RP asserts that even with 121 FAs rodded that there would not be an adverse effect on the CH (ref. [57]).
7. Overall, my judgment is that the impact to the F&C design are a necessary and justifiable consequence of the RP pursuing boron-free operation. I can see that the RP has sought to exploit RGP, minimise F&C risks SFAIRP or has provided commitments to do so.

**Potassium hydroxide**

1. As a result of adopting boron-free normal operation, an option existed to change primary circuit alkalising agent from LiOH to KOH. Various benefits are cited by the RP including reduce primary circuit corrosion. Implications on the fuel’s crud and corrosion performance is considered later in this assessment (Section ‎4.2.3.4).

#### Identification and compliance with fuel safety criteria

1. SAP ERC.1 (ref. [19]) states that the identification of safety criteria and compliance with associated design limits, ensure that the reactor’s fundamental safety functions can be delivered with an appropriate degree of confidence for its permitted operating modes.
2. One of the most comprehensive listings of design limits applicable to PWRs, is the OECD NEA Fuel Safety Criteria Technical Review Second Edition (ref. [39]). My expectation is that the RP’s design limits (ref. [50]) compare favourably to those specified within or that variance is suitably reported upon and justified.
3. From my sample I noted that some of the NEA criteria were identified by the RP as not applicable. Given the boron-free design the exclusions are appropriate.
4. Where criteria were yet to be assigned a limit by the RP, either needing test data or FV input, I noted that those aligned with the NEA’s classification of “complete or sufficient information is not available” i.e. identified as being reactor design specific. So again, are appropriate.
5. I also considered the RPs design limits (ref. [50]) against safety criteria selected from TAG-075 (ref. [31]).

* Peak Clad Temperature (PCT) limits: A non-Loss Of Cooling Accident (LOCA) PCT limit is identified (ref. [50]). I recognise the value cited from use within a previous GDA and internationally, and one which should prevent the clad becoming brittle or melting during infrequent fast transients. A LOCA PCT limit is also specified. Again, the value is consistent with international practice; dependant upon comparable cladding hydrogen content being demonstrated. I note the RP currently predicts 20% margin to the LOCA PCT limit (ref. [61]).
* Peak Fuel Temperature: A limit is identified (ref. [50]). The limit cited is markedly lower than unirradiatied UO2 melt temperatures (>2800 C) and the RP has provided assurance that pessimistic BU assumptions were used in its derivation. For Step 3, substantiation of this value will need to be produced to demonstrate that it is pessimistic. A Peak Fuel Temperature for Gd doped fuel was not assigned a value and was marked correctly as requiring FV input. (Note: Values for both design limits have laterly been provided within the RP’s Fuel Performance Analysis document (ref. [55]))
* DNBR is identified for steady-state normal operation (ref. [62]) – I recognise the proposed Westinghouse W3 correlation and the limit cited as being standard industry practice. An Electric Power Research Institute (EPRI) correlation is also specified, for when conditions are outside the range of W3. Recognising that excess margins can constrain design optimisation, I welcome the RP’s intention to work with the FV to identify and apply more tailored correlations and to identify new limits in due course. The RP predicts margin to the DNBR limit for all analysed scenarios apart from for some Control Rod Ejection (CRE) accidents at intermediate powers (ref. [63]); where I agree that options exist for the RP to pull this back into compliance with their design limits.
* The use of the 95/95 criterion is described, as are the numbers of rods allowed to exceed DNBR limits during faults (ref. [61]). The approach mirrors that of other PWR designers’ and operators’ and so constitutes RGP. I note that as part of the CRE analysis (ref. [63]) which resulted in breaching the DNBR limit, fuel failures were predicted to be <10%, which meets the RPs design criteria for (Design Basis Condition) DBC-4 faults, provided such failures are demonstated as ALARP.

1. From my sample I conclude that the fuel design limits proposed are in line with expectations, and I consider that the RP’s analysis has provided me with adequate confidence that the design will be capable of achieving compliance with its design criteria.
2. Compliance with design limits are further sampled and reported upon later within this assessment (Sections ‎4.2.2 to ‎4.2.6).

#### Conclusions on the basis-of design

1. I conclude that appropriate optioneering has been performed by the RP and that through the use of sensitivity studies a basis-of-design has been produced, suitable for future development (SAP SC.4).
2. RGP including appropriate design codes, standards and OpEx have been identified; their relevance is understood, and arrangements are in place to address any shortfalls.
3. The RP's analysis has provided me with adequate confidence that the design will be capable of achieving compliance with its design criteria. Where non-compliances have been identified, options exist to return these to compliance (SAP ERC.1).
4. The approaches taken by the RP has demonstrably reduced, or is on course to reduce risks ALARP, whilst working within the constraints of what the RP considers their commercial opportunity. My judgement is that the SMR is likely capable of being developed further to meet the operational purposes of the facility (SAP EKP.1) and RP’s RTP aspirations.

### Reactor start-up

1. SAP ERC.4 states that “The core should be designed so that parameters and conditions important to safety can be monitored in all operational and design basis fault conditions and appropriate recovery actions taken in the event of adverse conditions being detected”. Whilst UK regulatory expectations are set principally by ONR SAPs, similar expectations exist within IAEA references including SSR 2/1 (ref. [32]), SSG-73 (ref. [37]), and SSG-52 (ref. [35]).

#### Safe approach to criticality

1. Safe reactor start-up involves measuring and monitoring neutron populations, controlling their rate of increase, and facilitation of emergency shutdown should those measurements go outside of specified set points.

**Neutron sources**

1. NSs are used to provide a signal for reactivity monitoring during the initial approach to criticality and during re-starts. NSs are typically classed as either primary, a spontaneous neutron emitter; or secondary, a neutron emitter requiring activation through irradiation which provides a longer-term neutron flux after any primaries have decayed.
2. The RP’s NS decision record (ref. [64]) recommends that primary and secondary NSs are used as a design baseline position but with a view to removing the need for secondary NSs as the design progresses.
3. I am satisfied with the intent to retain primary NSs (used for initial criticality) because the core’s estimated critical rod positions would lack any empirical basis, meaning there would be significant reliance on monitoring over the complete source range; and there would be no intrinsic source present which could justifiably replace them.
4. For the secondary NSs (used for re-starts), arguably a DiD measure, I am satisfied that their removal could be justifiable as long as core conditions can be adequately monitored during re-start – which may come from the core’s intrinsic source; or that suitable and sufficient DiD arrangements are provided elsewhere to adequately compensate for their omission to meet my expectations against SAP EKP.3. Through such a demonstration, risks need not necessarily be increased, whilst dose reductions are achieved through avoiding unnecessary nuclear materials.
5. During re-starts, if additional NSs are required because the primaries have decayed but the intrinsic source is too low, the RP identifies a sub-option of inserting a second primary NS instead of a secondary (ref. [56]). Given the differing dose profiles for each type of NS, I would expect the RP to justify which type constitutes the ALARP option within their safety case to meet my expectation under SAP SC.4.
6. Given that the RP has already committed to look at the number, strength and placement of NS’s based on ALARP arguments (ref. [56]) and has already identified options to maintain DiD for those times when intrinsic source is low, I am content with the RP’s intended implementation of its decision record (ref. [64]).

