Office for Nuclear Regulation

New Reactors Division – Generic Design Assessment

Step 4 Assessment of Fuel and Core Design for the UK HPR1000 Reactor

Assessment Report ONR-NR-AR-21-021
Revision 0
January 2022

EXECUTIVE SUMMARY

This report presents the findings of my assessment of the Fuel and Core aspects of the UK HPR1000 reactor design undertaken as part of the Office for Nuclear Regulation's (ONR) Generic Design Assessment (GDA). My assessment was carried out using the Pre-Construction Safety Report (PCSR) and supporting documentation submitted by the Requesting Party (RP).

The objective of my assessment was to make a judgement, from a Fuel and Core perspective, on whether the generic UK HPR1000 design could be built and operated in Great Britain, in a way that is acceptably safe (subject to site specific assessment and licensing), as an input into ONR's overall decision on whether to grant a Design Acceptance Confirmation (DAC).

The scope of my GDA assessment was to review the safety aspects of the generic UK HPR1000 design by examining the claims, arguments and supporting evidence in the safety case. My GDA Step 4 assessment built upon the work undertaken in GDA Steps 2 and 3, and enabled a judgement to be made on the adequacy of the Fuel and Core information contained within the PCSR and supporting documentation.

My assessment focussed on the adequacy of the following aspects of the generic UK HPR1000 safety case:

- the reactor core nuclear design, reactor core thermal hydraulic design and reactor fuel system thermo-mechanical design;
- reactor fuel and core data provided for the purpose of design basis analysis;
- parts of the reactor core safety case in which interactions between nuclear, thermal hydraulic and / or thermo-mechanical phenomena in the reactor core are particularly important;
- parts of the reactor core safety case associated with how the UK HPR1000 plant will be operated;
- evidence underlying the validity of the reactor fuel and core computer codes used; and
- the explicit demonstration that the reactor fuel and core designs reduce risks As Low As Reasonably Practicable (ALARP).

The conclusions from my assessment are:

- the reactor core nuclear design, reactor core thermal hydraulic design and reactor fuel system thermo-mechanical design are generally adequate when judged against the Safety Assessment Principles (SAPs) and international relevant good practice;
- reactor core data provided for the purpose of design basis analysis are adequate when judged against the SAPs but I identified shortfalls in the evidence underlying a small selection of the fuel acceptance criteria, which should be addressed by the licensee;
- the parts of the reactor core safety case I sampled in which interactions between nuclear, thermal hydraulic and / or thermo-mechanical phenomena in the reactor core are particularly important are largely adequate, but I identified areas where further work is needed to fully substantiate the RP's claim that a coolable geometry will be maintained in the unlikely event of a LB-LOCA;
- the parts of the reactor core safety case I sampled associated with how the UK HPR1000 plant will be operated are largely adequate for the purpose of GDA, but I identified shortfalls against relevant good practice in the strategy for failed fuel management during operation;

- further work is required post-GDA to ensure that all safety case assumptions and requirements associated with operating rules, commissioning and Examination, Maintenance, Inspection and Testing (EMIT) are sufficiently clear and traceable;
- the evidence underlying the validity of the reactor fuel and core computer codes and associated documentation are generally adequate for the purposes of their specific applications in the fuel and core safety case in GDA; and
- an explicit demonstration that the reactor fuel and core designs reduce risks ALARP has been provided, which addresses the key expectations for new reactor designs within ONR ALARP guidance and is adequate for GDA.

These conclusions are based upon the following factors:

- a detailed and in-depth technical assessment, on a sampling basis, of a wide scope of safety submissions at all levels of the hierarchy of the generic UK HPR1000 safety case documentation;
- independent reviews and analysis of key aspects of the generic safety case undertaken by Technical Support Contractors (TSC); and
- detailed technical interactions on many occasions with the RP, alongside the assessment of the responses to the substantial number of Regulatory Queries (RQ) and Regulatory Observations (RO) raised during the GDA.

A number of matters remain, which I judge are appropriate for a licensee to consider and take forward in its site-specific safety submissions. These matters do not undermine the generic UK HPR1000 design and safety submissions, but are primarily concerned with the provision of site-specific safety case evidence which will become available as the project progresses through the detailed design, construction and commissioning stages. I have captured these matters in 11 Assessment Findings.

Overall, based on my assessment undertaken in accordance with ONR's procedures, the claims, arguments and evidence laid down within the PCSR and supporting documentation submitted as part of the GDA process present an adequate safety case for the generic UK HPR1000 design. I recommend that from a Fuel and Core perspective a DAC may be granted.

LIST OF ABBREVIATIONS

ALARP	As Low As Reasonably Practicable
ATWS	Anticipated Transient Without Scram
ASME	American Society of Mechanical Engineers
BAT	Best Available Techniques
BMS	Business Management System
BOC	Beginning of Cycle
BSL	Basic Safety Level (in SAPs)
BSO	Basic Safety Objective (in SAPs)
C&I	Control and Instrumentation
CGN	China General Nuclear Power Corporation Ltd
CHF	Critical Heat Flux
CILC	Crud Induced Localised Corrosion
CIPS	Crud Induced Power Shift
CRGA	Control Rod Guide Assembly
CVCS	Chemical and Volume Control System
DAC	Design Acceptance Confirmation
DBA	Design Basis Analysis
DBC	Design Basis Condition
DEC	Design Extension Condition
DMGL	Delivery Management Group Lead
DNB	Departure from Nucleate Boiling
DNBR	Departure from Nucleate Boiling Ratio
ELPO	Extended Low Power Operation
EMIT	Examination, Maintenance, Inspection and Testing
EOC	End of Cycle
F_{dH}	Enthalpy rise hot channel factor
FFRD	Fuel Fragmentation, Relocation and Dispersal
FMEA	Failure Modes and Effects Analysis
F_Q	3D heat flux hot channel factor
GDA	Generic Design Assessment
GNI	General Nuclear International Ltd.
GNSL	General Nuclear System Ltd.
GRS	Gesellschaft Fuer Reaktor Sicherheit
HIC	High Integrity Component
IAEA	International Atomic Energy Agency
IB-LOCA	Intermediate Break LOCA
IEF	Initiating Event Frequency

JEFF	Joint Evaluated Fission and Fusion File
JENDL	Japanese Evaluated Nuclear Data Library
LB-LOCA	Large Break LOCA
LOCA	Loss of Coolant Accident
M/P	Measured-to-Predicted
MDSL	Master Document Submission List
MOC	Middle of Cycle
MTC	Moderator Temperature Coefficient
NEA	Nuclear Energy Agency (within OECD)
NSS	Nuclear Sampling System
OECD	Organisation for Economic Cooperation and Development
ONR	Office for Nuclear Regulation
OpEx	Operating Experience
PCER	Pre-construction Environmental Report
PCI	Pellet Clad Interaction
PCMI	Pellet Clad Mechanical Interaction
PCSR	Pre-construction Safety Report
PCT	Peak Clad Temperature
PFC	Primary Frequency Control
PSA	Probabilistic Safety Analysis
PWR	Pressurised Water Reactor
RAFPE	Radially Averaged Fuel Pellet Enthalpy
RCCA	Rod Cluster Control Assembly
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RGP	Relevant Good Practice
RO	Regulatory Observation
RP	Requesting Party
RPS	Reactor Protection System
RPV	Reactor Pressure Vessel
RQ	Regulatory Query
SAP(s)	Safety Assessment Principle(s)
SCC	Stress Corrosion Cracking
SCCA	Stationary Core Component Assembly
SDM	Shutdown Margin
SED	Strain Energy Density
SFAIRP	So Far As Is Reasonably Practicable
SFIS	Spent Fuel Interim Storage
SFR	Safety Functional Requirement

SG	Steam Generator
SoDA	(Environment Agency's) Statement of Design Acceptability
SPND	Self-Powered Neutron Detector
SSC	Structures, Systems and Components
TAG	Technical Assessment Guide(s)
TSC	Technical Support Contractor
US NRC	US Nuclear Regulatory Commission
WENRA	Western European Nuclear Regulators' Association
WIMS	Winfrith Improved Multigroup Scheme
WLUP	WIMS-D Library Update Programme
V&V	Verification and Validation

TABLE OF CONTENTS

1 INTRODUCTION 9
1.1 Background 9
1.2 Scope of this Report 9
1.3 Methodology 10
2 ASSESSMENT STRATEGY 11
2.1 Assessment Scope 11
2.2 Sampling Strategy 11
2.3 Out of Scope Items 12
2.4 Standards and Criteria 13
2.5 Use of Technical Support Contractors 14
2.6 Integration with Other Assessment Topics 15
3 REQUESTING PARTY'S SAFETY CASE 17
3.1 Introduction to the Generic UK HPR1000 Design 17
3.2 The Generic UK HPR1000 Safety Case 18
3.3 Requesting Party Organisation and Information-Sharing 20
4 ONR ASSESSMENT 21
4.1 Structure of Assessment Undertaken 21
4.2 Core Nuclear Design 22
4.3 Neutronic and Kinetics Data for use in Fault Analysis 33
4.4 Fuel System Design 40
4.5 Fuel Design Criteria 54
4.6 Thermal Hydraulic Design and Criteria 65
4.7 Protection Against Pellet Clad Interaction 76
4.8 Fuel Behaviour in a Loss of Coolant Accident 81
4.9 Fuel Deposits 91
4.10 Core Mis-loading and In-Core Neutron Detectors 97
4.11 Management of Failed Fuel in Operation 105
4.12 Operating Limits and Conditions, Commissioning and EMIT 108
4.13 Computer Code Validity 112
4.14 Demonstration that Relevant Risks Have Been Reduced to ALARP 128
4.15 Consolidated Safety Case 135
4.16 Comparison with Standards, Guidance and Relevant Good Practice 137
5 CONCLUSIONS AND RECOMMENDATIONS 138
5.1 Conclusions 138
5.2 Recommendations 139
6 REFERENCES 140

Table(s)

Table 1: Work Packages Undertaken by TSCs

Annex(es)

Annex 1: Relevant Safety Assessment Principles Considered During the Assessment
Annex 2: Assessment Findings
Annex 3: PCSR Chapter 5 Document Hierarchies

1 INTRODUCTION

1.1 Background

1. This report presents my assessment conducted as part of the Office for Nuclear Regulation (ONR) Generic Design Assessment (GDA) for the generic UK HPR1000 design on the topic of Fuel and Core.

2. The UK HPR1000 is a Pressurised Water Reactor (PWR) design proposed for deployment in the UK. General Nuclear System Ltd (GNSL) is a UK-registered company that was established to implement the GDA on the UK HPR1000 design on behalf of three joint requesting parties (RP), i.e. China General Nuclear Power Corporation (CGN), EDF SA and General Nuclear International Ltd (GNI).

3. GDA is a process undertaken jointly by the ONR and the Environment Agency. Information on the GDA process is provided in a series of documents published on the joint regulators' website (www.onr.org.uk/new-reactors/index.htm). The outcome from the GDA process sought by the RP is a Design Acceptance Confirmation (DAC) from ONR and a Statement of Design Acceptability (SoDA) from the Environment Agency.

4. The GDA for the generic UK HPR1000 design followed a step-wise approach in a claims-argument-evidence hierarchy which commenced in 2017. Major technical interactions started in Step 2 which focussed on an examination of the main claims made by the RP for the UK HPR1000. In Step 3, the arguments which underpin those claims were examined. The Step 2 reports for individual technical areas, and the summary reports for Steps 2 and 3 are published on the joint regulators' website. The objective of Step 4 was to complete an in-depth assessment of the evidence presented by the RP to support and form the basis of the safety and security cases.

5. The full range of items that form part of ONR's assessment is provided in ONR's GDA Guidance to Requesting Parties (Ref. 1). These include:

- Consideration of issues identified during the earlier Step 2 and 3 assessments.
- Judging the design against the Safety Assessment Principles (SAPs) (Ref. 2) and whether the proposed design ensures risks are As Low As Reasonably Practicable (ALARP).
- Reviewing details of the RP's design controls and quality control arrangements to secure compliance with the design intent.
- Establishing whether the system performance, safety classification, and reliability requirements are substantiated by a more detailed engineering design.
- Assessing arrangements for ensuring and assuring that safety claims and assumptions will be realised in the final as-built design.
- Resolution of identified nuclear safety and security issues, or identifying paths for resolution.

6. The purpose of this report is therefore to summarise my assessment in the Fuel and Core topic, which provides an input to the ONR decision on whether to grant a DAC or otherwise. This assessment was focused on the submissions made by the RP throughout GDA, including those provided in response to the Regulatory Queries (RQ) and Regulatory Observations (RO) I raised. Any ROs issued to the RP are published on the GDA joint regulators' website, together with the corresponding resolution plans.

1.2 Scope of this Report

7. This report presents the findings of my assessment of the Fuel and Core aspects of the generic UK HPR1000 design undertaken as part of GDA. I carried out my assessment using the Pre-construction Safety Report (PCSR) (Ref. 3) and supporting

documentation submitted by the RP. My assessment was focussed on considering whether the generic safety case provides an adequate justification for the generic UK HPR1000 design, in line with the objectives for GDA.

1.3 Methodology

8. The methodology for my assessment follows ONR's guidance on the mechanics of assessment (Ref. 4).

9. My assessment was undertaken in accordance with the requirements of ONR's How2 Business Management System (BMS). ONR's SAPs, together with supporting Technical Assessment Guides (TAG), were used as the basis for my assessment. Further details are provided in Section 2. The outputs from my assessment are consistent with ONR's GDA Guidance to RPs (Ref. 1).

2 ASSESSMENT STRATEGY

10. The strategy for my assessment of the Fuel and Core aspects of the generic UK HPR1000 design and safety case is set out in this section. This identifies the scope of the assessment and the standards and criteria that have been applied.

2.1 Assessment Scope

11. A detailed description of my approach to this assessment can be found in assessment plan ONR-GDA-UKHPR1000-AP-19-013 (Ref. 6).

12. I considered all the main submissions within the remit of my assessment scope, to various degrees of breadth and depth. I chose to concentrate my assessment on those aspects that I judged to have the greatest safety significance. My assessment was also influenced by the claims made by the RP, my previous experience of similar designs for other reactors, and gaps I identified in the original submissions made by the RP. A particular focus of my assessment has been the relevant RQs and ROs raised, and the resolution thereof. I raised an RQ where more information was required to allow me to reach a judgement about a claim or argument in the safety case. I raised an RO where I identified a potential shortfall in the safety case that I judged required significant work to address and would benefit from a greater degree of regulatory control over the programme to resolution.

13. The Fuel and Core safety case includes documentation describing and substantiating the fuel design and core designs themselves, as well as documentation presenting specialist analysis and data to support other parts of the safety case, including fault analyses. I have assessed both these aspects of the safety case.

2.2 Sampling Strategy

14. In line with ONR’s guidance on the mechanics of assessment (Ref. 4) and informed by other relevant standards and guidance identified in subsection 2.4, I chose a sample of the RP’s submissions to undertake my assessment. The main themes I considered were the nuclear design of the core, the thermal hydraulic design of the core and the thermo-mechanical design of the fuel system. By considering these three themes I ensured that my assessment of the engineering designs of the fuel and core were comprehensive.

15. Within each of these three main themes, I considered the adequacy of the design bases used, the safety margins presented and the limits or performance data provided for use in fault analysis. I also considered the adequacy of the evidence underlying the computer codes used in the analyses. In all of these areas, I have reviewed the safety case documentation and then sampled more deeply in specific areas of greater safety significance or where I observed potential shortfalls.

16. These three main themes of fuel and core design cannot be considered in isolation of each other in the safety case because nuclear, thermal hydraulic and thermo-mechanical phenomena all interact. Therefore, within my assessment I have also sampled areas of the safety case where these interactions are particularly important and in which I initially observed potential shortfalls, to gain confidence that any shortfalls were adequately addressed in GDA. These topics include safety analyses associated with Pellet Clad Interaction (PCI), fuel modelling in a Loss of Coolant Accident (LOCA) and, in conjunction with Chemistry inspectors, the impact of fuel deposits (‘crud’).

17. I have also considered parts of the fuel and core safety case associated with how the UK HPR1000 plant will be operated. This included the reduction of risks posed by core mis-loads (due to their high safety significance), the categorisation of in-core flux

monitoring functions (due to the use of monitoring methods that have been developed newly by the RP for HPR1000) and the strategy for management of failed fuel (which I initially observed was not coherently presented). I also examined whether safety case assumptions about operating limits and conditions, examination, maintenance, inspection and testing (EMIT) activities and physics tests were clearly identified.

18. Finally, I have sampled the RP's overall demonstration that the fuel and core designs reduce all relevant risks to ALARP.

19. Backed by relevant standards and guidance (subsection 2.4), I judge that this strategy has allowed me to reach a well-informed view on the overall adequacy of the fuel and core designs and safety case, and therefore whether or not to recommend provision of a DAC. I have also ensured that the strategy adequately covers the list of shortfalls for follow-up that were identified in the GDA Step 3 assessment and captured in Ref. 5.

2.3 Out of Scope Items

20. The following items were outside the scope of my assessment:

- Visits to the fuel fabrication facility to verify that arrangements in place to control the manufacturing process and product quality are adequate to support the assumptions of the safety case. I became aware early in Step 4 that the location of manufacture for UK HPR1000 fuel has not yet been selected. It will be for the licensee and its fuel supplier to decide where fuel will be manufactured, and what arrangements will be used to ensure the manufacturing processes and product quality are adequately controlled.
- Visits to the RP's safety case analysis team(s) to verify that arrangements for quality assurance associated with technical analysis, computer code and data management are adequately implemented. In mitigation, I have been able to gain confidence in correct application of some of the codes for GDA through a confirmatory analysis programme. It will be for the licensee to decide what computer codes it utilises to support its safety case and operations, and what arrangements it uses to control their use and ensure their adequacy.
- Assessment of the consequences on pressure drop and heat transfer in the fuel assembly of any debris introduced via coolant re-circulated from containment under cooling conditions after a LOCA. For GDA, I have excluded this topic from my scope because the design and supply of system filters and insulation, which influence the debris source term, have not yet been finalised. Some of the wider aspects of this topic are considered within the Fault Studies Assessment Report (Ref. 7).
- Assessment of fuel behaviour in faults in which significant core melt is predicted. ONR's assessment of this topic is within the scope of the Severe Accident Analysis Assessment Report (Ref. 8).
- Assessment of the criticality safety case for storage of new and irradiated fuel in the fuel building. ONR's assessment of this topic is within the scope of the Radiation Protection and Criticality Assessment Report (Ref. 9).
- Assessment of any safety case or documentation associated with use of mixed oxide (MOX) fuel, or any other fuel except for low-enriched UO_2 fuel. The entirety of the UK HPR1000 Fuel and Core safety case submitted in GDA is based on the use of low-enriched UO_2 fuel, described further in Section 3.
- Assessment of the safety case for the transition (second and third) cycle core designs. To build confidence in the fundamental design of UK HPR1000 and in accordance with advice in the GDA technical guidance (Ref. 10) I have focussed my assessment of the UK HPR1000 nuclear design on the first cycle and equilibrium cycle core designs. However, the RP have provided some information for the transition cycles and I have used this to inform my judgements about the adequacy of the wider safety case where appropriate.

- Assessment of any safety case or documentation associated with shorter or longer equilibrium cycle designs than 18 months. My detailed assessment in this report is all based upon the 18-month equilibrium cycle that forms part of the design reference for GDA.
- Assessment of fully mature fuel safety case information for Spent Fuel Interim Storage (SFIS) operations. The SFIS design presented in GDA is at concept stage and the scope of my assessment, described in sub-section 4.5.1.2, was limited accordingly.

2.4 Standards and Criteria

21. The relevant standards and criteria adopted within this assessment are principally the SAPs, TAGs, relevant national and international standards, and Relevant Good Practice (RGP) informed from existing practices adopted on nuclear licensed sites in Great Britain. The key SAPs and any relevant TAGs, national and international standards and guidance are detailed within this subsection. RGP, where applicable, is cited within the body of the assessment.

2.4.1 Safety Assessment Principles

22. The SAPs (Ref. 2) constitute the regulatory principles against which ONR judge the adequacy of safety cases. A full list of the SAPs I applied within my Fuel and Core assessment is included within Annex 1 of this report.

23. The key SAPs applied within my assessment were EKP.1, EKP.2, EKP.3, EKP.4, EAD.1, EAD.2, ERC.1, ERC.2, ERC.3, ERC.4, FA.7, AV.1, AV.2 and AV.3.

24. The EKP and ERC SAPs set engineering principles that are fundamental to the design of the reactor core and fuel. The EAD SAPs are important because of the significant way in which fuel and core performance evolve during operation as fuel is depleted, as well as the degradation mechanisms that affect fuel assemblies under irradiation. SAP FA.7 is important because it sets an expectation that fault consequences be predicted conservatively, which has implications for all of the fuel and core data and methods used in fault analyses. The AV SAPs are important because 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 set expectations for assuring the validity of data and models used within, or outputted by, such codes.

25. Specific advice is given to inspectors in NS-TAST-GD-075 (Ref. 11) on the interpretation of the EKP and ERC SAPs for fuel and core design. In many places in this report, I have referred to the advice in NS-TAST-GD-075 rather than to the EKP SAPs because the advice in Ref. 11 is more directly relevant to my assessment.

2.4.2 Technical Assessment Guides

26. The following TAGs were used as part of this assessment:

- NS-TAST-GD-005, Guidance on the Demonstration of ALARP (Ref. 12)
- NS-TAST-GD-006, Design Basis Analysis (Ref. 13)
- NS-TAST-GD-042, Validation of Computer Codes and Analysis Methods (Ref. 14)
- NS-TAST-GD-075, Safety of Nuclear Fuel in Power Reactors (Ref. 11)
- NS-TAST-GD-081, Safety Aspects Specific to Storage of Spent Nuclear Fuel (Ref. 15)
- NS-TAST-GD-094, Categorisation of Safety Functions and Classification of Structures, Systems and Components (SSC) (Ref. 16)
- NS-TAST-GD-096, Guidance on Mechanics of Assessment (Ref. 4)

2.4.3 National and International Standards and Guidance

27. The following standards and guidance were used as part of this assessment:

- International Atomic Energy Agency (IAEA) Specific Safety Requirements SSR-2/1, Safety of Nuclear Power Plants: Design (Ref. 17)
- IAEA Specific Safety Guide SSG-2, Deterministic Safety Analysis for Nuclear Power Plants (Ref. 18)
- IAEA Specific Safety Guide SSG-52, Design of the Reactor Core for Nuclear Power Plants (Ref. 19)
- Western European Nuclear Regulators' Association (WENRA) Report, Safety of new NPP designs (Ref. 20)

28. The relevant SAPs and TAGs have been benchmarked against IAEA and WENRA guidance available at the time of publication. In particular, Ref. 11 and Ref. 13 were both updated in 2020 and explicitly benchmarked against IAEA guidance in Ref. 18 and Ref. 19, which were published in 2019. Throughout most of Section 4 this report I have taken credit for this and referred primarily to the SAPs and TAGs rather than directly to international guidance. However, I have referred specifically to IAEA guidance where it provides the most directly relevant advice on some topics.

29. I also used Ref. 19 alongside Ref. 11 as a benchmark when developing my assessment scope and sampling strategy (subsection 2.2).

30. I have used some other international sources of advice and Operating Experience (OpEx) to inform my judgments recorded in this report, but which I would not consider to be established standards. They are referred to where applicable in Section 4. An example I have used multiple times is the Organisation for Economic Co-operation and Development / Nuclear Energy Agency (OECD/NEA) Nuclear Fuel Safety Criteria Technical Review (Ref. 21).

2.5 Use of Technical Support Contractors

31. It is usual in GDA for ONR to use Technical Support Contractors (TSC) to provide access to independent advice and experience, analysis techniques and models, and to enable ONR's inspectors to focus on regulatory decision making.

32. Table 1 sets out the areas in which I used TSCs to support my assessment. I required this support to provide independent advice and experience on the topic of code validation as well as access to analysis tools and infrastructure for confirmatory analyses.

Table 1: Work Packages Undertaken by TSCs

Number	Description
1	ONR396 – Confirmatory Analyses. This included a set of confirmatory core physics analyses to: 1) Provide data for comparison with the output from the RP's core physics analyses to provide greater confidence in the neutronics modelling used in UK HPR1000. 2) Provide transient data for use in confirmatory Fault Studies analyses for comparison with the output from the RP's Fault Studies analyses (supporting Fault Studies inspectors' assessments).

Number	Description
2	ONR385 – Code Documentation Reviews. This included a set of independent reviews of the documentation and evidence associated with the following computer codes to support my judgments about code validity: PINE, COCO, POPLAR, BIRCH, PALM, LINDEN.

33. Whilst the TSC undertook detailed technical reviews, this was done under my direction and close supervision. The regulatory judgment on the adequacy, or otherwise, of the generic UK HPR1000 safety case in this report has been made exclusively by ONR.

34. The output of Work Package 1 is summarised in Ref. 22. Ref. 22 references out to additional underlying reports, which I have also referenced where relevant within Section 4 of this report. The outputs of Work Package 2 are reported in separate documents for each code, which I have referenced in sub-section 4.13 of this report.

2.6 Integration with Other Assessment Topics

35. GDA requires the submission of an adequate, coherent and holistic generic safety case. Regulatory assessment cannot be carried out in isolation as there are often issues that span multiple disciplines. I have therefore worked closely with a number of other ONR inspectors and the Environment Agency to inform both their and my assessments. The key interactions in which my assessment was informed by other disciplines were:

- Fault Studies inspectors have assessed the overall adequacy of the RP's Design Basis Analysis (DBA), including radiological consequence predictions. A number of judgements I made in my assessment were informed by sensitivity analyses conducted by the RP using the DBA as a starting point.
- Fault Studies inspectors have assessed the adequacy of the RP's fault identification, fault sequence development and design basis categorisation according to fault frequency. A number of judgements I made in my assessment were informed by the frequencies and categories assigned to certain faults.
- Fault Studies inspectors have led the assessment of Fault Studies computer codes including LOCUST and GINKGO. The RP has used these codes in conjunction with the Fuel and Core codes I have assessed and aspects of our assessment have therefore been collaborative.
- Fault Studies inspectors have assessed the RP's generic process for categorisation of safety functions and classification of SSCs. I have relied on their assessment when conducting my own assessment of the categorisation of functions for in-core instrumentation.
- Chemistry inspectors have assessed aspects of the fuel deposits safety justification that are associated with water chemistry phenomena and plant materials. Water chemistry, materials and core-related phenomena are heavily inter-related in this area so the assessment was undertaken collaboratively.
- Chemistry inspectors have assessed the adequacy of specific quantified coolant activity limits for the purpose of failed fuel identification during operation.
- Structural Integrity inspectors have assessed the adequacy of the structural integrity case for the Reactor Pressure Vessel (RPV) internals, including sensitivity to irradiation damage. An understanding of the sensitivity of the lower core plate to irradiation damage informed my assessment of part of the fuel system design.

- Probabilistic Safety Analysis (PSA) inspectors have assessed the Initiating Event Frequency (IEF) calculations provided to me by the RP for core mis-loading faults and have assessed the importance of different fault types to overall plant risk.
- Electrical Engineering inspectors have assessed the overall position of UK HPR1000 with respect to grid code compliance. We have worked collaboratively in this area and I have assessed the Fuel and Core aspects of some reports that were submitted to support the RP's claims about grid code compliance. Electrical Engineering inspectors have raised an Assessment Finding on the topic of grid code compliance and its resolution is likely to require further fuel and core analysis by the licensee.

3 REQUESTING PARTY'S SAFETY CASE

3.1 Introduction to the Generic UK HPR1000 Design

36. The generic UK HPR1000 design is described in detail in the PCSR (Ref. 3). It is a three-loop PWR designed by CGN using the Chinese Hualong technology. The generic UK HPR1000 design has evolved from reactors which have been constructed and operated in China since the late 1980s, including the M310 design used at Daya Bay and Ling'ao (Units 1 and 2), the CPR1000, the CPR1000^{+} and the more recent ACPR1000. The first two units of CGN's HPR1000, Fangchenggang Nuclear Power Plant (NPP) Units 3 and 4, are under construction in China and Unit 3 is the reference plant for the generic UK HPR1000 design. The generic UK HPR1000 design is claimed to have a lifetime of at least 60 years and has a nominal electric output of 1,180 MW.

37. The reactor core contains 177 zirconium alloy clad uranium dioxide (UO_2) fuel assemblies. Reactivity is controlled by a combination of control rods, soluble boron in the coolant and fixed burnable poisons within the fuel. The fuel assemblies in the design for GDA each contain 264 fuel rods with an active length of 12 feet (3.658 m), 24 control rod guide tubes and a single instrument tube in a 17x17 array; a common design for PWRs. The specific fuel design in the GDA design is the Framatome 'AFA 3GAA' model and the specific clad material used is 'M5', a proprietary alloy developed by Framatome. The fixed burnable poison used is distributed gadolinia (Gd_2O_3).

38. The control rods in the GDA design are in the form of Rod Cluster Control Assemblies (RCCA), each having 24 absorber tubes connected by a spider assembly at the top. These are a common design for PWRs. The specific RCCA design in the GDA design is the Framatome 'HARMONI' model. Some of the RCCAs are designated 'grey rods' of lower reactivity worth. These grey rod RCCAs have 8 tubes containing silver-indium-cadmium absorber material and 16 tubes containing stainless steel. The other RCCAs designated 'black' all have 24 tubes containing silver-indium-cadmium absorber material. There are 68 RCCAs in total, organised into several banks for control purposes:

- four shutdown banks ('S banks') dedicated to providing shutdown capability, which are always fully extracted during power operation;
- one temperature regulation bank ('R bank') to regulate temperature in response to small reactivity changes and to control axial offset, which is always partially inserted at the top of the core; and
- four power compensation banks ('G banks' and 'N banks'), the two G banks consisting of all the grey rod RCCAs and the N banks consisting of black RCCAs, all of which are fully extracted at full power, but increasingly inserted as power is reduced below baseload in order to compensate for the reactivity change caused by the power reduction.