**Core monitoring**

1. The core’s approach to criticality is monitored by ex-core detectors and is described in two documents: Neutron Source Assessment Issue 1 (ref. [65]), which provides a justification for the use of primary NSs and analyses the initial core’s approach to criticality; and Issue 2 (ref. [56]), which provides a justification for omitting secondary NS and analyses the re-start of an equilibrium core.
2. I sampled Issue 2 (ref. [56]) because at equilibrium the primary NSs will have decayed and in the absence of installed secondary NSs core monitoring will be solely dependent upon intrinsic source. Whereas for the initial core (ref. [65]), primary NS numbers and strengths will be intentionally set to ensure a response at the detectors.
3. I observed that the RP’s analysis uses pessimistic assumptions as described within their analysis methodology (ref. [66]). Only ex-core detectors had been assumed; temporary in-core detectors had not been specified.
4. The analysis showed significant margin to the limits of detectability at the detectors, with the RP concluding that monitoring requirements for the equilibrium core can be achieved with intrinsic source alone. However, it is my judgement that it will likely be a transition core, which may become limiting in terms of analysis, after the primary NSs have decayed but before meaningful intrinsic source has built-up. The ability to monitor the restarts of transition cores, or of the equilibrium core following extended shutdown periods are yet to be performed.
5. For Step 2 I am content with the RP having chosen to focus on the more routine configurations of start-up and equilibrium. However, future analyses will require detector types and locations to be finalised, the rod withdrawal programmes established, and a clear articulation of and a justification for the bounding time in life. Subject to this being deterministically demonstrated my expectations for ERC.4 can then be met.
6. Rod movement sensitivities were also explored as part of the equilibrium analysis (ref. [56]), where it is reported that central control rod movement is unlikely to be detected at Keff <0.91. The RP provides assurance that it is unlikely to result in an inadvertent criticality as the core is so sub-critical.
7. Unless suitably mitigated, the potential for undetected rod movements can reduce a reactor fault tolerance for example, to Control Rod Withdrawal Accidents (CRWA). That said, overall I expect the RR-SMR’s fault tolerance to be improved over many other PWRs due to the removal of boron, as in the absence of boron, start-up no longer requires the combination of rod withdrawal and boron dilution concurrently. Should a fault occur such as a BDF, recovery actions to deliver shutdown require CR actuation alone.
8. The RP commits to modelling fault conditions during start-ups including CWRAs (ref. [56]). The RP also notes options to manage any CRWA risk through mitigations such as central CRs being moved in conjunction with peripheral CRs or including additional DiD measures such as supplementary methods to monitor rod position during start-up. I welcome a demonstration of the fault tolerance of the engineering design and of the effectiveness of the safety measures to allow my expectation against SAP FA.4 to be met.
9. I am satisfied that my expectations against SAP EKP.2 which states that “the sensitivity of the facility to potential faults should be minimised” are capable of being met.

#### Conclusions on reactor start-up

1. The RP has provided arguments for the removal of secondary NSs and a justification to show that core monitoring during start-ups can still be assured (SAP ERC.4).
2. I determined that the analysis presented was conservative and I agree the decision to seek to omit secondary NSs from the design for equilibrium cores is possible whilst still being able to monitor start-up. Further, that the removal of the secondary NSs need not be a reduction in DiD on the basis that there are not necessarily any negative impacts to risk prevention, protection or mitigation (SAP EKP.3). Conversely, risk reduction can be achieved though avoiding additional nuclear materials, potentially moving the design toward a more ALARP position.
3. Transition cores and extend maintenance periods are not covered by the reactor start-up analysis. Should core monitoring not be possible at these times in life the RP concludes some one-off NSs could be used. The RP acknowledges the need to reduce risks ALARP for start-up operations (SAP SC.4).
4. The core’s fault tolerance during start-up should be improved when compared with a borated core, due to the absence of BDFs but reasonably practicable DiD measures do need to be identified and implemented for other potential faults, including CRWAs (SAP EKP.2).

### Reactor operation

1. Within ONR’s guidance, expectations for ensuring safe reactor operation are numerous but most succinctly provided within SAP ERC.1 which states that “The design and operation of the reactor should ensure the fundamental safety functions are delivered with an appropriate degree of confidence for permitted operating modes of the reactor”. SAPs ERC.3 and EKP.1 are also relevant, describing how the core “should be stable in normal operation and should not undergo sudden changes of condition when operating parameters go outside their permitted range”, and that “the underpinning safety aim for any nuclear facility should be an inherently safe design”, respectively.
2. To form a judgement on whether safe operation can be delivered by the RP’s design I assessed the following target areas:

* Coefficients of reactivity (‎4.2.3.1) – acheivement of an inherently safe design capable of self-regulation and load following.
* Load following (‎4.2.3.2) – whether fuel integrity can be maintained in response to sudden changes in operating parameters.
* Reactivity control (‎4.2.3.3) – ability to deliver stable operation through cycle, as the fuel and control materials are burnt-up.
* Core chemistry (‎4.2.3.4) – maintainance of fuel integrity within the specified primary circuit chemistry.

#### Coefficients of reactivity

1. Expanding on SAP EKP.1, TAG-075 (ref. [31]) identifies that “Changes in temperature, coolant voiding, core geometry or the nuclear characteristics of components that could occur in normal operation or fault conditions should not cause uncontrollably large or rapid increases in reactivity”. Important to the demonstration of adequate performance in this area are the reactor’s coefficients of reactivity, which ensure the stability of the reactor.
2. I sampled the RP’s E3S case to check the total power coefficient of reactivity was negative through life, or that any effects of positive reactivity coefficients had been identified and demonstrated as being inconsequential.
3. Within the RP’s Reactor Physics Design Basis (ref. [67]) it states the requirement that reactivity coefficients must be negative for all points in duty operation. To ensure this, the Justification of Design limits (ref. [50]) sets a minimum limit of 0 pcm/K for the most important coefficients; the fast-acting Doppler (Power) Coefficient (DPC) and the relatively slower-acting Moderator Temperature Coefficient (MTC). Compliance with the limits is reported upon within the Reactor Physics Performance Assessment (ref. [52]).
4. All components of the power coefficient of reactivity can be seen to all be negative (ref. [52]). This provides me confidence that the power coefficient of reactivity will also be negative although the values for this though life are yet to be calculated.
5. The RP notes that DPC and MTC design limits will eventually 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 and to ensure acceptable margins to fuel melt during faults (ref. [50]) thereby meeting my fault tolerance expectations against SAP EKP.2.
6. I judge that the coefficients reported upon indicate the ability of the reactor to maintain controlled reactivity such that the core will naturally be tolerant to power increasing faults and maintain load following and self-regulating behaviours.
7. The outstanding work will provide similar assurance for faulted conditions.
8. The RP’s arrangements meet my expectations against SAPs ERC.3 and EKP.1 for Step 2.