39. Other non-fuel core components include neutron sources and in-core instrumentation. The neutron sources are used to provide a stable, detectable neutron flux signal during re-load and start-up. In-core instrumentation provides for RPV level measurement, core outlet temperature measurement and in-core neutron flux measurement. The in-core neutron flux detectors are arranged in 42 fuel assembly locations across the core and in seven axial segments to cover the full height of the fuel. They are Self-Powered Neutron Detectors (SPNDs) that use rhodium as the active material.

40. The core is contained within a steel RPV and held in place by a stainless steel core barrel, lower core support plate and upper core support plate, in a configuration recognisable from other PWRs worldwide. The outlet plenum above the core contains a number of control rod guide assemblies used to position the RCCAs accurately with respect to the fuel assembly guide tubes. The lower core support plate has an attached flow distribution device consisting of a perforated colander designed to

provide for more even flow at core inlet. The reflector region surrounding the core (inside the core barrel) is mostly water, with stainless steel baffle and former plates used to direct the desired amount of flow through the reflector region.

41. The RPV is connected to the Reactor Coolant System (RCS) components, including the Reactor Coolant Pumps (RCP), Steam Generators (SG), pressuriser and associated piping, in a three-loop configuration. The design also includes a number of auxiliary systems that allow normal operation of the plant, as well as active and passive safety systems to provide protection in the case of faults, all contained within a number of dedicated buildings.

42. The reactor building houses the reactor and primary circuit and is based on a double-walled containment with a large free volume. Three separate safeguard buildings surround the reactor building and house safety systems and the main control room. The fuel building is also adjacent to the reactor, and contains the fuel handling and short term storage facilities. Finally, the nuclear auxiliary building contains a number of systems that support operation of the reactor. In combination with the diesel, personnel access and equipment access buildings, these constitute the nuclear island for the generic UK HPR1000 design.

43. The formal record of the generic UK HPR1000 design at the end of GDA is captured in the Design Reference Report (Ref. 23).

3.2 The Generic UK HPR1000 Safety Case

44. In this subsection I provide an overview of the Fuel and Core aspects of the generic UK HPR1000 safety case as provided by the RP during GDA. Details of the technical content of the documentation and my assessment of it are reported in Section 4.

45. The UK HPR1000 GDA safety case has one overarching safety objective according to PCSR Chapter 1. This is that the generic design could be constructed, operated, and decommissioned in the UK on a site bounded by the generic site envelope in a way that is safe, secure and that protects people and the environment. The RP has defined a series of high level claims to provide structure to its safety case. The link between this safety objective and the reactor core in the PCSR is through Claim 3: “The design and intended construction and operation of the UK HPR1000 will protect the workers and the public by providing multiple levels of defence to fulfil the fundamental safety functions, reducing the nuclear safety risks to a level that is ALARP.”

46. The RP has indicated in PCSR Chapter 1 that the reactor core is addressed in sub-claim 3.3: “The design of the processes and systems has been substantiated and the safety aspects of operation and management have been substantiated.” In this context the substantiation is against the three fundamental safety functions (control of reactivity, removal of heat, confinement of radioactive material) and against the objective to reduce nuclear safety risks ALARP.

47. The detailed description of the RP’s definition of the reactor core is in PCSR Chapter 5 (Ref. 3). This document describes the design and the safety case claims on the SSCs that make up the reactor core, including the fuel. These claims are that:

- the Safety Functional Requirements (SFR) or design basis have been derived for the reactor core design;
- the reactor core design satisfies the SFRs or design basis;
- all reasonably practicable measures have been adopted to improve the design;
- the reactor core performance will be validated by commissioning and testing; and
- the effects of ageing of the reactor core have been addressed in the design and suitable EMIT are specified.

48. The RP has identified sub-claims, arguments and evidence for each of the SFRs, most of which are divided into groups. There are SFRs for the fuel system design, the nuclear design of the core and thermal hydraulic design of the core. In PCSR Chapter 5, the RP details the safety functions in these groups and references to further documentation to provide evidence that the generic UK HPR1000 design can achieve them.

49. The fuel system design section and supporting documents contain a systematic evaluation of phenomena that can cause a loss of radiological material confinement through fuel cladding failure. This evaluation is intended to show that fuel failures due to these phenomena are precluded in Design Basis Condition (DBC)-1 operations, DBC-2 faults and more frequent DBC-3 faults. In practice, this means any fault with an IEF greater than 10^{-3} per year. The demonstration that fuel failures are reduced ALARP in less frequent DBCs is referenced out to PCSR Chapter 12 (Fault Studies).

50. Separately the fuel system design includes an evaluation of conditions that could prevent control of reactivity. These are conditions that impact the ability of control rods to insert. This evaluation shows that none of the analysed faults will prevent control rod insertion.

51. The nuclear design section and supporting documents contain an evaluation of nuclear parameters including burnup, reactivity control, power distributions, stability and Shutdown Margin (SDM) in various conditions to show that design basis limits are met. This is intended to support claims that there are adequate margins between operating parameters and reactor protection setpoints in DBC-1, and that protective actions will be triggered that prevent fuel failures and allow operators to restore DBC-1 operations when a DBC-2 or frequent DBC-3 fault occurs.

52. The thermal hydraulic design is a detailed evaluation of the core cooling behaviour during DBC-1 and DBC-2 conditions. This is intended to demonstrate that in all these conditions no fuel rods will experience Departure from Nucleate Boiling (DNB), no fuel rods will reach the fuel melting temperature, core flow meets the minimum requirement and hydrodynamic instability will not occur. The evaluation shows that these objectives are met, with allowances for uncertainties. Examination of core cooling in less frequent DBCs is referenced out to PCSR Chapter 12.

53. PCSR Chapter 5 has a section to justify that the reactor core design has reduced risks ALARP. The RP has described the evolution of the generic UK HPR1000 design from its predecessors and carried out a review of RGP and relevant OpEx. Following risk assessment using DBA, the RP has made a modification to a trip setting within the Reactor Protection System (RPS) to avoid PCI fuel failures in frequent faults. It argues that further modifications, to reduce the number of fuel rods experiencing DNB in infrequent faults, are not reasonably practicable. On this basis, the RP concludes that the fuel and core design has reduced risk ALARP.

54. The RP has included two sections to address future EMIT and commissioning activities. These explain that the generic design will allow the licensee to carry out appropriate commissioning testing to validate the analysis methods and assumptions, and give an indication of the types of EMIT activities that can be implemented to mitigate the impacts of ageing and in-service degradation. This includes a discussion on the RP's strategy for failed fuel management. The PCSR presents these as an overview because they will be developed further by the site licensee.

55. PCSR Chapter 5 also includes an appendix and reference to supporting documents that describe the computer codes used and provide Verification and Validation (V&V) evidence.

56. The fuel and core design supports some of the fault analysis reported in PCSR Chapter 12. This is to support Claim 3.2 “A comprehensive fault and hazard analysis has been used to specify the requirements on the safety measures and inform emergency arrangements” and Claim 3.4 “The safety assessment shows that the nuclear safety risks are ALARP”. Against these claims the fuel and core analysis provides both neutronic data and acceptance criteria that, if met, ensure the delivery of the three fundamental safety functions in faults.

57. The RP’s documentation underpinning PCSR Chapter 5 is arranged in a tiered structure. PCSR Chapter 5 itself is a Tier 1 document. The key documents describing the design, design bases and substantiation in each area described above are Tier 2 documents. The further documents required to provide detailed supporting evidence are Tier 3 documents. Tier 2 documents are mostly references to the PCSR and Tier 3 documents are mostly references to Tier 2 documents. The document hierarchy is described within the RP’s Production Strategy for Fuel and Core Design (Ref. 24). Ref. 24 presents useful diagrams showing the hierarchy for fuel system design, nuclear design and thermal hydraulic design, which I have included in Annex 3 to this report.

3.3 Requesting Party Organisation and Information-Sharing

58. As described in Section 1, GNSL is a UK-registered company that was established to implement the GDA on the UK HPR1000 design on behalf of CGN, EDF SA and GNI.

59. As noted in ONR’s summary report for Step 3 of GDA (Ref. 25), the UK HPR1000 core design changed from using CGN’s STEP-12 fuel design to the AFA 3GAA fuel designed by Framatome. The technical and safety aspects of this change were considered as part of my assessment, but this change also brought additional complexities to the RP’s organisation that were specific to the Fuel and Core topic, and therefore are noted here. Specifically, CGN sub-contracted Framatome to provide fuel system design information and a range of supporting safety case evidence in GDA.

60. All the safety case documents submitted in the Fuel and Core topic in GDA were authored by either CGN or Framatome. Most of these were submitted via GNSL. However, for commercial reasons, a small number of documents were unable to be shared amongst all the RP organisations, so were submitted by CGN or Framatome directly to ONR. The documents affected mostly related to computer code V&V evidence and a small subset of the evidence underlying the fuel design and fuel performance analysis.

61. Importantly during GDA:

- the RP was able to put arrangements in place to share relevant information to enable development of their safety case submissions; and
- my own access to information has been unaffected by this matter so I have been able to undertake a meaningful assessment of the safety case.

62. However, such constraints on information sharing, if they were carried forward to the licensee, could present a barrier to these documents being used to fully understand the risks associated with its activities or to further develop the site-specific safety case. SAP MS.2 (Capable Organisation) and the associated guidance in Ref. 2 describe a series of expectations associated with a dutyholder’s organisation, access to information and ‘intelligent customer’ role. It will be important that the licensee implements an organisational structure and information-sharing mechanisms that allow it to develop and maintain an adequate intelligent customer capability. Such matters would be an important consideration in any future licensing activities undertaken by ONR, but are outside the scope of my assessment during GDA.

4 ONR ASSESSMENT

4.1 Structure of Assessment Undertaken

63. The structure of this section of my report is aligned with that of my assessment scope and sampling strategy described in Section 2.

64. The main three themes considered in my assessment were the nuclear design of the core, the thermal hydraulic design of the core and the thermo-mechanical design of the fuel system. Within each of these three main themes, I considered the adequacy of the design bases used, the safety margins presented, the limits or performance data provided for use in fault analysis and the computer codes, modelling methods and assumptions used. For ease of readability I have separated out subsections of this report on neutronic data from that on the nuclear design and on fuel design criteria and safety limits from that on fuel system design. I have also separated out all my assessments of the validity of computer codes and modelling methods, which are placed together towards the end of the report.

65. In addition, I have sampled several safety-significant areas in which interactions between neutronic, thermal hydraulic and/or fuel thermo-mechanical phenomena are particularly important. My assessment of PCI safety analyses, fuel modelling in LOCA and the impact of fuel deposits are each contained in their own dedicated subsections of this report.

66. Further, I have considered safety-significant parts of the fuel and core safety case associated with how the UK HPR1000 plant will be operated. All of these topics have links back to the three main themes discussed above but are assessed in separate subsections of this report. I have combined my assessments of the reduction of the risks posed by core mis-loads and of the requirements for monitoring using in-core instrumentation in a single subsection because detection of a core mis-load is reliant upon the adequacy of the instrumentation specified within the design. Separate subsections are used to record my assessment of the management of failed fuel in operation, and of operating limits and conditions, commissioning and EMIT.

67. My report concludes with my assessment of the overall demonstration that risks have been reduced ALARP and of whether all of the safety case information presented to me in GDA has been adequately consolidated in to the PCSR and supporting documents by the end of the process. I have also provided a summary of the standards, guidance and RGP that I have used at the end. The resulting structure of this sub-section is as follows:

- Core nuclear design
- Neutronic and kinetics data for use in fault analysis
- Fuel system design
- Fuel design criteria
- Thermal hydraulic design and criteria
- Protection against PCI
- Fuel behaviour in a Large Break LOCA (LB-LOCA)
- Fuel deposits
- Core mis-loading and in-core neutron detectors
- Management of failed fuel in operation
- Operating limits and conditions, commissioning and EMIT
- Computer code validity
- Demonstration that relevant risks have been reduced ALARP
- Consolidated safety case
- Comparison with standards, guidance and RGP

4.2 Core Nuclear Design

4.2.1 Assessment

68. The nuclear design of a reactor core includes the definition of the fuel assembly arrangement in the core, the selection of fuel assembly types for every location in the core, and the selection of fuel enrichments and burnable poison loadings in each fuel rod. It includes definition of core re-load designs intended to make appropriate use of burnt fuel assemblies and eventually reach a stable 'equilibrium cycle'. It includes the selection of locations in the core for RCCAs (and their groupings), for in-core instrumentation, for neutron sources and the selection of any materials with the potential to affect core neutronics. It also includes definition of a number of operating limits and conditions including RCCA bank insertion limits, boron concentration limits and the intended operating cycle length.

69. The nuclear design is fundamental to achieving the desired power output, cycle length and fuel burnup for a reactor design. In turn, these parameters are fundamental to achieving the desired economic performance. From a safety perspective, the nuclear design has a direct impact on all three fundamental safety functions referred to by SAP ERC.1 because it determines global and local reactivity behaviour and heat production in the core. It effectively sets a wide variety of nuclear performance parameters, which have a significant influence on the plant response to abnormal conditions, the worth of protection systems and the safety margin available in faults. As a result, an inadequate or poorly-modelled nuclear design will likely lead to a loss of fault tolerance.

70. The nuclear design of a reactor core with all the above in mind is a complex multi-objective optimisation task with many variables. NS-TAST-GD-075 (Ref. 11) states "The inspector should be satisfied that limits have been placed on key core design parameters that influence the plant response to abnormal conditions and the worth of protection systems. These limits should be consistent with the data provided to fault analysis. Such limits place active constraints on the core design."

71. It is not possible to assess each element of the nuclear design (for example, a particular fuel enrichment decision) in isolation because the reactor's performance is a result of interaction between all of these elements together. My assessment of the nuclear design therefore has several strands, to allow me to a reach judgement on whether the complete nuclear design meets regulatory expectations:

- an assessment of the design objectives and limits that the RP has set on the nuclear design that are important to safety, and the evidence provided that the design complies with them (reported in this subsection);
- an assessment of whether the design has been developed or else evolved from its predecessors in such a way as to improve safety (reported in this subsection and in subsection 4.14);
- an assessment of how feedback from design basis fault analysis has been used to further improve the design and reduce risks ALARP (reported in subsection 4.14); and
- an assessment of the adequacy of nuclear data provided for use in fault analyses (reported in subsection 4.3).

72. I have discussed the second and third items above in subsection 4.14 as well as this subsection because of their importance to the RP's demonstration that the fuel and core design reduces risks ALARP.

73. I have applied SAPs EKP.2, ERC.1, ERC.2, ERC.3, ERC.4, EAD.1, FA.7 and AV.3 in reaching a judgement about the adequacy of the nuclear design for UK HPR1000. I have also used NS-TAST-GD-075 (Ref. 11) and SSG-52 (Ref. 19) to benchmark my

expectations for aspects of the nuclear design that should be evaluated and/or limited by the design process.