#### Load following

1. For F&C, load following is the management of cyclic power changes due to the changing power output demanded from the reactor. IAEA SSG-52 identifies that load following “…may complicate the reliability and ageing assessments of some structures, systems and components” (ref. [35]).
2. During cyclic power changes a FA’s reliability is mostly at risk from Pellet Clad Interaction (PCI) during power increasing manoeuvres; especially if the fuel is deconditioned. The fuel is protected by limiting power ascension ramp-rates. Ageing affects manifest themselves in the longer term; most acutely through exceedance of cladding fatigue usage factors (FUF). FUFs do not affect the ability of the fuel to perform cyclic power changes but rather limits the total number (and extent) of allowable power change over a FA’s operational life.
3. Cyclic power demands are to be met through CR operations (varying in-core temperatures was discounted (ref. [57])) where the core’s coefficients of reactivity will work to return the reactor to a safe equilibrium at the new power (Section ‎4.2.3.1).
4. The RP assumes a notional ramp rate of 3-5% of rated power per minute, which is a value routinely proposed for reactors delivering a load following capability and is in keeping with European Utility Requirements (EUR). The RP expects to meet this target whilst operating rules are implemented to limit the rate of power changes; allowing cladding stresses to relax and aggressive chemical species to dissipate before damage is accrued.
5. To ensure safe operation the RP notes an intention to calculate ramp rates and compare the results against FV specific limits (ref. [46]). Noting the fuel and cladding materials being proposed are standard FV products, subject to the FVs analysis methods used to set the limits having been shown as applicable and adequately validated, this approach seems appropriate.
6. EUR also defines a capability of daily load cycling operation between 50% and 100 % of rated power. The RP identifies that the core will facilitate load following transients as low as 75% and that any further load following will be facilitated through the secondary system i.e. turbine bypass (ref. [58]). I would expect this self-imposed limit to provide some benefit to the fuel through reducing the:

* Overall time spent in load following and hence reduce overall risks to the fuel; which will spend longer in steady-state conditions (at 75%);
* Likelihood of the fuel becoming deconditioned allowing application of a simpler set of operational rules; and
* Number and extent of CR movements required, slowing consumption of the fuel’s FUFs.

1. I am content with this limit although I anticipate that a plant-wide justification will need consideration before it is finally set. Given that the RP is yet to fully define all their requirements for managing cyclic power changes and neither have they detailed their operational procedures, I do not see the potential for this value to change as a significant issue.
2. Whilst an adequate safety analysis remains outstanding, I have not identified anything within the design that should preclude load following operations. The fuel and cladding materials have been used extensively elsewhere meaning their performance should be well characterised; the FV methods should have substantial levels of pre-existing validation data; the reactors coefficients of reactivity are all negative, and should transients be limited down to 75% this would reduce requirements placed against the fuel.
3. I am confident that the RP’s intended approach will allow for the production of a set of load following capability claims and arguments, and dependent upon the results of their analysis, evidence to meet my expectations against SAPs ERC.1.

#### Reactivity Control

1. The amount of core reactivity and exact location of peak reactivity, changes through cycle as the fuel and control materials are burnt up. Ideally, reactivity would be both uniform across a core and stable throughout a cycle. In reality, neutrons are continually lost from the system and core neutronic conditions evolve; with the extent of useful reactivity, available for power production at any point in cycle, being limited by those areas of peak reactivity.
2. Failure to control power peaking can lead to exceedance of thermal limits and fuel integrity being challenged under normal and design basis transient conditions.
3. To maintain stable levels of core-wide reactivity through cycle and in doing also curtail power peaking, duty-boron is routinely used within PWRs. In the absence of boron, the RP has sole reliance on the use of CRs.
4. To improve uniformity across the core, reflectors are used. For the RP, whose core is small with relatively high geometric buckling a heavy metal type of radial reflector is specified, rather than the more routine core-baffle type.
5. Power peaking is generally measured in terms of the heat flux hot channel factor (FQ) and the nuclear enthalpy rise hot channel factor (FΔH). These nuclear data, along with other decoupling criteria, are used in the fault analysis (ref. [41]) and fuel performance calculations to demonstrate safety.

**Control rods**

1. The RP identifies the use of CR sequencing to manage the through cycle depletion of the fuel and control materials. This involves lowering the reactor power at a variety of points throughout the cycle to facilitate the exchange of the inserted CRs for different sets of CRs, before returning back to the desired power. This approach has been informed by RGP, mimicking the operation of BWRs which are also boron-free.
2. CR sequencing and its impacts on power peaking is reported on within the RP’s Reactor Physics Performance Assessment (ref. [52]). Analysis is presented for an 18-month equilibrium cycle for the nominal full power equilibrium xenon (Xe) depletion of It.6 (the nominal depletion) and two lower reactivity states (with higher CR positions) and two higher reactivity states (with lower CR positions).
3. Only the nominal depletion has had its CR sequences optimised, where the maximum FQ and FΔH values are shown to be 2.17 and 1.59 respectively. FQ and FΔH values for the other reactivity states are higher. The RP uses a test case to demonstrate that optimisation will be effective in such instances, reducing one FQ from 2.45 to 2.16.
4. The RP has not yet set acceptance limits for FQ and FΔH. Instead, for Step 2 the RP is reliant on demonstrating compliance with fuel performance fuel design criteria (for normal operation and for powers that remain below the bounding power history) (ref. [55]); and through, showing that relevant safety acceptance criteria such as DNB have been met against a series of bounding postulated faulted conditions (ref. [61]).
5. The results presented by the RP do demonstrate adequate performance. However, the Reactor Physics Performance Assessment (ref. [52]) will need to be extended to cover full range of reactor physics design basis conditions and then fed back into the fuel performance assessments; and a full set of fault analysis will need to be performed to understand the extent of available margin and the core’s sensitivity to power peaking. Only then will the potentially limiting conditions have been demonstrated as being acceptable, allowing my expectations against SAP ERC.3 to be met.

**Radial neutron reflector**

1. The radial neutron reflector is a passive component which sits inside the core barrel, scattering back (reflecting) neutrons, improving the core’s neutron economy and smoothing out radial power peaking. The reflector’s mechanical properties are maintained through cooling provided by core by-pass flow and allows the RPV to be protected from irradiation damage.
2. IAEA SSG-52 (ref. [35]) says that “these structures should be designed […] so that their mechanical performance does not jeopardize the performance of any reactor core safety functions…”.
3. The reflector’s mechanical design is described within the Core Design Description (ref. [60]). It is a Class 2 component (ref. [68]); which I consider appropriate given its function, and note that it is in keeping with the RP’s own classification system.
4. The modelling report (ref. [59]) describes the “expected” density of coolant holes, which are modelled as homogenised compositions with the appropriate proportions of metal and water. This simplification allows the sensitivity to physics parameters to be understood. Accepting that the design has not reached maturity and considering this approach in terms of SAPs AV.1, I am content. However, until the exact location and dimensions of the holes is known and modelled, uncertainty will remain on the extent to which the reflector’s mechanical performance could affect safety functions. Specifically, core by-pass flow affecting core cooling; or the extent to which the cooling holes may influence RPV irradiation (ref. [44]). The RP commits to repeat the analysis with higher fidelity models using water ‘pins’ to actual locations within a solid assembly (ref. [59]).
5. Sensitivity studies have also been performed within the modelling report (ref. [59]) to understand its neutronic performance. From my sample, I was able to observe that the sensitivities studies are informative, each presenting their results in terms of potential impacts to core reactivity (Kinf), allowing their relative importance to be compared and understood. The RP’s application of the sensitivity studies met my expectations regarding SAP AV.6.
6. The RP’s analysis of power peaking accounts for the contribution from the modelling radial reflector (ref. [52]). I observed that the highest power peaking was towards the centre of the core as expected, with the peripheral FAs power peaking having been increased working to smooth-out the profile.
7. The RP commits to comparing their final reflector modelling results with Monte Carlo models. In my judgement this activity would be valuable as Monte Carlo methods are well placed to model phenomena with significant uncertainty, such as solving the neutron transport equation for inhomogeneous materials.
8. The detailed mechanical design of the reflector is not complete at this stage. However, at GDA Step 2 I am satisfied with the extent of the RP’s analysis and use of sensitivities studies to understand its performance. Based on the variation in the parameters analysed by the RP, I have concluded that there is a low risk of cliff-edge effects on the physics behaviour of the core. Further that the reflector will be effective at smoothing out radial power peaking. However, as already committed to by the RP, the RP will need to repeat the analysis with higher fidelity models within GDA Step 3 to understand the actual performance to be expected.

#### Impacts of core chemistry on fuel condition

1. The RPs case for the management of fuel cladding crud and corrosion are described within a chemistry specialism (ref. [40]) badged document titled the Minimisation of Fuel Cladding Corrosion document (ref. [51]). The document seeks to demonstrate that the effects of primary circuit chemistry being based upon boron-free KOH, instead of LiOH and boron are benign.

**Corrosion**

1. Fuel cladding corrosion is directly impacted by the chemistry regime in the form of pH, choice of alkalising agent, quantity of dissolved oxygen present (redox potential) and extent and type of impurities within the primary circuit. These factors not only affect the fuel cladding’s corrosion film thickness but can also affect corrosion film morphology, in particular the thickness of the barrier layer and by inference the fuel cladding’s Hydrogen Pick Up Factor (HPUF); these factors can all affect fuel failure rates.
2. The document (ref. [51]) suggests a reduced corrosion rate and a thicker barrier layer (with reduced HPUF) are possible as a result of adopting KOH chemistry. This conclusion seems reasonable from the data presented but the tests cited are a series of single-effects tests, which do not reflect the exact chemistry regime and nor do they use the correct cladding material (Optimised ZIRLO™). Regarding the cladding, the RP does present some data that supports their claim that performance of their Zr alloy would be consistent with the wider data sets used.
3. In my judgement the document is valuable in terms of a proof of concept. However, a justified prediction of the RP’s fuel’s cladding corrosion performance in the RP’s chemistry remains outstanding and will need to be provided within Step 3 to demonstrate adequate through life margins exist to meet my expectations against SAP EAD.2.

**Crud**

1. Crud is a mixture of corrosion products and debris from the primary circuit that has become affixed to the fuel pin heat transfer surfaces (HTS). Excessive crud can lead to fuel failures through reduced heat transfer, and Crud Induced Localised Corrosion (CILC) can occur if aggressive species accumulate within it. Crud build-up is heavily affected by the primary chemistry, which is the principle means of controlling plant wide corrosion.
2. The document (ref. [51]) suggests reduced crud deposition is possible through achieving reduced primary circuit corrosion. Notwithstanding any management of impurities which may become concentrated in the crud, improvements to clad integrity are claimed from the associated reduction in crud thicknesses. Again, this conclusion seems reasonable from the data presented but again the tests are single-effects tests which do not reflect the exact chemistry regime and nor do they use the correct clad material.
3. Potential for impurity build up in the crud is considered, as is the extent to which Zinc may be dosed into the primary circuit and also become entrained. The RP claims that all the discussed impurities should be removed during power operation by the ion exchange columns, and that a critical Zinc concentration threshold exists below which there would be no detrimental impact to fuel cladding. The reasoned arguments and the analysis presented, which includes some consideration of ZIRLO™ (non-optimised) supports the RPs claims of reduced crud; and subject to appropriate management, potentially a reduction of aggressive species contained within it.
4. I judge the document is valuable in terms of a proof of concept. However, a justified prediction of the RP’s fuel’s cladding crud performance in the RP’s chemistry remains outstanding and will need to be provided within Step 3 to demonstrate adequate through life margins exist to meet my expectations against SAP EAD.2.

**Fuel cladding and fuel cladding performance**

1. Cladding materials available from potential FVs are identified within the document (ref. [51]). Those identified have all been used in PWRs and BWRs previously, have been developed over many years, and have had their in-core performance progressively optimised to balance various fuel integrity risks. The result is a range of products, all with slightly different compositions and microstructures but all with proven corrosion performance. However, the OpEx of their performance is based upon a LiOH environment not KOH, and changing the chemistry severs the link to that empirical data.
2. The RP has provided arguments for why Optimised ZIRLO™ is considered the most suitable of the material presented. The arguments are reasonable. Although, I consider the choice between the not-that-dissimilar cladding materials, none of which are proven in KOH, likely of lower safety significance overall when compared to the RP’s decision to change the primary circuit chemistry.
3. The RP describes that it does not intend to provide quantitative data for the corrosion rates of Optimised ZIRLO™ but instead to produce a qualitative comparison of the claddings in both LiOH and boron, and KOH chemistries upon which it will draw its conclusions.
4. Given that it not reasonably practical for the RP to exhaustively test it’s cladding under all the chemistry conditions possible through life (start-up, normal operation and shutdown chemistries) and the transitions between them, I consider the RP’s intent to perform tests to establish a link to the existing data sets appropriate. A successful campaign could result in the development of a semi-empirical understanding, before PIE can be performed and OpEx obtained from in reactor operations.
5. In collaboration with the FV, a prediction of the extent of fuel crud deposition is planned to be developed by the RP within Step 3. The suitability of the chemistry regime and associated chemistry controls can be evidenced by the thickness, and the make-up of the crud; both of which influence fuel failure risks including CILC. I consider the development of the crud deposition model important to understand these risks, which will have a significant correlation to boiling rates on the fuel’s HTS, may be faster acting than the build-up of uniform corrosion, and as a reasonably practicable measure to undertake.
6. Whilst access to empirical data at the design stage is important, it forms only part of the through-life safety arrangements. SAP EKP.3 requires DiD be considered and the RP acknowledges this by identifying various measures that may be implemented to prevent, protect and mitigate the effects of crud and corrosion build-up longer term. They include:

* Limits for cladding oxidation; PCT; nucleate boiling, and an allowable clad concentration of potassium (ref. [62]);
* Monitoring crud and corrosion build-up by visual inspection of the fuel cladding (ref. [69]);
* On-site and off-site PIE should be considered to ensure crud and corrosion behaviour in a boron-free, potassium hydroxide (KOH) chemistry is as expected and will not lead to fuel performance margin loss (ref. [70]); and
* Identification of fuel cleaning options to remove crud should it form excessively (see ‎4.2.4.3).

1. The application of DiD will be important to ensure that risks related to crud and corrosion deposition rates and thicknesses, morphology, and impurities present with potentially detrimental effects are understood and protected against and risks reduced ALARP.