74. The RP has submitted the nuclear design basis for the UK HPR1000 core design (Ref. 26). This presents the limits set on the nuclear design to ensure that adequate safety margin exists to either criteria at which one of the fundamental safety functions could be compromised, or to assumptions made in fault analyses. Design information and nuclear data has been provided in full for the UK HPR1000 first cycle core design (Ref. 27) and for the predicted equilibrium cycle core design (Ref. 28). Complete core designs and a sub-set of the nuclear data have also been provided for the transition cycles (cycles 2 and 3) within the fuel management report (Ref. 29). This is sufficient for me to make judgements about the adequacy of the UK HPR1000 nuclear design and its evaluation against the nuclear design basis.

75. The demonstration that risks have been reduced ALARP for the UK HPR1000 reactor core design is presented by the RP in Ref. 30, which I have assessed against guidance from NS-TAST-GD-005 (Ref. 12) in subsection 4.14 of this report. Ref. 30 states that the UK HPR1000 core design is an evolution of that used in the Chinese CPR1000 operating plants. The changes from the CPR1000 core design primarily constitute the addition of 20 fuel assemblies at the edge of the core (which slightly reduces the height/diameter ratio, reducing neutron leakage) and a slight reduction in the core average power density. The RP argues that the reduced power density will result in a higher thermal margin and hence provide a safety benefit, whilst the reduced neutron leakage will result in slightly higher average discharge fuel burnup and therefore fewer spent fuel assemblies per unit of energy production.

76. I am satisfied with the arguments presented that the nuclear design of the core has evolved from the design used in the CPR1000 fleet in China in ways that should improve safety. I have not observed any potential safety detriments to the changes. However, safety improvements can only be demonstrated through the detailed analysis that is assessed later in this report and in the Fault Studies Assessment Report (Ref. 7). Other changes from the CPR1000 core design, such as the number of RCCAs and locations of ex-core detectors, simply follow from the increased number of fuel assemblies. The only aspect of the core design that constitutes a significant change from CPR1000 is the use of in-core SPNDs. The detailed design of these detectors will be finalised post GDA, but Ref. 30 provides an adequate explanation for the selected number of SPNDs and their locations. I judge that these detectors should provide for a practical safety improvement in UK HPR1000 over the CPR1000 fleet by providing the operators with earlier visibility of any unexpected distortions in the core power distribution. I have more fully assessed their functionality and safety classification in subsection 4.10 of this report.

77. Ref. 30 also briefly compares the 18-month equilibrium cycle that has been selected in GDA with 12-month and 24-month options. The data presented includes a comparison of achievable average fuel discharge burnup, cycle length and key nuclear safety parameters for the three designs. The 18-month cycle used in the GDA design is stated to be the best choice because the analysis shows that the nuclear safety parameter limits are met, because it provides for better average discharge burnup than the other designs and because it is aligned with OpEx at the CPR1000 plants from which the generic UK HPR1000 design has evolved. I am therefore satisfied that the decision to use an 18-month equilibrium cycle is supported by the available evidence and my detailed assessment in this report is all based upon the 18-month equilibrium cycle that forms part of the design reference for GDA. However, I note that I have not seen evidence that the 12-month or 24-month options considered in Ref. 30 were optimised and I recognise that the licensee may wish to design a different equilibrium fuel cycle. Were this to be the case, it would be subject to appropriate ONR attention as part of routine regulatory interventions associated with licensing and permissioning.

78. The nuclear design reports for the first and equilibrium cycle core designs (Ref. 27 and Ref. 28) present core loading patterns, locations of RCCAs, primary and secondary source assemblies and in-core instruments. The chosen locations provide for symmetrical reactivity insertion with each RCCA bank, reasonable core coverage with shutdown bank RCCAs and reasonable core coverage with each group of in-core SPNDs. In my opinion, this is a necessary enabler for an adequate shutdown system and adequate core monitoring capability, as expected by SAPs ERC.2 and ERC.4. However, these capabilities should also be verified by suitable quantitative analysis. I have discussed the quantitative analysis for the shutdown system in sub-section 4.2.1.4 and for the in-core detectors in subsection 4.10 of this report.

79. I have observed that the fuel enrichments and poison loadings used by the UK HPR1000 nuclear designs are within ranges used in other PWRs.

80. In my opinion the nuclear design basis (Ref. 26) shows that the RP has taken a systematic approach to the core nuclear design and that most of the nuclear performance parameters I would expect to see, using guidance from Ref. 11 and Ref. 19, are captured. Where exceptions exist, I have discussed them in this report.

81. I have reviewed all UK HPR1000 submissions associated with the nuclear design of the core to satisfy myself that adequate limits are placed on nuclear parameters and that the results of the design evaluation show compliance with the stated limits. On some specific topics that are either particularly important to reduction of risk or where additional information was required to form a judgment on the argument being made in the safety case, I sampled the underlying evidence more deeply. These areas are as follows:

- Burnup Limits
- Power Distributions
- Moderator Temperature Coefficient (MTC)
- Control rods and SDM
- Tolerance to core mis-loading faults

82. The last of these topics is discussed in subsection 4.10 due to its close relationship with the requirements on in-core instrumentation. The others are covered in the remainder of this subsection.

4.2.1.1 Burnup Limits

83. The safe working life of nuclear fuel rods and fuel assemblies is usually defined in terms of ‘burnup’, which is a measure of the time-integrated irradiation per unit mass that the fuel has undergone in the reactor core. Fuel behaviour in normal operations and faults varies as a function of burnup, so it is important not only to define a maximum burnup beyond which fuel must not be irradiated further, but also to demonstrate that the validation limits of computer codes and methods applied to fuel in the safety case bound this limit. The burnup reached by the fuel is a function of the nuclear design of the core. Informed by SAP EAD.1, my assessment expectations are firstly that burnup limits be derived to which the fuel can operate safely and for which fuel behaviour can be predicted using validated methods, and secondly that the predicted peak fuel burnup at End of Cycle (EOC) in the equilibrium cycle design does not exceed these limits.

84. Ref. 26 presents design limits for maximum fuel burnup of 57,000 MWd/Te (fuel rod average) and 52,000 MWd/Te (fuel assembly average). These values are relatively modest compared to other PWRs that have passed through GDA in the UK and are well within the range of burnups experienced internationally by Framatome AFA 3GAA fuel assemblies with M5 clad, presented in the fuel assembly OpEx report (Ref. 31). I judge the level of fuel reliability in normal operation shown for this fuel type and burnup

range by Ref. 31 to be adequate, as discussed latterly in subsection 4.4. I am also satisfied that the fuel criteria used in fault analyses are valid up to the specified burnup limits, implying that if fault analysis concludes that the criteria are met, then fuel within these burnup limits should behave in the expected manner. My assessment of the fuel criteria used in fault analyses is reported in subsection 4.5.

85. The specified burnup limits above are well within the validation envelope of the fuel performance code COPERNIC (Ref. 32) for UO_2 fuel rods on a rod-average basis and I have also satisfied myself that the nuclear design codes PINE and COCO will provide adequate modelling of the core up to this burn up range (see subsection 4.13 for my assessment of these codes). However, I observed during my assessment that the limit of the COPERNIC validation range for UO_2-Gd_2O_3 (that is, gadolinia-poisoned) fuel rods is slightly lower than the fuel rod average burnup limit specified for the nuclear design in Ref. 26. I therefore looked for assurance that the UO_2-Gd_2O_3 fuel rods in the UK HPR1000 core are not in practice predicted to reach burnups that exceed the COPERNIC validation range for that fuel type.

86. The nuclear design report for the equilibrium cycle (Ref. 28) and fuel management report (Ref. 29) present burnup maps of the whole core to demonstrate that the core design meets the burnup limits specified above with some additional margin at the end of the equilibrium cycle. However, Ref. 28 and Ref. 29 do not specifically present the peak burnup reached by UO_2-Gd_2O_3 fuel rods. I therefore requested evidence through RQ-UKHPR1000-0574 (Ref. 33) that the lower COPERNIC validation limit for UO_2-Gd_2O_3 rods still bounds the peak burnup values reached in the UK HPR1000 core for this fuel type. The RP's response to the RQ showed this to be true because the peak burnup in UO_2-Gd_2O_3 fuel rods is significantly lower than that in UO_2 fuel rods, due to the lower fuel enrichment used in the UO_2-Gd_2O_3 rods. I was therefore satisfied that the peak fuel burnup reached by all fuel rods in the presented core designs is well bounded both by OpEx and by relevant core computer code validation limits, for both UO_2 and UO_2-Gd_2O_3 fuel rods. I consider it a minor shortfall in the safety case that a separate, lower fuel rod burnup limit is not specified in the nuclear design basis for UO_2-Gd_2O_3 rods that limits their burnup to within the validation range of the COPERNIC computer code.

87. Overall, I judge that the burnup limits presented in the UK HPR1000 safety case are adequate and I am satisfied that the presented core designs meet the limits with reasonable margin.

4.2.1.2 Power Distributions

88. The local power produced in a reactor core varies considerably because of the different rates of fission, neutron moderation, neutron absorption and neutron leakage in different parts of the core. The safety margin available to fuel design criteria in faults is generally smallest in the locations of highest power. The ratio between peak local power and average core power is usually quantified in terms of 'power peaking factors'. The magnitude of these power peaking factors directly drives the local power in the hottest parts of the core. They have a significant impact on the ability to remove sufficient heat from the core and on the fault tolerance of the plant.

89. Informed by SAPs ERC.1, EKP.2, FA.7 and NS-TAST-GD-075 (Ref. 11), my expectations are therefore that reasonably practicable steps be taken in core design to limit power peaking factors and that, most importantly, they are evaluated for each core design and core operating state for which fault analysis is required in the safety case, to show that fault analysis assumptions are valid.

90. Limits are set by the RP in the nuclear design basis (Ref. 26) and fuel management report (Ref. 29) on power peaking factors in the UK HPR1000 nuclear design. In normal operational states these include the total 3-D heat flux hot channel factor

(termed F_Q) and enthalpy rise hot channel factor (termed F_{dH}), which is a measure of 2-D radial power peaking. As discussed further in subsection 4.3, I have satisfied myself that these limits are consistent with assumptions made in relevant fault analyses.

91. The RP's safety case claims associated with clad corrosion and fuel deposits require that F_{dH} be limited to keep hot channel void fraction below 5% in normal operation (Ref. 34), irrespective of the safety margin available in faults. Evidence was not initially provided that the stated limit on F_{dH} in Ref. 29 would keep hot channel void fraction below 5% in normal operation for UK HPR1000. However, the RP submitted this evidence in response to RQ-UKHPR1000-0835 (Ref. 33) and subsequently in an analysis of thermal and boiling parameters for the UK HPR1000 core (Ref. 35).

92. The first cycle and equilibrium cycle core designs developed against the RP's power distribution limits are presented in Ref. 27 and Ref. 28. No particular core design measures (such as use of axial fuel enrichment zoning or 'axial blankets') have been taken to reduce axial power peaking in the UK HPR1000 core. However, the design limit on F_Q and the peak local heat flux are both lower than equivalent values for AP1000 at the time of GDA or for Hinkley Point C at PCSR3 stage, so I am satisfied such measures are not necessary. Having assessed the nuclear design methods used in subsection 4.13 of this report, I judge that evidence provided by the RP in Ref. 27, Ref. 28 and the neutronic data for LOCA analysis report (Ref. 36) is adequate to demonstrate that the stated limit on F_Q is met by the first and equilibrium cycle core designs in all relevant core operating states.

93. I observed that the first cycle core loading pattern effectively flattens the radial power distribution across the core, with more reactive fuel placed at the core edge. Ref. 27 shows that there is large margin to the specified F_{dH} limit for the first cycle, providing for additional fault tolerance in this cycle. However, the equilibrium cycle loading pattern presented in Ref. 28 uses a lower-leakage loading pattern with less reactive assemblies at the core edge, which leads to more radial power peaking and higher F_{dH}. The equilibrium cycle core design also uses relatively high concentrations of gadolinia poison in some fuel rods (albeit within the ranges used successfully in other PWR designs), which will increase peak F_{dH} due to the different reactivity of these fuel rods compared to those around them.

94. It is important that F_{dH} adheres to the stated design limit in all core operating states so I have sampled the safety case in particular depth on this topic. F_{dH} varies strongly between cycles and through a cycle, and as a function of RCCA bank insertion. In the fuel management report (Ref. 29), F_{dH} is evaluated throughout using an 'all-rods-out' condition. In both Ref. 27 and Ref. 28, it is also evaluated in an overly-pessimistic configuration with the R bank of RCCAs fully inserted (well beyond the bank insertion limits). In the latter case, the limit on F_{dH} is shown to be breached and I judged that neither evaluation was adequate to provide evidence that the F_{dH} limit will be met throughout the permitted RCCA bank insertion range. In practice the R bank of RCCAs will be inserted a small way during operation at full power, the permitted range (between 'bite point' and insertion limit) being defined in Ref. 27 and Ref. 28 for each cycle design. I requested further evidence on this topic and in response to RQ-UKHPR1000-0684 (Ref. 33), the RP presented additional radial power distribution maps calculated with the R bank inserted to its insertion limit. This data was sufficient to satisfy me that the F_{dH} limit for UK HPR1000 is met throughout the allowable RCCA bank insertion range at full power for these core designs, at the most limiting times in cycle. However, I note that when uncertainties are considered in the calculation the limit is only just met, with no significant additional margin.

95. A comprehensive evaluation against the F_{dH} limit for the period before xenon equilibrium is reached at Beginning of Cycle (BOC) is not presented in Ref. 27 or Ref. 28. A summary of data presented in Ref. 29 suggests that the limit may be breached in

the xenon-free BOC core operating state for the second cycle, but that positive margin should be retained for the other cycles. The second cycle is not in the scope of my assessment (see subsection 2.3) and I anticipate that an evaluation of F_{dH} margin will be a fundamental part of the licensee's second cycle design work. However, I sought to understand the RP's reasons for the omission of detailed data for first and equilibrium cycles. The RP argued that the theoretical BOC, xenon-free, full power condition cannot occur in practice, and therefore argued that an explicit F_{dH} evaluation against the usual limit is not required in this condition. It also presented low-power radial power distribution data in response to RQ-UKHPR1000-0684 (Ref. 33) to show that the simplified equation it uses to calculate a bounding F_{dH} for the purposes of fault analyses at low power (for example, during power raise) is conservative. I consider that these arguments have some merit but still judge the lack of detailed F_{dH} data for the first and equilibrium cycle core designs in this period of operation to be a minor shortfall in the safety case.