#### Conclusions on reactor operation

1. Through my assessment of the RP’s reactor during operation I was able to confirm that the coefficients of reactivity were all negative, indicating the ability of the reactor to maintain controlled reactivity (SAP ERC.3). Therefore, the core will naturally be tolerant to power increasing faults and maintain load following and self-regulating behaviours (SAP ERC.1). Furthermore, although the requirements for load following remain largely unset, there are no fundamental reasons why load following should be precluded by the RPs design decisions made to date (SAP ERC.1).
2. I sampled the RPs analysis to confirm that stable reactivity conditions will be maintained through cycle as the core’s fuel and control materials are burnt up. Specifically, I looked at the use of CRs and at the heavy metal radial reflector.
3. The RP’s use of CR sequencing was demonstrated as being capable at reducing power peaking. However, the RP is yet to set design limits for power peaking and until the full range of reactor physics design basis conditions have been considered and additional fault analysis conducted, power peaking performance is yet to fully be demonstrated (SAP ERC.3).
4. The assessment of the reflector demonstrated smoothing out of the radial power profile and the associated sensitivity studies provided me confidence that there does not appear to be a high risk of cliff-edge effects on the physics behaviour of the core (SAP AV.6). However, a lack of maturity in the reflector’s physical design and in the fidelity of its modelling (SAP AV.1) means that core by-pass flow and any prediction for the likely peak neutron fluence that will be incident upon the RPV has a level of unquantified uncertainty associated with it (SAP ERC.1).
5. Regarding the impacts the KOH primary circuit chemistry may have on the fuel’s crud and corrosion though operation, the RP has provided proof-of-concept arguments suggesting potentially improved performance compared with that of LiOH.
6. The RPs choice to use pre-existing cladding materials, should allow for a semi-empirical understanding to be generated through the performance of targeted testing and analysis; mapping the KOH data points generated to the wealth of pre-existing LiOH corrosion test data. For crud, which has a strong correlation to HTS boiling rates, the RP with the FV intends to establish thickness estimates (SAP EAD.2).
7. The RP has identified a set of candidate DiD safety measures to prevent, protect and mitigate the effects of crud and corrosion build-up through EIMT (SAP EKP.3). The RP needs to define what is to be implemented and to demonstrate that risk have been reduced ALARP.

### Reactor shutdown and post-shutdown operations

1. From normal critical operation the reactor can either undergo a planned shutdown or in the event of instrumentation trip-values being reached (or through manual activation) the Scram safety measure can be initiated. Both are intended to place the reactor in a safe state, allowing for reactor system EIMT to locate, understand and rectify any faults, including undertaking refuelling.

#### Shutdown margin and reactivity hold down

1. My expectations for reactor shutdown are informed by SSR 2/1 (ref. [32]) Requirement 46, which states “Reactor shutdown means shall be provided […] and that the shutdown condition can be maintained even for the most reactive conditions of the reactor core”. This is also described within SAP ERC.2 (ref. [19]) which also states that the “means for shutting down the reactor shall consist of at least two diverse and independent systems”.
2. In addition, SAP EDR.4 (ref. [19]) and IAEA SSG-2 (ref. [33]) identify the single failure criterion in design basis accidents. For shutdown this means a demonstration that any single random failure within the safety function i.e. the limiting CR (‘Worst Stuck Rod’ (WSR)) fully withdrawn, will not prevent its success.

**Planned shutdown**

1. For the RP’s design, duty reactivity control including reactor shutdown and reactivity hold down for all operating conditions is performed by actuation of the CRs alone. Duty reactivity control is a safety class 3 system, slow acting and active (motor driven).
2. Because the design is boron-free during normal operation, there is no risk of boron dilution as a reactivity fault. As a result, the RP’s successful demonstration of adequate SDM for the most reactive core state (cold-zero-power (CZP) with no Xe) also provides a demonstration of hold down.
3. Sufficiency of the shutdown is defined by the RP within its Justification of Design Limits document (ref. [50]). I consider these design limits appropriate and recognisable from other operational PWRs designs. Allowances are identified separately for methods’ uncertainties for subcritical conditions (ref. [48]) and for the effects of control rod depletion (ref. [47]) on shutdown.
4. The Reactor Physics Performance Assessment document (ref. [52]) reports upon SDMs for core design It.6. I confirmed the results related to CZP with no Xe, with a WSR and accounted for uncertainties. The results are subject to sensitivity studies to account for variations in preceding cycle length (nominal/short/long). The reported results are compliant with the RPs stated design limits.
5. A subsequent, comparable analysis of core design It.7 (ref. [45]) also reports that full shutdown and hold down is available at all times during the fuel cycles.
6. These results satisfy my expectation against shutdown, hold down, and single failure tolerance i.e. SAPs ERC.2 and EDR.4.

**Scram**

1. The RP identifies the Scram system as a Class 1 safety measure, that will only initiate in the event of instrumentation trip values being reached or if manually activated. Scram is fast and passive (gravity driven).
2. Scram is the first of two independent protective safety measures that provide CoR during faulted operation of the reactor and is described as being compatible with Emergency Core Cooling (ECC) for CoFT.
3. The substantiation of the Scram protective safety measures is presented in a Safety Measure Design Description document (SMDD) (ref. [71]), in which it is described that a key measure of success is to prevent the reactor from reaching the reactivity and thermal limits set on the fuel.
4. I take confidence from the Scram safety measure’s safety functional requirement compliance, reported upon within the SMDD, which routinely reports for the fault analysed that Scram “will successfully initiate to meet acceptance criteria” and that ”the appropriate SDM to CZP with one stuck control rod is demonstrated”. However, I note the many assumptions made in the successful management of transients reliant upon the Scram system including the performance of the ex-core detectors; time to actuation; CR drop times; the core design limits used, etc.
5. A full suite of performance assessment for all faults will need to be performed by the RP, as committed to within their SMDD. I expect that this will remove the assumptions, and demonstrate that insertion of CRs is not impeded in operational states or accident conditions, other than severe accidents as per IAEA SSR-2/1 Requirement 44 (ref. [32]).
6. I confirmed that Scram is intended to fail safe. For example, should the electrical supplies fail to the CRDMs, the CRs will drop in. This approach meets my expectations regarding SAP EDR.1.

**Alternative Shutdown Function**

1. Should the CRs fail to release on a Scram signal (or other scenarios where Scram does not occur) a secondary means of shutdown is provided. The Alternative Shutdown Function (ASF) is identified as a Class 2 safety measure, which injects soluble boron (potassium tetraborate) into the primary circuit (ref. [72]).
2. ASF is active (pumped) but is slower to initiate than Scram. ASF is initiated on failure to shutdown the reactor using the Scram function. ASF is described as compatible with Passive Decay Heat Removal (PDHR) for CoFT. This second protective safety measure to provide CoR during faulted operation of the reactor plant is diverse to the CRs. This approach meets my expectations against ERC.2.
3. The substantiation of the ASF protective safety measures is presented in a SMDD (ref. [72]), in which it is described that a key measure of success is to prevent the reactor from reaching the reactivity and thermal limits set on the fuel.
4. On review of their safety functional requirement compliance within the SMDD, I observed that for the fault analysed for ASF that it is expected to successfully initiate to meet acceptance criteria.
5. I acknowledge that at this stage many assumptions have been made in the successful management of transients reliant upon the ASF system including reliance on negative reactivity feedback effects to counteract any immediate increase in reactivity; the total time for the boron to reach the top of the RPV; boron concentrations, worths, etc. However, I am satisfied that the assumptions made seem appropriate and that a full suite of performance assessment for all faults, specifically including assessment of reactivity faults will be performed by the RP as committed to within their SMDDs.

#### Criticality safety during core load/refuel

1. The RP’s Criticality Assessments for In-Core and Storage document (ref. [54]) provides analysis of the design for criticality safety during refuel. The RP has set an administrative margin for in-core activities to prevent inadvertent criticality. Due account has been made for the performance of the analysis method against critical benchmarks and uncertainties in the use of the method (ref. [73]). An administrative margin to criticality for normal conditions i.e. for a fully built core has also been set (ref. [50]) and is similarly reduced when accounting for uncertainties.
2. The number and position of the ex-core detectors remains to be defined so assumptions have been made about them in the analysis. The calculations were performed at cold temperatures and ambient atmospheric pressure, with no external neutron sources being present to pessimise the results. In-core detectors are not assumed, although it is expected that they will be inserted into the central instrumentation tube of those FAs without CRs present during core load (ref. [74]). The RP is planning to continue to develop the design of the core monitoring systems in their Step 3 analysis.

**Core load/refuel – with no misloads**

1. The RP analysed a core load for the initial and for the more reactive equilibrium core, for seven different core load patterns. Both cores (initial and equilibrium) for all core load patterns showed progressive increases in core reactivity as FAs were sequentially loaded into the core. There were no intermediate fuel patterns that were more reactive than the fully formed core, which meets my expectation against IAEA SSG-73 (ref. [37]).
2. When fully built both initial and equilibrium cores (with no mis-loads) had margin to the administrative margin to criticality for normal conditions meeting my expectations against SAP ERC.1.

**Core mis-load**

1. The most likely cause of a possible inadvertent criticality is when errors in the loading sequence lead to the highest reactivity FAs being placed next to each other. The precise sequence of errors will determine when in the loading sequence an inadvertent criticality could occur.
2. SSG-52 (ref. [35]) says that ”…the reactor core should be designed such that the consequences of the worst misloaded fuel assembly, if any, remain within nuclear design limits and fuel design limits”. The commentary under SAP ERC.1 states that “no single moveable fissile assembly, moderator or absorber when added to or removed from the core should increase the reactivity by an amount greater than the shutdown margin, with an appropriate allowance for uncertainty”.
3. The RP has conducted misload analysis for the more reactive equilibrium core only (ref. [54]) which is bounding from a criticality safety perspective. Transition cores have not yet been analysed.
4. The administrative margin and the uncertainty used for the analysis is consistent with the one stuck rod shutdown margin taken from the Justification of Design Limits (ref. [50]). However, those in-core design limits appear only to be applicable to Modes 1-4, not to refuelling. In response to a Regulatory Query (RQ) (RQ-01169, ref. [75]) the RP advises that further work has already been identified to account for refuelling and cold shutdown conditions.
5. The RP’s analysis states that “The results show that two unrodded assemblies will breach the fault conditions criticality criterion but remain subcritical” (ref. [54]). In response to RQ-01169 (ref. [75]) the RP provided clarification that “this statement refers to two binary swaps of FAs (two rodded assemblies being moved to incorrect locations)” and that “it is expected that a single binary swap could not breach fault condition criticality criterion”.
6. Given the clarification on the type of fault that was analysed, I am content that the results demonstrate that a single fault alone would not result in limits being breached. Hence, I am content that the RP’s design shows consistency with SAP ERC.1 and IAEA SSG-52 (ref. [35]).
7. The analysis goes on to show that resultant impacts to the load sequence could breach the administrative margin for in-core activities (to prevent inadvertent criticality) by the time the core is complete. However, it is also demonstrated that the core can be monitored continuously, consistent with my SAP ERC.4 expectations, and that multiple faults would need to have occurred and not to have been identified throughout the loading process. The RP’s analysis of failures consequential upon the initiating fault, for the sequence core loading sequence modelled meets my expectations against SAP FA.6.
8. Regarding these ‘multiple faults’ and informed by TAG 075 (ref. [31]) and SAP EKP.3 I would expect DiD to be in place to mitigate the risk of inadvertent criticality. I am satisfied that the RP is aware of the need for DiD measures to be provided and intends on demonstrating that risks have been reduced ALARP. I take my confidence from the commitments made in the RP’s Definition of Postulated Initiating Events and Derivation of Initiating Event Frequencies (ref. [76]) to provide appropriate preventative measures to reduce the event frequency to ensure the risks are managed, and from the identification of candidate safety measures within the Fault Schedule (ref. [77]):

* Duty (DiD1) – Operator controls on FA locations according to loading pattern.
* Preventive (DiD2) – Operators check FA numbers, locations and control rods.
* Protective (DiD3) – Operators compare source-range detector response that expected. Engineered protection to be considered.

1. Based on the evidence I have sampled it is clear that the core is tolerant to single failures; that it is possible to monitor the core reactivity continuously, that actions can be taken before criticality limits are breached; and that DiD measures are intended to be implemented to further reduce risks ALARP.

#### On-site post-irradiation fuel management

1. FAs are burnt-up in the core, removed during refuelling, and either returned to the core for subsequent irradiation or stored in the SFP. The capacity of and the capabilities provided for within the SFP are dependent upon the number of FAs generated, their residence time whilst awaiting final disposition, and their as-irradiated condition. Failure to provide adequate safe spent fuel storage may, in time, have a negative impact on the risk profile elsewhere across the plant.
2. My expectations are that RP’s SFP design identifies and accounts for the anticipated throughput of FA and that appropriate risks and activities have been accounted for.

**Spent fuel pool capacity**

1. Sizing of SFPs can be viewed in terms of physical capacity, or in terms of its thermal or criticality safety capacity; any may become limiting preventing FA imports. My assessment considers physical capacity only, with thermal and criticality safety aspects of the SFP within scope of the Mechanical Engineering (ref. [42]) and Radiological Protection and Criticality (ref. [43]) specialisms respectively.
2. The Spent Fuel Storage and Cask Loading System [FAB] SDD (ref. [78]) identifies an array of storage cells 26x20, providing 520 SFP storage positions. This capacity has been rounded up to form a square array from the 507 positions identified in the RP’s Decision Record (ref. [79]) as offering the best combination of cooling time (10 years), footprint, contingency spaces, and compliance with expected safety criteria.
3. Storage positions are provided for spent, partially-spent and new FAs, high-activity/fissile components (e.g. damaged FAs) and low-activity/non-fissile components (e.g. damaged Control Rod Housing Columns). Sufficient spare capacity is provided to perform a full core off-load at any time. There is no specific allowance for CRs as these will be housed in their respective FAs.
4. I found the decision record (ref. [79]) informing this approach thorough and comprehensive, and I consider that the record will be of value in a future demonstration against SAP SC.4 when describing why risks have been reduced ALARP.
5. The OpEx the RP had obtained from an American operator, Constellation, was used to great effect in informing decisions, and analysis of storage provisions from a range of other operators/designers taken from open-source literature was used effectively as benchmarks. The use of such a range of OpEx provides me confidence in the RP’s conclusions, risk identification and risk mitigation.
6. Given the long timescales associated with fuel storage, minor changes to assumptions can have significant consequences on storage capacity. I raised an RQ (RQ-01166, ref. [75]) to explore reliance on those assumptions, informing my view of its safe working life. Of note was that:

* 10 year storage capacity: Through using a mixed fuel age cask loading strategy, some fuel will be cooled for less than 10 years before its removal thereby reducing residency time and providing margin for future operational changes.
* Cask remainders: The capacity assessment includes 23 storage positions for spent fuel that may need to undergo an additional storage cycle in the SFP in order to ensure that each cask (of 24 positions) can be fully occupied. Under a mixed fuel age cask loading strategy, this number is conservative as the ages of fuel within a cask will be more evenly distributed resulting in many of the 23 ‘remainders’ in the SFP being empty.
* The fuel rack to pool liner spacing: At the current design storage pitch the SFP could physically house a further two FAs along the entire perimeter of the fuel storage area, significantly increasing capacity without increasing the SFP size.
* Failed Fuel: An allowance of 24 FAs has been made for failed fuel. In response to a further RQ (RQ-01074, ref. [75]) I was able to confirm that the RP’s failure rates were based upon IAEA data (ref. [38]) and were pessimistic.