96. The F_{dH} data provided in Ref. 27, Ref. 28 and Ref. 29 is shown without uncertainty added, implying greater safety margin than actually exists. In response to RQ-UKHPR1000-0574 (Ref. 33), the RP provided details of the uncertainties applicable to F_{dH} predictions. The main sources of uncertainty declared are that due to the nuclear design calculation itself, that due to uncertainties in xenon distribution and that due to potential RCCA misalignment. Engineering uncertainties on F_{dH} are also accounted for separately by the RP in thermal hydraulic calculations. The data in the RQ response showed that the F_{dH} limit would still be met with all uncertainties accounted for. I have gained confidence in the adequacy of the uncertainty allowances by verifying that the calculational nuclear uncertainty factor was consistent with that derived from code validation work (see subsection 4.13) and by comparing the uncertainties with those applied for PWRs in previous GDA. I am now satisfied that power distribution uncertainties are adequately catered for by the generic UK HPR1000 safety case, in accordance with my expectations derived from SAP AV.3.

97. Taking all of the above evidence together, I judged the RP's evaluations of F_{dH} to be adequate for the purposes of GDA. I consider the lack of a comprehensive survey of F_{dH} for the short period of operation during which xenon builds up to its equilibrium level to be a minor shortfall in the safety case.

98. I have also looked for evidence that feedback from DBA has been used to identify potential improvements to the core design. In this context, there are two design basis intact circuit faults in the UK HPR1000 safety case that are predicted to cause some fuel rods to undergo DNB and lose clad integrity. In one of these, a locked rotor fault, safety margins could theoretically be improved by reducing the normal operation core F_{dH} limit. In an ALARP assessment report for DNB analysis (Ref. 37), the RP has provided arguments and supporting evidence to show that it is not reasonably practicable to reduce F_{dH} through a core re-design to reduce the consequences of a locked rotor fault (which are reported in the relevant transient analysis report, Ref. 38).

99. I have assessed the demonstration of ALARP holistically in subsection 4.14. From the perspective of reducing F_{dH} the RP shows that it is possible to achieve a significant improvement by changing to an "out-in" loading pattern for the equilibrium cycle with the most reactive fuel at the edge of the core, more similar to the first cycle design. However, the RP argues that the change is not reasonably practicable because of the impact predicted on achievable cycle length and discharge burnup (which significantly impact the plant economics) as well as a predicted reduction in SDM. I am satisfied with the evidence provided in the PCSR Chapter 5 ALARP demonstration (Ref. 30) and Ref. 37 to support the RP's arguments. I also note that using an out-in loading pattern for the equilibrium cycle would increase neutron leakage from the core and consequent irradiation damage of the RPV.

100. Overall, I am satisfied that adequate design limits have been set on power peaking factors, that adequate evaluations of the core design have been completed to ensure these limits are met, and that it is not reasonably practicable to reduce them further through a core re-design. Some small benefit may be practicable through modification to RCCA bank insertion limits, which I have discussed in 4.2.1.4 and subsection 4.14.

4.2.1.3 Moderator Temperature Coefficient

101. The neutronic stability of a nuclear reactor core and its response to fault conditions is dependent on the way in which core reactivity changes in response to a change in conditions, particularly core power and temperature. This response is quantified in terms of reactivity coefficients. NS-TAST-GD-075 (Ref. 11) advises 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.” The most important reactivity coefficients for the response of a PWR in fault conditions are the Doppler coefficient (measuring response to changes in fuel temperature or power) and the MTC (measuring response to changes in moderator temperature and therefore density). Clear limits are stated by the RP in Ref. 26 for the Doppler coefficient to be negative and the MTC to be non-positive during all powered operation. In my opinion, informed by Ref. 11 as well as SAPs EKP.2 and ERC.3, this is good practice.

102. I observed some potential shortfalls in the evaluation of MTC against the limit (contained in Ref. 27, Ref. 28 and Ref. 29), so I decided to sample this part of the safety case in further depth.

103. Like F_{dH} data, the MTC predictions provided in Ref. 27, Ref. 28 and Ref. 29 are shown without uncertainty added, implying greater safety margin than actually exists. In response to RQ-UKHPR1000-0574 (Ref. 33), the RP quantified the uncertainty applicable to MTC predictions. It provided evidence the allowance was sufficient and I have also gained additional confidence by comparing the uncertainty with that applied for PWRs in previous GDA. I am now satisfied that this uncertainty is adequately catered for in accordance with my expectations derived from SAP AV.3.

104. Ref. 27 and Ref. 29 show that the MTC is slightly positive in the first cycle core design at BOC when at hot zero power. By implication, the MTC would be slightly positive early in the power raise process and therefore be non-compliant with the stated limit.

105. The RP’s position as explained in Ref. 29 is that the MTC calculation has been completed with all RCCAs fully withdrawn and in practice the MTC will be maintained non-positive by RCCA insertion during the power-raise process. RCCA insertion will reduce the critical boron concentration at a given power level, which has the effect of making MTC more negative. Ref. 29 also states that “neutronics calculations show that the moderator temperature coefficient is negative if the power compensation banks are inserted to their calibration position, this induces that the complete compensation banks withdrawal may be forbidden at beginning of cycle”. I judged that if this practical approach to control of MTC were to be taken then it was important that the calculations of required RCCA bank insertion were presented and the required operating rules were clearly identified.

106. In response to RQ-UKHPR1000-0574 (Ref. 33), the RP provided quantified predictions of the minimum RCCA insertions that would be required to maintain MTC non-positive when at hot-zero power. Furthermore, in response to RQ-UKHPR1000-0684 (Ref. 33) the RP provided a description of the proposed start-up and power raise sequence for UK HPR1000. This includes a set of zero-power physics tests undertaken when criticality is reached but before going to power, which include an explicit measurement of MTC. The RQ response describes operating rules to ensure that the plant cannot progress beyond the zero power state with a positive MTC. I am satisfied that the risk

of a positive MTC occurring at power at the beginning of the first cycle would be adequately controlled by these operating rules. They are briefly summarised in the RP's 'synthesis report' on positive MTC (Ref. 39) but not detailed. The RP has logged a commitment in their own post-GDA commitment list (Ref. 40) that RQ responses relating to physics tests and commissioning will be consolidated to the safety case post GDA, which I understand should ensure that the detail is not lost. However, in my opinion, a clear requirement for operating rules of this nature should be recorded in the safety case. This is not the only shortfall I have identified of this type, so in subsection 4.12 I have listed all those I have identified and recommended a means by which they can be addressed.

107. I observe from data in the references provided that the total power coefficient (accounting for both the MTC and the Doppler coefficient) will remain negative in core conditions even with all RCCAs fully withdrawn (and therefore higher boron concentration). This reduces the risk associated with a positive MTC in some faults.

108. Overall, I judge that the coefficients of reactivity for the UK HPR1000 core are such that control of reactivity can be maintained and the core should be stable in powered operation. I consider it a minor shortfall that the MTC does not meet the RP's own design basis limit in some possible plant conditions at the beginning of the first cycle, because operating rules should be in place to prevent this from happening in practice. However, further work is necessary to ensure the requirement for these operating rules is sufficiently clear in the safety case, as discussed in subsection 4.12.

109. In some operating conditions when the plant is shutdown with high boron concentrations, the MTC will inevitably be positive and no specific design rule is applied. In my opinion, informed by SAP FA.7, the most important consideration for shutdown conditions is that bounding neutronic and kinetic data be supplied for the purpose of fault analysis, in order to allow for a conservative analysis of fault consequences. I have assessed this part of the UK HPR1000 safety case in subsection 4.3.

4.2.1.4 Control Rods and Shutdown Margin

110. The UK HPR1000 reactor core can be shut down either by inserting banks of RCCAs manufactured with a silver-indium-cadmium neutron absorber, or by increasing the concentration of neutron-absorbing boric acid in the core coolant. Both of these variables are also used to control core reactivity in normal operations.

111. The generic UK HPR1000 design meets the expectation set by SAP ERC.2 that two diverse systems be provided for shutting down the reactor. As part of the wider plant design, the systems used to inject boric acid and control its concentration in the RCS have been assessed by Fault Studies and Chemistry inspectors. I have assessed the design of the RCCAs within the mechanical shutdown system. I have also assessed the calculations of SDM presented by the RP to demonstrate the mechanical shutdown system has adequate reactivity worth. Informed by SAP ERC.2, SAP FA.7, Ref. 11 and Ref. 19, my main expectations in this assessment were as follows:

- the range of allowable RCCA insertions should be defined in the safety case for all RCCA banks;
- definition of the range of allowable RCCA insertions should consider the impact on SDM and on the consequences of reactivity insertion faults as part of a demonstration that risks are reduced ALARP, because deeper RCCA insertions can increase the consequences from such faults;
- a demonstration should be provided that suitable and sufficient SDM exists in the case that a fault occurs when operating at the limits of permitted operation (for example, with RCCAs inserted to their insertion limits); and

- SDM calculations should account for relevant uncertainties in a conservative manner and assume that one of the RCCAs fails to insert.

RCCA Insertion Limits

112. The UK HPR1000 nuclear design contains a total of 68 RCCAs, 12 of which are grey rod RCCAs, split in to S, R, G and N banks for control purposes as described in subsection 3.1.

113. The nuclear design reports for the first and equilibrium cycles (Ref. 27 and Ref. 28) present the exact location in the core of each RCCA. They also present minimum operational insertion (bite point) curves for the R bank, maximum operational insertion (insertion limit) curves for the R bank, and calibration curves (defining the exact required insertion as a function of reactor power) for each G and N bank. I am therefore satisfied that the range of allowable insertions are clearly defined for all RCCA banks in power operation.

114. Ref. 27 and Ref. 28 state that the maximum insertion of the R bank has been restricted to meet requirements on RCCA ejection fault safety criteria, F_{dH} and SDM. This qualitative statement indicates that the correct considerations have been made in setting the limit, but no underlying evidence is provided. I have observed that along with a locked rotor fault (see paragraph 98), a design basis RCCA fault is predicted to cause some fuel rods to undergo DNB and lose clad integrity. I have therefore sought evidence from the RP that it is not reasonably practicable to reduce the consequences of this fault through improvements to the RCCA bank designs or insertion limits. The RP's arguments and evidence on this point are contained in Ref. 37, which I assess in subsection 4.14 as part of the overall ALARP demonstration for the reactor core.

Shutdown Margin

115. Ref. 26, Ref. 27 and Ref. 28 present minimum SDM requirements from BOC to EOC. Ref. 26 explains that the minimum SDM requirements are set by the main steam line break fault, implicitly claiming this to be the limiting fault, such that if SDM is sufficient for this fault then it will be sufficient for all design basis faults.

116. Ref. 26 explains that adequate SDM is required to overcome the positive reactivity added in a main steam line break fault and prevent re-criticality from occurring after the reactor has shut down, at any time in the operating cycle. However, I reviewed the main steam line break transient analysis report (Ref. 43) and observed that re-criticality does in fact occur when the minimum SDM is assumed within the fault analysis. As such, I consider that the basis stated in Ref. 26 for the minimum SDM requirement is inaccurate. In response to RQ-UKHPR1000-0574 (Ref. 33), the RP clarified that it considers the SDM to be sufficient as long as the fault acceptance criteria are met in this fault, rather than requiring explicitly that re-criticality does not occur. I judged the RP's arguments (and therefore the stated SDM requirement) to be adequate because they are consistent with the approach taken for other PWRs in the UK. I consider the inaccurate basis stated for the SDM requirement in Ref. 26 to be a minor shortfall in the safety case.

117. Nuclear design calculations are presented in the Nuclear Design Reports for the First Cycle and Equilibrium Cycle (Ref. 27 and Ref. 28) to show that the presented core designs and RCCA insertion limits meet the minimum SDM requirements in Ref. 26, throughout the operating cycle. Ref. 27 and Ref. 28 also present details about the assumptions made and uncertainties captured in the analysis. In my opinion, accounting for guidance in Ref. 11 and Ref. 19, the approach taken to uncertainties is comprehensive. The RCCA with highest reactivity worth in each case is assumed to be stuck out of the core and the specified insertion limits are adequately accounted for in the analysis. Other appropriate uncertainty allowances are also included. Furthermore,

the reports show that the predicted SDM is significantly in excess of the minimum stated requirement throughout the first and equilibrium cycles. I consider this to be a strength of the UK HPR1000 first and equilibrium core designs.

118. From reviewing detailed geometry data within the fuel assembly and RCCA design descriptions (Ref. 44 and Ref. 45) I have observed that even when fully inserted, the lower end of the RCCA absorber material will sit some distance above the bottom of the active fuel region in the UK HPR1000 core. This design characteristic is novel in the UK and means there is significant axial heterogeneity at the bottom of the core with the RCCAs inserted, which I judged could prove a challenge to model using traditional 3D diffusion-based physics codes. I therefore requested additional validation and sensitivity studies to show that the SDM predictions were adequate for these core designs using the COCO physics code, including that the uncertainty allowance associated with modelling of total RCCA reactivity worth was bounding. My assessment of this topic is reported in subsection 4.13.

4.2.1.5 Other Nuclear Design Aspects

119. I also undertook a briefer review of all the other parameter limits in the UK HPR1000 nuclear design basis and their evaluations for the first and equilibrium cycle designs (Ref. 26, Ref. 27 and Ref. 28).

120. Ref. 26 does not include any requirement for the core to remain stable to xenon oscillations, which may occur if the power changes in one part of the core are effectively decoupled from those in another due to the core's physical size and differences in local reactivity. Drawing on SAP ERC.3, I would expect to see a requirement that any such oscillations would decay over time for this design and that the core is therefore stable. I requested further information on this topic in RQ-UKHPR1000-0574 (Ref. 33). In response the RP provided a qualitative summary of its arguments for how and why the core would be stable to radial, azimuthal and axial xenon oscillations, together with a commitment to produce a further quantitative analysis as evidence.

121. I considered the information provided by the RP to be sufficient and judged it unnecessary to sample the quantitative evidence in GDA. The UK HPR1000 core design is relatively short compared to other modern PWRs that have been through GDA in the UK so I judge should be less susceptible to axial oscillations. I have also seen evidence in Ref. 26 that limits are set on axial offset (a measure of the relative magnitude of power in the top and bottom of the core), which will help to control such oscillations. I do not consider the UK HPR1000 core design particularly likely to suffer from radial or azimuthal oscillations and I note that the in-core neutron detectors (discussed further in subsection 4.10) should enable any unexpected asymmetry in power distribution that could lead to such effects following a power manoeuvre to be observed by operators. Overall, I therefore consider the lack of any explicit requirements associated with xenon stability in the nuclear design basis to be a minor shortfall.

122. Ref. 26 includes minimum reactor core sub-criticality margins under different shutdown conditions, which align with criteria commonly applied to other nuclear reactors. When the core is shutdown for refuelling, the RP's criteria is that k-eff be < 0.95 with all RCCAs inserted, and < 0.99 with all RCCAs extracted. I am satisfied that the criteria provide suitable and sufficient margin to criticality for a whole core, in accordance with the principle of SAP ERC.1. Ref. 27 and Ref. 28 present minimum required boron concentrations as a function of core burnup for each cycle to ensure these criteria are met. I am satisfied these calculations are conservative and that there is sufficient margin between the boron concentration specified during a refuelling outage and the

concentrations required to meet the sub-criticality criteria. My assessment of criticality margins in the specific case of core re-load is presented in subsection 4.10.