1. The RP concludes (RQ-01166, ref. [75]) that “there is therefore no risk of insufficient storage capacity in the SFP in any given cycle”. Given the arguments presented and the provision of the extra 13 positions in forming a square storage array, I agree that future capacity shortfalls are unlikely.
2. The SFP has been classified by the RP as Safety Class 1 (ref. [78]), as it provides the principal means of fulfilling Category A safety functions to support CoFT and CoR. IAEA SSG-15 (ref. [34]) identifies these critical safety functions but also CoRM.
3. From the perspective of the fuel, CoRM implicitly requires the effectiveness of the fuel cladding as a passive barrier to be considered during spent fuel storage. Accordingly, fuel integrity criteria should be specified to take into account the proposed fuel irradiations and appropriate post-irradiation storage and degradation mechanisms. Through response to RQ-01167 (ref. [75]) the RP has committed to the production of appropriate submissions during GDA Step 3. Once produced my expectations related to SAP EAD.1 can be assessed.

**Fuel assembly cleaning**

1. There is a risk that crud may build-up on fuel pin HTSs during irradiation, eroding fuel performance limits. In the first instance crud should be prevented (Section ‎4.2.3.4) but provision should also be made for multiple independent barriers to protect against fault progression i.e. DiD.
2. I sampled the RP’s E3S case for evidence of EIMT being planned, to meet my expectations related to SAP EKP.3 and whether space for performing such risk mitigation had been allocated.
3. The RP’s Fuel and Core PIE Strategy (ref. [69]) specially notes that “extra focus should be made on PIE techniques which identify the quantity and form of clad corrosion and crud deposits” and that “visual Inspection shall also be conducted to determine the presence and magnitude of crud deposition on the fuel pin surface”. I am content that the use of those inspection results will allow either a demonstration of continued compliance with the safety case for the partially burnt up fuel, or inform the need for maintenance in the form of fuel cleaning to return the fuel to compliance.
4. The RP identifies fuel cleaning options to remove crud should it form excessively within its Fuel Cleaning Methodology – Decision Record (ref. [80]). Options considered are ultrasonic, chemical, ice, and water-jet cleaning. I consider that the options reviewed were adequately wide ranging and the down selection of Ultrasonic Fuel Cleaning (UFC) clearly justified.
5. The RP has developed the UFC concept and provided a SDD (ref. [81]). At Definition Review 1 stage (ref. [82]) the UFC equipment’s maturity lags many of the other SSCs. However, I am content because the design of the SFP, where the equipment will be used, only requires some bounding size assumptions to have been made and otherwise the UFC is a stand-alone piece of equipment. Further, with the UFC system intended to be based upon commercial-of-the-shelf technology, not only will there be increased certainty on the size of the equipment but also on the performance levels that can be expected from it. I observe this has been facilitated by the choice of ‘standard' 17x17 FAs (Section ‎4.2.1.1).
6. Finally, I note that the RP intends to deploy the cleaning system within the PIE area of the SFP (ref. [81]).
7. Even in the absence of details on the exact composition or understanding the extent of thickness or adherence of the crud, the RP’s identification of inspections and for a FA cleaning solution to manage this risk is prudent, the technology choice logical, and its design maturity adequate for this stage of GDA.

#### Conclusions on reactor shutdown and post-shutdown operations

1. I performed an assessment of the RP’s arrangements for and during reactor shutdown. Specifically, I looked at SDM and reactivity hold down, criticality safety during core load, and on-site post-irradiation fuel management.
2. For SDM, I concluded that because the design is boron-free during normal operation, it is not subject to the risk of BDFs and as a result, the RP’s successful demonstration of adequate SDM for the most reactive core state also provides a demonstration of hold down.
3. For planned shutdowns and for Scram, the RP showed adequate SDM (SAP ERC.2), including with one-stuck rod, demonstrating single failure tolerance (SAPs EDR.4). Further that the Scram system is intended to fail safe, for example should the electrical supplies fail to the CRDMs, the CRs will drop in (SAP EDR.1).
4. As committed to within their SMDD, a full suite of performance assessment for all faults for Scram and ASF will need to be performed by the RP. Also, a demonstration that insertion of CRs will not be impeded in operational states and in accident conditions, other than severe accidents will also be required (IAEA SSR-2/1 Requirement 44).
5. For core load with no mis-loads, margin to the administrative margin to criticality for normal conditions existed (SAP ERC.1) and there were no intermediate fuel patterns that were more reactive than the fully formed core (IAEA SSG-73).
6. When misloaded, the core was seen to be tolerant to single failures (SAP EDR.4); that it is possible to monitor the core reactivity continuously (SAP ERC.4), that opportunity exists to take action before criticality limits are breached (SAP FA.6); and that DiD measures are intended to be implemented to further reduce risks ALARP (SAP EKP.3).
7. Regarding on-site post-irradiation fuel management arrangements, I assessed the physical capacity of the SFP capacity and the provision of DiD measure in the form of inspection and FA cleaning.
8. I found the decision record informing the capacity and capabilities of the SFP thorough and comprehensive, and consider that the record will be of value in a future demonstration against SAP SC.4 when describing why risks have been reduced ALARP.
9. I consider that the SFP option chosen appears effective at balancing the size and space requirements against the various capacity risks. Overall, and given the arguments presented, I judge that future capacity shortfalls are unlikely.
10. The RP has provisionally identified a FA cleaning capability, should inspections indicate that crud has built up excessively during operation. Planning for this capability is prudent, the technology choice logical, and its design maturity adequate for this stage of GDA.
11. The RP is still to provide design limits for fuel storage that take into account the proposed fuel irradiations and appropriate post-irradiation storage and degradation mechanisms (SAP EAD.1), although this is planned for Step 3.

### E3S case

1. The up issue of the E3S case Chapter 4: Reactor Fuel and Core, from E3S case Version 1 at the start of Step 2 (ref. [83]) to E3S case Version 2 (ref. [3]) at the end of the Step records the extent of progress made by the RP. I performed a consistency check between Version 2 and my assessment and I was able to confirm that its contents reflect the Tier 2 and 3 documents I have sampled and reported upon.
2. I also sampled the RP’s Design Reference Report (ref. [84]) which was consistent with the submissions that I have assessed.
3. From a both a high-level overview of the safety case and through my sampling of it to inform my assessment, I identified no potential gaps, omissions or limitations which may result in a shortfall of sufficient significance that safety case claims may not be supported by an adequate future substantiation.

### Adequacy of the computer codes’ validation basis

1. I used a TSC to review the validation basis of the computer codes intended to be used by the RP. This activity was performed allowing potential gaps in the codes’ validation relative to the RP’s design to be identified, and conclusions to be drawn on the adequacy of the RPs verification and validation (V&V) plans. Failure by the RP to identify gaps and or identify a suitable means to address them may result in difficulties in substantiating the design later in GDA and beyond.
2. For F&C my sample was focused on sub-channel thermal hydraulics, reactor physics and fuel performance.
3. The basis for the assessment was principally the ONR SAPs AV series (Paragraph ‎25) supplemented with appropriate international standards (ref. [33]) and RGP in the form of the TSC’s own experience as a code developer.
4. The review focused on:

* Appropriateness of the codes for the applications proposed by the RP;
* The RP’s general approach to code validation;
* Adequacy of the RP’s Phenomena Identification and Ranking Tables, and Test and Assessments Matrix in support of the planned validation programme; and
* The identification of validation gaps, and the feasibility of reading across existing data to fill those gaps.

1. Some ’open points’ were reported upon by the TSC (ref. [85]) which I intend to follow-up on within my Step 3 assessment. Of note was that:

* The RP should explain the strategy for the V&V of the new Critical Heat Flux correlation and the steps to be taken to validate it.
* The RP should report upon all relevant uncertanties in their calculation of the minimum DNBR.
* The RP should explain the methodology for the quantification of the effect of spacer grids on the flow path in the FA model and thus on the minimum DNBR during transient analyses.
* The RP should provide details on the nodalisation of the model for the fuel performance assessments.
* The RP should explain the influence of the depletion of control rods on relevant parameters such as the shutdown margin, the integral and the differential rod worth.

1. Other open points were identified but were either of a lower potential significance to the design and its V&V or have already been identified elsewhere within this report so are not repeated here.
2. I have discussed all open points originating from the TSC’s report with the TSC and the RP and an RQ (RQ-01289, ref. [75]) was raised to document this. From my review of the RP’s response I am confident that the RP can or has provided the clarifications required, and/or has the means and intent to conduct appropriate activities to address them.
3. From my assessment and considering that of my TSC, I conclude that the RPs overall approach to validation is explained clearly and the current level of maturity of the validation is sufficient for GDA. Also, that should the RP deliver against its V&V strategy and the residual open points be addressed, that there is nothing that should preclude an adequate future demonstration: that the computer codes are being used appropriately and that V&V is in place to underpin their performance; and, that the approach meets my expectations against the ONR SAP AV series.

# Conclusions

## Conclusions

1. This report presents the Step 2 F&C assessment for the GDA of the Rolls-Royce SMR design. The focus of my assessment in this Step was towards the fundamental adequacy of the design and safety case. I have assessed the Tier 1 E3S chapters and relevant supporting documentation provided by Rolls-Royce SMR Limited to form my judgements. I targeted my assessment, in accordance with my assessment plan (ref. [23]), at the content of most relevance to F&C against the expectations of ONR’s SAPs, TAGs and other guidance which ONR regards as relevant good practice.
2. Based upon my assessment, I have concluded the following:

* The identification, location and sizing of key SSCs in the F&C design have been set and in my judgement are suitable to underpin the reactor core’s defined Rated Thermal Power of 1,358 MWth.
* The RP's analysis has provided me with adequate confidence that the design will be capable of achieving compliance with its design criteria. Where non-compliances have been identified, options exist to return these to compliance.
* Appropriate optioneering has been performed by the RP and RGP including appropriate design codes, standards and OpEx have been identified; their relevance are understood, and arrangements are in place to address any shortfalls.
* The ability to omit Secondary NSs from the design, without a reduction in DiD has been demonstrated for the equilibrium core. Subject to analysis of the transition cores and of extended maintenance periods, their complete removal may become justifiable.
* During the approach to criticality, when highly subcritical and when using ex-core detectors alone, movements of the central control rod are not detectable. Reasonably practicable DiD measures need to be implemented by the RP during such times, to ensure the reactor’s fault tolerance.
* Coefficients of reactivity were all shown to be negative, indicating the ability of the reactor to maintain controlled reactivity such that the core will naturally be tolerant to power increasing faults and maintain load following and self-regulating behaviours.
* At this point in the design the requirements for load following remain largely unset. However, there are no fundamental reasons why load following should be precluded by the RPs design decisions made to date.
* The neutronic performance of the radial heavy metal reflector has been demonstrated as being capable of smoothing out of radial power peaking. However, a lack of maturity in the reflector’s physical design and in the fidelity of its modelling means that core by-pass flow and any prediction for the likely peak neutron fluence that will be incident upon the RPV has a level of unquantified uncertainty associated with it.
* The RP’s use of CR sequencing has been demonstrated as being capable at controlling reactivity through cycle and reducing power peaking, when sequencing patterns are optimised. However, the RP is yet to set design limits for power peaking and power peaking performance is yet to fully be demonstrated for all times in life.
* The RP has provided proof-of-concept arguments supporting the claim of improved fuel cladding HTS crud and corrosion performance in potassium based primary circuit chemistry, compared to lithium.
* The RP identifies a host of candidate DiD safety measures to prevent, protect and mitigate the effects of crud and corrosion build-up through EIMT. The RP needs to define what is to be implemented and how they will demonstrate that risk have been reduced ALARP.
* For planned shutdowns and for Scram, the RP’s analysis shows adequate SDM, including with one-stuck rod demonstrating single failure tolerance.
* The design is boron-free during normal operation so a demonstation of SDM for the most reactive core state also ensures hold down.
* The Scram system is fail safe, for example should the electrical supplies fail to the CRDMs, the CRs will drop in.
* A demonstration that insertion of CRs will not be impeded in operational states and in accident conditions, other than severe accidents is required.
* The ability to monitor core load was demonstrated by making assumptions about ex-core neutronic detector performance. In-core detectors were not assumed. The RP will consider temporary in-core monitoring for inclusion in later design development.
* For a core load with misloaded fuel, the equilibrium core was seen to be tolerant to single failures. The ability to monitor core conditions was demonstrated but various assumptions were required about detector performance which remain at a low maturity. It was also seen that opportunity exists to take action before criticality limits are breached; and that DiD measures are intended to be implemented to reduce risks ALARP.
* The SFP option chosen appears effective at balancing the size and space requirements against the various capacity risks.
* The RP is still to produce design limits for fuel storage that take into account the proposed fuel irradiations and appropriate post-irradiation storage and degradation mechanisms.
* The RP has provisionally identified a FA cleaning capability should crud build-up excessively during operation. Provision of this capability is prudent and the technology choice logical.
* The RP‘s overall approach to validation is explained clearly and the current level of maturity of the validation is sufficient for GDA. There is nothing that should preclude an adequate future demonstration that the computer codes are being used appropriately and that V&V is in place to underpin their performance.
* The up issue of the E3S case Chapter 4: Reactor Fuel and Core, from Issue 1 at the start of Step 2, to Issue 3 at the end of the Step 2 records the extent of progress made by the RP. Issue 3 reflects the Tier 2 and 3 documents I have sampled.
* From my sampling of the RP’s safety case I identified no potential gaps, omissions or limitations which may result in a shortfall of sufficient significance that safety case claims may not be supported by an adequate future substantiation.

1. Overall, based on my assessment to date, and subject to the provision and assessment of suitable and sufficient supporting evidence, I have not identified any fundamental safety shortfalls that could prevent ONR permissioning the construction of a power station based on the generic Rolls-Royce SMR design.

## Recommendations

1. My recommendations are as follows:

* Recommendation 1: ONR should consider the outcomes from my assessment as part of the decision to progress to Step 3 of GDA for the generic Rolls-Royce SMR design.

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# Appendix 1 – Relevant SAPs considered during the assessment

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| SAP No. | SAP Title |
| AV.1 | Theoretical models |
| AV.6 | Sensitivity studies |
| EAD.1 | Safe working life |
| EAD.2 | Lifetime margins |
| EKP.1 | Inherent safety |
| EKP.2 | Fault tolerance |
| EKP.3 | Defence in depth |
| ERC.1 | Design and operation of reactors |
| ERC.2 | Shutdown systems |
| ERC.3 | Stability in normal operation |
| ERC.4 | Monitoring of parameters important to safety |
| FA.4 | Fault tolerance |
| FA.6 | Fault Sequences |
| SC.4 | Safety case characteristics |