123. I have also briefly reviewed the RP's evaluation of Doppler coefficients. It shows that the requirements in Ref. 26 are met with adequate margin over the required range of core conditions. In most core operating states the margins available to the limits are large. I judge that these large margins will provide for additional fault tolerance.

Neutron Sources

124. I reviewed the justification for the number and type of neutron sources specified in the nuclear design. While they are necessary to provide a stable, detectable neutron flux signal during re-load and start-up, such sources become additional nuclear material requiring disposal and require additional measures to be taken during manufacture, transport, installation, operation and removal to ensure that doses incurred by personnel are minimised.

125. The GDA core design includes three primary neutron source assemblies and three secondary neutron source assemblies in the first cycle core design (Ref. 27), and three secondary neutron source assemblies in the equilibrium cycle core design (Ref. 28). I am satisfied that loading sequence operating procedures and an analysis of detector count rates described in the analysis report for the first cycle loading sequence (Ref. 46) show that the neutron count rate will be sufficiently high to be both detectable and stable throughout the first cycle loading process.

126. Once the equilibrium cycle is reached, the core's intrinsic neutron source will become much higher due to the presence of a large quantity of irradiated fuel from previous cycles. There is some incentive to remove secondary neutron source assemblies from the design for the reasons outlined in paragraph 124. As a result, the RP has submitted an optioneering report to review the design and requirement for secondary neutron source assemblies in UK HPR1000 (Ref. 47). The report concludes that several options are feasible, including keeping the design as it stands, or removal of the secondary sources completely after the third cycle. Both of these options have precedent. The report states that more detailed optioneering will be performed in the site-specific phase to enable the licensee to select the best option. From a Fuel and Core perspective, I support the performance of optioneering activities to determine the ALARP option, balancing the requirements of core monitoring against the impacts of production, through life management and eventual disposal of neutron sources.

4.2.2 Strengths

127. Following my assessment of the UK HPR1000 nuclear design and safety case I have identified the following strengths:

- Overall, I am now satisfied that sufficient requirements have been placed on the UK HPR1000 nuclear design such that it should meet the principles of the relevant SAPs, in particular EKP.2, EAD.1, ERC. 1, ERC. 2 and ERC. 3.
- In particular, burnup design limits are well within the range of burnup experienced worldwide by Framatome AFA 3GAA fuel assemblies with M5 clad.
- The majority of nuclear design basis requirements sampled are met with adequate margin over the required range of core conditions. In most core operating states the margins available to the limits are large. This means that the true core response to faults will be more benign than predicted by the fault analyses.

4.2.3 Outcomes

128. Following my assessment of the UK HPR1000 nuclear design and safety case I have identified several minor shortfalls.

4.2.4 Conclusion

129. Based on the outcome of my assessment of the UK HPR1000 nuclear design and safety case, I have concluded that they are adequate for GDA. I have raised one Assessment Finding through my assessment of the overall ALARP demonstration for the core design (subsection 4.14) that affects the nuclear design. In other respects I am satisfied that relevant expectations I have derived from SAPs EKP.1, EKP.2, EKP.3, EAD.1, ERC.1, ERC.2, ERC.3, ERC.4 and FA.7 are met. I am therefore satisfied that following resolution of that Assessment Finding, the nuclear design will help to ensure that risks to health and safety as a result of operation of the UK HPR1000 plant are reduced ALARP.

130. Other areas of my assessment are inter-dependent with my assessment of the nuclear design, in particular subsection 4.3 on neutronic and kinetic data, subsection 4.13 on computer code validity and subsection 4.14 on the overall demonstration that risks have been reduced ALARP.

4.3 Neutronic and Kinetics Data for use in Fault Analysis

4.3.1 Assessment

131. Many fault analyses for UK HPR1000 use simplified models of the reactor core with neutronic and kinetic data inputs provided from 3D nuclear design calculations. ONR's assessment of the fault analyses themselves is largely reported in the Fault Studies Step 4 Assessment Report (Ref. 7). The purpose of this subsection of my report is to assess the adequacy of the neutronic and kinetic data inputs provided for fault analysis, on a sampling basis.

132. I consider the most important SAPs in this part of my assessment to be FA.7 and AV.3. I expect that a demonstration be provided that the neutronic and kinetic data enables conservative fault analyses. The nuclear analysis codes used to produce this data for UK HPR1000 are best estimate codes, meaning that they are designed to produce a set of nuclear predictions that are as accurate as possible for a particular set of conditions with a particular set of inputs, rather than being designed to produce results that are always conservative. Many of the inputs supplied to the codes, such as geometry and nuclear cross-section data, are also of a best estimate nature. I therefore expect uncertainties in the codes' predictions to be understood and accounted for. As the values of most neutronic and kinetic parameters vary between cycles and as a function of burnup or power, I also expect that it is clear what core operating states the data is intended to be used for and, when a common set of data is used for a range of conditions, that it is bounding for all of them.

133. The RP's safety case includes a number of documents reporting on core neutronic data generated for various purposes. I elected to sample the following four reports because of their importance to safety: General Nuclear Data and Key Neutronic Data (Ref. 48), Neutronic data for LOCA Analysis (Ref. 36), Power Envelop for Frequent Fault (Ref. 49) and the Decay Heat Report (Ref. 50). Ref. 48 and Ref. 36 provide a variety of bounding neutronic and kinetic data that is used in the analysis of a range of faults, some with potentially high consequences. Ref. 49 provides peak power and power ramp rates in normal operation and frequent faults as a function of fuel rod burnup. These data are used in the fuel rod design justification (discussed in subsection 4.4 of this report) to provide a demonstration that some fuel design criteria

such as fission gas pressure or clad strain are not exceeded in faults. Ref. 50 provides decay heat data for use in the majority of fault analyses.

4.3.1.1 General Nuclear Data for Fault Analysis

134. Ref. 48 provides data for the negative reactivity insertion curve following reactor trip, moderator and doppler reactivity coefficients, effective delayed neutron fraction, prompt neutron lifetime, R bank differential worth and SDM. Each individual data item is calculated to envelop all core operating states. I judge that adequate margin has been incorporated in most data in Ref. 48 to cater for uncertainties and to provide additional safety margin on top. I observed two potential exceptions to this, which were assumptions about the integral RCCA bank worth and the MTC during shutdown.

135. The overall combination of the recommended data provides for very conservative predictions of the core response in faults. In general I judge this to be an adequate approach. However, it will produce some fault analysis predictions that are somewhat unrealistic (overly conservative) and do not represent the true safety margin. This is because the limiting values of some of the different bounding parameters presented together are not predicted to occur in reality in similar core operating states or even the same operating cycle.

Integral RCCA Bank Worth

136. In Ref. 48 the calculation of negative reactivity insertion curve as a function of insertion depth following reactor trip assumes a total integral worth for all RCCA banks of [redacted], rather than a calculated value for each core operating state. The RP has presented evidence that the [redacted] figure is bounding for all conditions in the equilibrium cycle design and bounding above [redacted]% full power for all cycles. However, during some of cycle 1 (and cycle 2) the [redacted] figure is not conservative below [redacted]% power.

137. In response to RQ-UKHPR1000-1047 (Ref. 33) the RP identified which fault analyses used this data at low powers. Two cases were identified, an RCCA bank withdrawal (Ref. 51) and excessive secondary load increase (Ref. 52), in which the negative reactivity insertion curve calculated with the [redacted] assumption is applied at powers below [redacted]%. However, these are not limiting cases and the RP has supplied additional Departure from Nucleate Boiling Ratio (DNBR) results in response to RQ-UKHPR1000-1047 for these specific cases with a lower, bounding integral bank worth to show that significant safety margin remains. I judge this to be sufficient for GDA because significant safety margin exists in these cases and because the overall combination of neutronic and kinetic data supplied is still likely to produce conservative results. However, I consider the use of a non-conservative integral bank worth assumption at low power to be a minor shortfall in the safety case. Both of the two faults identified above are considered more widely in the Ref. 7.

MTC during Shutdown

138. In Ref. 48 the maximum absolute value specified for MTC is 0 pcm/ºC. As identified previously in this report, in some operating conditions when the plant is shutdown with high boron concentrations, the MTC will inevitably be positive, so a value of 0 pcm/ºC will not be bounding. In collaboration with Fault Studies inspectors, I therefore raised RQ-UKHPR1000-0809 (Ref. 33) seeking further evidence that relevant fault analyses were not sensitive to a positive MTC or, where they were, seeking updated fault analyses to capture these effects and show that fuel criteria were still met. In response, the RP submitted its synthesis report on positive MTC (Ref. 39). This report describes the various states the core will go through when shut down and during the start-up process (from the point of refuelling through to initiation of power raise), presents the

range of MTC values that could exist in each plant state and identifies a list of existing fault analyses that could therefore be impacted by a positive MTC.

139. The RP makes qualitative arguments in Ref. 39 to explain why the majority of faults will not lead to a loss of fuel integrity if they occur from initiating conditions in which there is a positive MTC. The RP has undertaken additional fault analyses with a positive MTC to provide quantitative evidence that acceptance criteria are met for an RCCA ejection accident (Ref. 41) and an uncontrolled RCCA bank withdrawal from zero power (Ref. 53), both from an initial condition just prior to criticality as the R banks are being withdrawn. I am satisfied that the RP's selection of initial conditions should provide for a conservative analysis, when considering the different states described in the document that will produce a positive MTC. The set of results presented is not as thorough as in the individual fault analysis reports submitted under PCSR Chapter 12. However, the RP has been able to show that all of the acceptance criteria are met in these cases with larger safety margin than in the original analysis that was reported in Ref. 41 and Ref. 53. I judge this plausible because in the fault analysis with zero MTC, the RP consider a much wider range of inserted RCCA bank positions with an initial condition of k_{eff}=1 at zero power. The analysis in Ref. 39 only considers a more realistic but conservative set of initial RCCA bank positions that could occur with a positive MTC and k_{eff} close to 1 if operating rules are broadly complied with during start-up. As a result, the reactivity insertion and power peaking that occurs in the positive MTC cases is far less than in Ref. 41 and Ref. 53 for these two faults. The effects of these changes outweigh the effect of the positive MTC.

140. Overall, I am therefore satisfied the evidence in Ref. 39 shows that the possibility of a positive MTC when shut down does not mean the fault analyses in Ref. 41 and Ref. 53 are non-conservative. In my opinion further supporting evidence should have been provided to support the qualitative arguments presented for other faults. However, I judge the risk to the design from this to be low. I therefore consider this matter to constitute a minor shortfall in the safety case.

Other General Neutronic and Kinetic Data

141. As well as sampling Ref. 48, I sampled in conjunction with Fault Studies inspectors a range of the fault transient analysis reports submitted by the RP. This was to verify proper application of the data in Ref. 48 and support the Fault Studies Assessment Report (Ref. 7).

142. Although Ref. 48 does not specify a value of F_{dH} to be used in fault analyses, the majority of UK HPR1000 fault analyses that use the general neutronic data specified in Ref. 48 also use an assumption that F_{dH} is equal to the design limit specified in the fuel management report (Ref. 29), including uncertainty. As discussed in subsection 4.2, I am satisfied that adequate design limits have been set on power peaking factors and that adequate evaluations of the core design have been completed to ensure these limits are met. I therefore judge that the F_{dH} design limit is a conservative assumption to use for radial power peaking within these fault analyses.

143. Some fault analyses for UK HPR1000 use detailed models of the reactor core to provide specific neutronic data for use in that fault, rather than using the data provided in Ref. 48. This is particularly necessary for reactivity faults in which the power distribution is distorted by fast RCCA movements. I have not sampled the RP's calculations of individual neutronic parameters for these faults but have satisfied myself in conjunction with Fault Studies inspectors that specific neutronic calculations have been conducted for the correct range of faults and, as reported in subsection 4.13, that the codes used to do this work are adequate. I have also gained confidence that the neutronic data used is conservative through some of my TSC's confirmatory fault analyses undertaken during GDA (Ref. 22). The confirmatory analyses conducted

by my TSC for the RCCA bank withdrawal fault from zero power and for the rod drop fault both show that the RP's neutronic calculations give a very conservative set of results. These faults were the two analysed by my TSC for which the RP use specifically-calculated neutronic data.

144. For some of the Anticipated Transient Without Scram (ATWS) faults assessed in Ref. 7, the assumptions about reactivity coefficients have been changed from those recommended in Ref. 48, such that the selected coefficients do not bound those predicted at BOC in cycle 1 (and cycle 3). I am able to judge from the data in the fuel management report (Ref. 29) that most of the period when this is the case will be early in cycle 1, when the predicted F_{dH} is significantly lower than it is in other cycles. This should partially compensate for the slightly non-conservative MTC assumption used for these faults in terms of the impact on DNBR. However, this argument is not made by the RP. The Fault Studies inspector has considered this matter as part of their wider assessment of the adequacy of the safety case for ATWS faults in Ref. 7.

145. In my assessment of the risks associated with fuel deposits (subsection 4.9) I also considered whether any impact of deposits on the general neutronic parameters in Ref. 48 had been adequately addressed. In its fuel crud assessment report (Ref. 54) the RP has presented sensitivity analyses to demonstrate that the negative reactivity insertion curves and SDM data recommended for use in fault analyses in Ref. 48 remain conservative in the presence of fuel crud. Other neutronic parameters in Ref. 48 are not significantly affected. I am therefore satisfied that the data recommended in Ref. 48 for use in fault analyses will remain conservative in the presence of fuel crud.

4.3.1.2 Neutronic Data for LOCA Analyses

146. Ref. 36 presents neutronic data for use in all design basis LOCA analyses and LB-LOCA. This data includes changes in reactivity as a function of moderator density and a range of power peaking information designed for the specific modelling approach used for the reactor core in the UK HPR1000 LOCA analysis. Although LB-LOCA is outside of the generic UK HPR1000 design basis (see sub-section 4.8), Ref. 36 presents a single set of neutronic data for all LOCA analysis, together with evidence that it is conservative. The overall approach to the LOCA analysis including thermal hydraulic modelling is assessed in Ref. 7, with some fuel modelling aspects covered in subsection 4.8 of this report. I am satisfied that Ref. 36 demonstrates a conservative approach to provision of neutronic data for LOCA analyses, adequately accounting for uncertainties and covering the full extent of the operating cycle. This meets my expectations described in paragraph 132.

4.3.1.3 Power Envelope Data for Fuel Design Analysis in Frequent Faults

147. Ref. 49 defines bounding peak linear powers and peak linear power escalations to which the fuel could be subject in frequent faults, as a function of fuel rod burnup. Ref. 49 covers all frequent faults with a frequency of > 10^{-3} per year, but presents qualitative arguments for excluding some faults from quantitative analysis due to them being bounded by other faults, which I find to be reasonable. The results are presented as a function of burnup at intervals of 1000 MWd/Te, which I judge adequate to allow degradation due to burnup to be properly accounted for in the fuel design analysis, in accordance with SAP EAD.2. The analysis uses the PINE and COCO codes, which I have assessed in subsection 4.13 and am satisfied are suitable for this application. Following some interaction during GDA Step 4, the analysis in a previous version of Ref. 49 was revised to include an increased uncertainty allowance, which I am satisfied accounts adequately for uncertainties in outputs from these codes. With this update, I am satisfied that all aspects of the work meet my expectations described in paragraph 132.

4.3.1.4 Decay Heat Data

148. I have determined that the RP applies decay heat data of three types in its DBA:

- the decay heat data used for the short-term analysis of LOCA faults, in which compliance with fuel acceptance criteria is demonstrated, are calculated by the LOCUST-K code (Ref. 55) using the conservative approach contained in the US Nuclear Regulatory Commission (NRC) Appendix K methodology;
- the decay heat data used for the long-term analysis of LOCA faults, in which long-term cooling and sub-criticality are verified, are calculated specifically for the UK HPR1000 fuel using the GINKGO and PALM codes; PALM is a best-estimate code and an uncertainty of 2 standard deviations has been added to the PALM results for this application; and
- the decay heat data used for the analysis of other design basis faults are calculated specifically for the UK HPR1000 fuel using the GINKGO and PALM codes; an uncertainty of 1.645 standard deviations has been added to the PALM results for this application.

149. The RP has submitted a decay heat report (Ref. 50) to present the results of decay heat calculations for the UK HPR1000 fuel using GINKGO and PALM. This includes predictions of decay heat in the core and in the spent fuel pool.

150. GINKGO is a 1-D system code containing a point-kinetics representation of the core. Its validity is assessed by the ONR Fault Studies inspectors in Ref. 7. PALM is a depletion code used to provide source terms for decay heat analysis and also source terms for other work such as shielding and accident chemistry analyses. My assessment of PALM for application in decay heat calculations is reported in subsection 4.13 and I am satisfied with its validity for this purpose.

151. The decay heat term calculated by the RP using GINKGO is the power from residual fission reactions. These decay very quickly after shutdown and the RP claim they are negligible after approximately 600 seconds. Ref. 50 describes the assumptions used for important inputs to the core decay heat analysis, which I am satisfied are all conservative. Subject to the conclusions of Ref. 7 on the validity of the code, I am therefore satisfied that the residual fission power term is calculated conservatively for the purposes of DBA in accordance with SAP FA.7.

152. The decay heat terms calculated with PALM are the power due to continuing radioactive decay of fission products and actinides remaining in the fuel. The PALM analysis assumes that the core has been operated at full power prior to shutdown and accounts for all the different cycle designs within Ref. 29.

153. The spent fuel pool calculations in Ref. 50 neglect the residual fission power term calculated by GINKGO because of its short-term nature, which means it is negligible by the time fuel has been unloaded from the core. The total spent fuel pool decay power calculated using PALM uses conservative assumptions for the number of fuel assemblies in the pool and the decay time for the fuel assemblies after unloading from the core.

154. I have verified that the applicable uncertainties reported in the PALM V&V report (see subsection 4.13) have been applied to the results in Ref. 50 for both core decay heat and spent fuel pool decay heat. However, as described in paragraph 148 the actual magnitude of uncertainties applied by the RP is different for LOCA faults and other faults.

155. I am satisfied that the application of 2 standard deviations to the data for long-term LOCA analysis is consistent with RGP from previous GDA and will produce a conservative result. The application of 1.645 standard deviations for other design basis

faults is also consistent with the approach in one previous GDA. To verify that this approach is sufficiently conservative specifically for UK HPR1000, I have also undertaken my own comparison between the decay heat data in Ref. 50 and that produced by a conservative implementation of the ANSI/ANS-5.1-1994 decay heat standard (Ref. 55) with an uncertainty of 2 standard deviations. The results were similar, with differences of ~ 0.001-0.002% full power. Fault Studies inspectors have confirmed that the safety margins predicted for limiting faults in the UK HPR1000 DBA are not likely to be sensitive to such small changes in decay heat predictions. On balance, I therefore judge that the uncertainty allowances applied to decay heat predictions in Ref. 50 are adequate.

156. Overall, I am satisfied that the decay heat data generated for use in UK HPR1000 DBA meets my expectations described in paragraph 132.

4.3.1.5 Impact of Grid Code Compliance on Core Data for use in Fault Analysis

157. The RP has submitted Ref. 42 to present an analysis of potential gaps in compliance of the generic UK HPR1000 design with UK grid code requirements. These gaps are explored and the topic assessed holistically in the Electrical Engineering Assessment Report (Ref. 57). From a reactor core perspective, the impact of the UK grid code is that it requires power plants to provide certain capabilities for flexible operation, such as periods of low power operation and the ability to change power quickly in response to grid frequency variations.

158. Ref. 42 aims to demonstrate the feasibility of potential post-GDA design modifications to enable additional operating modes and close the identified gaps in grid code compliance. The reactor core safety case needs to demonstrate that the fundamental safety functions will be delivered with sufficient confidence in all permitted operating modes, as set out by SAP ERC.1. In this sub-section, I consider whether the impact of the potential design modifications on core neutronic and kinetic data have been analysed and considered by the RP in sufficient depth to support the conclusions of the feasibility study. In sub-section 4.7.1.3, I also consider the impact of the potential design modifications on PCI fault analysis.

159. Ref. 42 presents a potential post-GDA design modification to allow the UK HPR1000 plant to operate with a Primary Frequency Control (PFC) capability of +/- 10% full power, extended from the GDA design’s capability of +/- 3% full power. Due to the necessity to change primary power quickly in response to grid frequency fluctuations, this implies the need to manage the reactivity defect caused by a 10% power transient entirely using RCCA motion, because changes in boron concentration are too slow.

160. When a fast power reduction is required, RCCAs can in principle always be inserted to compensate the reactivity defect. However, if a fast power increase of up to 10% is required, sufficient RCCA worth may not always be available to withdraw from the core. Specifically, Ref. 42 states that only the R banks are expected to be inserted in the GDA core design during Extended Low Power Operation (ELPO), with the power reactivity defect compensated by increasing boron concentration when entering this mode. Ref. 42 describes a potential future design modification in which the plant would operate in ELPO mode by inserting the power compensation (G/N) banks rather than increasing boron concentration, so that if a fast power increase of up to 10% is subsequently required, sufficient RCCA worth is available to withdraw. This proposed modification would align the RCCA control logic for ELPO mode with that used for other power changes such as daily load follow operations, which the RP has described in more detail in a functional requirements document for the rod position, indication and control system (Ref. 58). However, the RP has identified in Ref. 42 that operating like this in ELPO would have an impact on the burnup distribution in the core due to the shadowing effects of the inserted RCCA banks, and consequently on some of the

neutronic core data that is used in DBA. Ref. 42 provides an analysis of the impact of this on a selection of faults to support a demonstration that the modification is feasible post-GDA.

161. I have assessed this topic with consideration of SAPs ERC.1 and FA.7, whilst recognising that the intent of Ref. 42 is to present a feasibility study and not a full safety case.

162. Within Ref. 42 the RP has analysed four different equilibrium cycle cases, to predict the effect of a single 30-day ELPO period occurring at BOC, Middle of Cycle (MOC) or EOC, when compared to the baseline case of full power throughout. It presents evidence that the effect on power peaking factors, axial offset, boron concentration and all generic neutronic data reported in Ref. 48 is small and bounded by existing safety case assumptions. It also presents evidence of the impact on a selection of reactivity faults that use specifically calculated neutronic data. The faults selected are an RCCA drop (Ref. 59, the frequent reactivity fault for which the RP show smallest margin to acceptance criteria) and an RCCA ejection (Ref. 41, the infrequent reactivity fault for which the RP show smallest margin to acceptance criteria). The RCCA drop results show that margin is improved over the base case, while the RCCA ejection results show that margin is reduced. This is due to a slight increase in the reactivity worth of the ejected RCCA.

163. I am satisfied that the RP has considered an appropriate range of core conditions, neutronic data and fault analyses to inform this feasibility study. However, informed by SAPs ERC.1 and FA.7 I observed two issues with the analysis that could potentially lead to gaps in a future safety case:

- The assumed maximum period of ELPO is 30 days due to constraints imposed by PCI analysis (see subsection 4.7). However, Ref. 42 makes separate claims that multiple periods of ELPO in a cycle could be allowed if PCI margin is recovered during operation at full power. A total duration of over 30 days ELPO in a cycle would not be bounded by the analysis discussed above, so the analysis may not cover all permitted operating modes.
- The predicted margin to fuel melt temperature in the RCCA ejection accident occurring after the end of 30 days ELPO is reduced in comparison to the existing safety case analysis. It remains positive, but small. Any further deterioration due to an increased total ELPO period could therefore lead to a prediction of local fuel melt. Although allowed by the RP's own technical acceptance criteria for an RCCA ejection accident (see subsection 4.5), this would make the radiological consequences analysis in the current safety case less conservative.

164. I requested further information in RQ-UKHPR1000-1729 (Ref. 33) to understand the potential challenge that these issues posed. In response, the RP has provided sensitivity analysis to show that if cycle-specific parameters are assumed for reactivity coefficients, delayed neutron fraction and prompt neutron lifetime then the margin to fuel melt in an RCCA ejection is substantially increased. The margin increase due to this analysis optimisation is much larger than the margin decrease due to the ELPO period. The RP therefore argues that a limitation of 30 days on total ELPO in a cycle will not be required. It has also added an additional appendix to Ref. 42, showing that margin to fuel melt is retained in an RCCA ejection accident, and all generic nuclear data remain bounded by safety case assumptions, if a 2 day ELPO / 5 day full power operating cycle is continued throughout a full fuel cycle.

165. Overall, I am satisfied that from a Fuel and Core perspective, the RP has demonstrated the feasibility of introducing the design modification described in Ref. 42 post GDA to improve PFC capability. Based on the information in Ref. 42, I judge that it

should be possible for the licensee to provide an adequate safety case for the core neutronic and kinetic data assumed by fault analysis in such a modified design. However, the information in Ref. 42 is not a complete safety case for the modified design. The core neutronic and kinetic data used in the safety case in GDA are adequate for UK HPR1000 with a PFC capability of +/- 3% full power.

166. The implementation of any design modification(s) made for the purposes of grid code compliance will be tracked through Assessment Finding AF-UKHPR1000-0020, which has been raised in Ref. 57. As part of the resolution to that Assessment Finding, if PFC capability is extended beyond that of the GDA design then I would expect the licensee to extend the existing feasibility study, completing further safety analyses to ensure that all core data used in DBA remains conservative for the full range of permitted operating modes.

4.3.2 Strengths

167. Following my assessment of the UK HPR1000 neutronic and kinetics data for use in fault analysis I have identified the following strengths:

- The documentation I have sampled shows that the RP has taken a conservative overall approach to the generation of nuclear data for use in DBA.
- Uncertainties have been adequately accounted for and evidence has been provided that the data bounds all relevant core operating states and cycle designs for the vast majority of the data presented.

4.3.3 Outcomes

168. I observed some minor shortfalls during my assessment of this part of the safety case.

4.3.4 Conclusions

169. Based on the outcome of my assessment of the neutronic and kinetic data provided for use in fault analysis, I have concluded that the data takes adequate account of uncertainties and is combined in a way that will provide for conservative DBA. It is valid for the circumstances for which it is provided except in a limited number of cases, for which the RP has provided evidence that the conclusions of the fault analyses are unaffected. Overall, my expectations derived primarily from SAPs FA.7 and AV.3 have been met.

4.4 Fuel System Design

4.4.1 Assessment

4.4.1.1 Design Summary

170. The UK HPR1000 fuel system design consists of the fuel assemblies, RCCAs and Stationary Core Component Assemblies (SCCAs). The UK HPR1000 fuel assembly is the Framatome AFA 3GAA model. GNSL has submitted the AFA 3GAA fuel assembly description document containing a detailed description (Ref. 44). The key aspects are:

- the fuel assembly contains a 17 x 17 array of fuel rod, guide tube or instrumentation tube locations with a total of 264 fuel rods in one assembly;
- the fuel rod active length is 12 feet (3.658 m);
- all fuel assemblies have 24 guide tubes and one instrument tube;
- for structural strength and stability, the fuel assemblies have top and bottom nozzles, eight structural grids and three mid span mixing grids;
- the nuclear fuel pellets are UO_2 or a composite of UO_2 and the burnable poison Gd_3O_2;

- the fuel rod cladding, guide tubes and mixing grids are manufactured from a Framatome proprietary zirconium alloy named M5; and
- the top nozzle, bottom nozzle and anti-debris device are manufactured from stainless steel (of various grades) and the hold-down spring from Inconel.

171. The RP has submitted its AFA 3GAA operating experience report (Ref. 31), which presents the extensive OpEx accumulated with AFA 3G fuel assemblies, distinguishing between those with M5 or other clad materials and between AFA 3GAA, AFA 3GA and other AFA 3G designs. Ref. 31 also provides separate data for assemblies with mid-span mixing grids (post-1998) and with updated spacer grid designs to improve assembly handling (post-2008). The data presented easily bounds the expected fuel burnup in the UK HPR1000. This provides a clear demonstration of the level of OpEx accumulated with fuel system designs very similar to that specified for the UK HPR1000, and also shows that learning from experience has been incorporated. I consider this good practice.

172. The RP has also submitted a specific ALARP demonstration report for the fuel system (Ref. 60). I have considered this against expectations derived from NS-TAST-GD-005 (Ref. 12), together with the wider core design ALARP demonstration, in subsection 4.14 of this report. It is therefore outside the scope of this subsection of my report.

173. The UK HPR1000's RCCA design is the Framatome HARMONI model, for which the RP has submitted a design description in Ref. 45. HARMONI RCCAs consist of a 'spider' structure that supports a cluster of 24 control absorber or stainless steel rods, which are inserted and withdrawn from the fuel assembly guide tubes. The control absorber rods contain silver-indium-cadmium absorber material within a stainless steel cladding. As described in subsection 3.1, the UK HPR1000 core includes both black RCCAs (used in the R, N and S banks) and grey RCCAs (used in the G banks). In black RCCAs, all 24 rods are control absorber rods. In grey RCCAs, 8 rods are control absorber rods and 16 rods are stainless steel.

174. The SCCAs (described in Ref. 61) are designed with guide tube ('thimble') plugs to minimise unnecessary flow bypass in fuel assemblies without RCCAs, or alternatively to hold the neutron sources inside the fuel assemblies. The role of the neutron sources is to enhance the neutron flux level when the core is sub-critical, for the purposes of monitoring of core conditions as discussed in subsection 4.2.1.5. The SCCAs consist of a spider structure supporting a cluster of rods that contain the radionuclide materials to generate neutrons. The rods are always inserted into the fuel assembly guide tubes during operation. I have not targeted the SCCA safety case for assessment because the consequences of SCCA failure are low relative to those for the fuel assembly or RCCA.

4.4.1.2 Key Guidance and Relevant Good Practice

175. I have primarily applied SAPs EKP.4, ERL.1, ERC.1 and ERC.2 in reaching judgements about the adequacy of the UK HPR1000 fuel system design substantiation. I expect that the safety case should identify the safety functions that the fuel system design is expected to provide and that any safety case claims on the reliability of safety functions should be justified through a suitable analysis. This should be based upon relevant OpEx and account for known physical phenomena.

176. I consider SAP ERC.1 to be relevant to all aspects of the fuel system design that affect the ability of the plant to meet the fundamental safety functions. I consider that ERC.2 is most applicable to the RCCA Design. Informed by ERC.2, my expectation is that the RCCA's shutdown function should not be inhibited by mechanical failure, distortion, erosion and corrosion of plant components, or by the physical behaviour of the reactor coolant, under normal operation or design basis fault conditions.

177. NS-TAST-GD-075 (Ref. 11) contains relevant advice to inspectors on fuel failure mechanisms in normal operations, design criteria for fault analyses and design criteria for the limits on the fuel system's structural components. It also refers to IAEA requirements and guidance in SSR-2/1 (Ref. 17) and SSG-52 (Ref. 19). I consider the following requirements in Ref. 17 to be RGP for fuel system design:

- Requirement 43: Performance of fuel elements and assemblies. Fuel elements and assemblies for the nuclear power plant shall be designed to maintain their structural integrity, and to withstand satisfactorily the anticipated radiation levels and other conditions in the reactor core, in combination with all the processes of deterioration that could occur in operational states.
- Requirement 44: Structural capability of the reactor core. The fuel elements and fuel assemblies and their supporting structures for the nuclear power plant shall be designed so that, in operational states and in accident conditions other than severe accidents, a geometry that allows for adequate cooling is maintained and the insertion of control rods is not impeded.

178. Ref. 17 has additional expectations for the designs of RCCA as part of Requirement 45. The design expectation is that the reactivity control devices should take account of wear and the effects of irradiation, such as burnup, changes in physical properties and production of gas. This is to ensure that these effects do not reduce the RCCA's effectiveness in controlling core reactivity or shutting down the reactor.

179. Ref. 19 provides more detailed technical guidance and expectations for fuel and RCCA designs. I have used this detailed guidance to inform my expectations and have referenced it where relevant in the following subsections.

180. In addition to guidance, I have identified OpEx for fuel failures from IAEA (Ref. 62) and a technical review of fuel safety criteria from OECD/NEA (Ref. 21). I used these documents to inform the technical basis for my detailed assessment of the RP's safety justification and as a source of independent information to inform my judgements.

181. My high level expectations for the safety case for the UK HPR1000 fuel system design are that it should identify the relevant safety functions, should provide adequate evidence to justify that the design meets those functions and should address the relevant phenomena that could challenge the safety functions.

4.4.1.3 Fuel System Safety Functional Requirements

182. I have structured my assessment of the UK HPR1000 Fuel System Design to start at the safety functional level. I used this to inform my judgement on the coverage of the RP's safety case (considering both SAPs EKP.4 and ERC.1).

183. The RP has presented safety functional requirements applicable to the fuel in power operation, start-up and shutdown within PCSR Chapter 5 (Ref. 3), which I have summarised as follows:

- in conjunction with the core nuclear and thermal hydraulic design, the fuel system design shall ensure that the heat produced in the fuel can be removed by the reactor coolant;
- in conjunction with the core nuclear design, the fuel system design shall ensure control of core reactivity and the ability to return the reactor to a safe state using two diverse shutdown systems;
- the fuel system design and performance shall preclude the release of radioactive material during operation in DBC-1 and DBC-2 by maintaining the integrity of fuel cladding;

- the fuel system design shall ensure the preservation of an assembly array geometry to enable the insertion of RCCAs to shut down the reactor in DBC-3 and DBC-4 conditions; and
- the fuel system design shall ensure the preservation of an assembly array geometry to enable the cooling of the reactor core in DBC-3 and DBC-4 conditions.

184. The safety functional requirements on the fuel system in Ref. 3 have been derived to meet the safety functions identified in PCSR Chapter 4, with consideration of RGP from the IAEA in Ref. 17. I am satisfied that this provides a structured approach to identification of safety functional requirements.

185. I judge that the first and second safety functional requirements in the list above are adequate. They meet the expectations set by paragraph 540 of the SAPs that the safety functions considered include control of reactivity and removal of heat from the core.

186. The third safety functional requirement in the list above is aligned with the third safety function expected by paragraph 540 of the SAPs, confinement of radioactive material. The functional requirement itself is adequate. I have assessed the design conditions in which it is applied against the expectations set by advice in Ref. 11. It only refers to DBC-1 and DBC-2 conditions, but the RP states in Ref. 3 that in practice this expectation also applies to DBC-3 faults with frequency greater than 10^{-3} per year. I observed that if only designed to these safety functional requirements, the fuel system design may not prevent the release of radioactive material from the fuel clad in infrequent faults with a frequency less than 10^{-3} per year or in Design Extension Conditions (DEC). However, together with Fault Studies inspectors I have sought a demonstration that the consequences of such faults are reduced ALARP for UK HPR1000. The RP has provided an assessment of the UK HPR1000 design in GDA against this expectation, which I have assessed from a Fuel and Core perspective in subsection 4.14 of this report.

187. The fourth and fifth safety functional requirements in the list above should ensure that the control of reactivity and removal of heat functions can be maintained in infrequent faults. The functional requirements themselves are adequate. I have assessed the design conditions in which they are applied against the expectations set by advice in Ref. 11. I observed that they apply only to DBC-3 and DBC-4 within Ref. 3, and not to DEC. However, in the RP's design condition list and acceptance criteria report (Ref. 63), it states that deterministic fault decoupling criteria for DBC-4 are adopted as strict criteria for DEC in which core melt is not expected (DEC-A). As a result, the same fuel design safety functions fulfilled in DBC-4 conditions will be shown to be fulfilled in DEC-A if the fault acceptance criteria are met in DEC-A fault analysis. ONR's assessment of the DEC-A fault analysis is largely reported in the Fault Studies Assessment Report (Ref. 7).

188. Overall, in my opinion the design conditions in which the third, fourth and fifth safety functional requirements in paragraph 183 are applied by the RP according to Ref. 3 contain some gaps against RGP. However, I am satisfied that submissions exist in the wider safety case that set out to address these gaps for the purposes of powered operations, start-up and shutdown.

189. I have observed no explicit safety functional requirement in Ref. 3 that the fuel system design shall preclude release of radioactive material by maintaining the integrity of fuel cladding during fuel handling and storage. However, the AFA 3GAA fuel assembly design has significant international pedigree (see paragraph 171), which gives me high confidence that the fuel can be handled safely if the fuel handling systems are appropriately designed (fuel handling system design is assessed by Mechanical

Engineering inspectors in Ref. 64). The RP's detailed requirements on fuel handling systems to ensure that fuel integrity is maintained during handling have been reported in a dedicated report on fuel failure mechanisms in fuel route (Ref. 65) and its detailed requirements to ensure that fuel integrity is maintained during wet and dry storage are reported in the long term storage of spent fuel – design criteria report (Ref. 66). I therefore consider the lack of an explicit safety functional requirement on the fuel in handling and storage to be only a minor shortfall in the safety case.

190. Overall, informed by SAPs EKP.4 and ERC.1, I consider that the RP's fuel system design safety functional requirements stated in PCSR Chapter 5 (Ref. 3) contain some minor shortfalls, but I judge that these are not significant enough to undermine my confidence in the fuel system design.

191. Adequate substantiation of the fuel safety functional requirements is necessary to provide sufficient confidence that the expectations of SAP ERC.1 will be met in normal operations and design basis faults. I have reviewed the adequacy of the fuel system design substantiation in the remainder of this subsection of the report.

4.4.1.4 Fuel Assembly Confinement Capability Substantiation

192. This subsection contains my assessment of the substantiation of UK HPR1000 fuel assemblies' confinement safety function. I have assessed the adequacy of the RP's justification to meet expectations set by SAP ERL.1. Informed by Ref. 11, I expect the UK HPR1000 safety case to present adequate arguments and evidence that fuel integrity will be maintained with consideration of each of the following fuel failure mechanisms:

- Grid to rod fretting
- Debris fretting
- Cladding corrosion
- Manufacturing defects
- Cladding collapse
- Clad overheating
- Fission gas release
- PCI

193. In Ref. 3 the RP has listed phenomena that can cause failure of the fuel clad in normal operations and frequent faults. I am satisfied that the relevant phenomena have all been identified.

194. At a holistic level, I have also observed that the AFA 3GAA OpEx report (Ref. 31) provides details of historical fuel reliability, showing the fraction of operating fuel rods that failed in each of the last eight years for different designs of Framatome fuel assembly similar to and including AFA 3GAA. There is a demonstration that the root cause of failures are understood and that improvements are being implemented to reduce their frequency, which I consider to be good practice. I also note that the average failure rate given for 12-foot 17x17 AFA 3G fuel assemblies (including AFA 3GAA) is [redacted] per fuel rod. Were this average failure rate to be repeated in UK HPR1000, it would equate to approximately 3 fuel rods failing throughout the life of the plant. The failure rate is lower than the overall average shown for AFA 3G of all lengths and compares well with advice in NS-TAST-GD-075 (Ref. 11) that "benchmark failure rates are less than one in 100,000 pins".

195. The RP has used fault analysis to demonstrate that the fuel clad will maintain its confinement function in frequent faults, summarised in Ref. 67. I have assessed the fuel aspects of this fault analysis through the technical acceptance criteria identified for the fuel in faults (subsection 4.5) and through the specific analysis conducted for PCI

(subsection 4.7). My assessment in this subsection is therefore primarily focused on mechanisms that could cause fuel clad failure in normal operations.

Grid to Rod Fretting

196. OpEx reported by IAEA (Ref. 62) shows that one of the main sources of in-reactor PWR fuel clad failure is grid to rod fretting. The RP has submitted evidence to demonstrate that this does not present a risk for the UK HPR1000 fuel assembly design in a topical report on grid to rod fretting (Ref. 68). This document provides analysis, the results of experiments and a summary of OpEx for the fuel assembly. It concludes that these sources all show that the design is resistant to this phenomenon. In addition, the RP has submitted analysis for vibration and fretting wear in the fuel assembly mechanical design report (Ref. 69). This includes an evaluation of the impact of cross flow. These results show that these phenomena are unlikely to impact the generic UK HPR1000 design using AFA 3GAA fuel assemblies.

197. Overall, I consider that the RP has presented evidence to demonstrate that the UK HPR1000 fuel design has adequate resistance to grid to rod fretting. This meets my expectations derived from ERL.1.

Debris Fretting

198. The UK HPR1000 fuel assembly has an anti-debris filter to reduce the risk from debris fretting. The RP has submitted a document that describes the capability of the fuel assembly's anti debris filters (Ref. 70). This presents the results of the filters' hydraulic experiments. The results show that the filter has successfully retained a selection of springs, metallic bristles, pins, slender or curly chips, rings, nuts or beads without degradation.

199. OpEx reported by IAEA (Ref. 62) shows that debris filters have reduced the numbers of fuel failures from debris. RGP to protect against debris causing fuel failures is also to use foreign material exclusion measures, according to NS-TAST-GD-075 (Ref. 11), but that is an area for the licensee to develop. As a result, I consider that the RP has submitted adequate evidence to demonstrate mitigation of fuel failure from debris for GDA, meeting my expectations derived from ERL.1.

Clad corrosion

200. Fuel clad corrosion can directly cause clad failure and also change the behaviour of the fuel in faults. The main source of corrosion during normal operation is oxidation of the zirconium in the clad by water at high temperature. This process also produces hydrogen that can diffuse in to the clad and form zirconium hydrides. The RP has submitted a fuel rod design report with analysis results that predict the expected amount of corrosion in normal operations at the end of life. This shows significant margin to the design limit of 100 µm oxide thickness reported in the fuel rod design report (Ref. 71).

201. The AFA 3GAA OpEx report (Ref. 31) details post irradiation examination results for fuel assembly oxidation, and a more limited set of results for clad hydriding. These show that M5 clad has significantly lower levels of oxidation and hydriding compared with Zircaloy-4 clad for equivalent burnup. Measured oxide thicknesses for M5 clad at burnups similar to the UK HPR1000 burnup limit range from just under 10µm to just over 30µm. The data has also been used in the COPERNIC fuel performance code validation report (Ref. 32) to validate the predictions of oxide thickness for UK HPR1000.

202. I consider that these sources of evidence are adequate to demonstrate that the UK HPR1000 fuel design is sufficiently resistant to corrosion phenomena during operation

within the burnup limits defined for UK HPR1000. This meets the expectations of ERL.1. Oxidation and hydriding of the clad also have implications for fuel integrity during long-term dry storage, a topic which I have assessed in subsection 4.5 of this report.

Manufacturing Defects

203. Fuel manufacturing defects can cause fuel failures. Historically, the main source has been defects in fuel rods' closure welds. The fuel vendor, Framatome, already supply fuel to a UK PWR, but the location of manufacture for UK HPR1000 fuel is not yet confirmed. My assessment assumes delivery of the design intent through manufacture and this is a matter for the licensee. Therefore, I have not sampled this area in GDA.

Cladding Collapse

204. During irradiation there is an accumulation of fission gas in the fuel rods that results in a greater pressure inside the fuel rod compared to the reactor coolant. However, for new fuel the initial helium pressure inside the fuel rods is generally not sufficient to completely balance the reactor coolant pressure. This causes inwards creep of the cladding. In some circumstances, if fuel pellet densification occurs and axial gaps grow between fuel pellets, then it is postulated that the clad strain could become excessive and failure occur.

205. The RP has listed creep deformation as a potential fuel failure mechanism in PCSR Chapter 5 (Ref. 3) and the fuel rod design report (Ref. 71) justifies the fuel design against excessive cladding strain. To prevent creep collapse, it states that M5 clad has good creep performance and the fuel pellet is resistant to densification, such that the risk of large axial gap formation is precluded. In addition, Ref. 31 presents creep data from experiments and post irradiation examination of fuel samples with M5 clad.

206. I consider that the RP has submitted adequate evidence to show that fuel cladding collapse is unlikely to impact the fuel design. This has met my expectations derived from ERL.1.

Fission Gas Release

207. The diameter of the cladding decreases during irradiation under the effect of creep until the pellet-cladding gap has closed. However, as the fuel is irradiated, fission gas release can cause the internal fuel rod pressure to rise. If the pellet-cladding gap re-opens due to this effect, it can lead to poor heat transfer within the fuel rod. To avoid re-opening the gap the differential pressure across the fuel clad should be kept below the gap reopening threshold.

208. Ref. 71 considers the impact of fission gas release. It presents analysis for a bounding internal fuel rod pressure against a calculated limit that would re-open the pellet and clad gap. This analysis shows a substantial pressure margin for the bounding case including uncertainties. I therefore consider that the RP has submitted adequate analysis evidence to account for the accumulation of fission gas release and this meets the expectations of ERL.1.

4.4.1.5 Fuel Assembly Structural Integrity Substantiation

209. This subsection contains my assessment of the UK HPR1000 fuel assemblies' structural integrity safety requirements and their substantiation. I have assessed these against the expectations of ERL.1 and ERC.1.

210. Informed by NS-TAST-GD-075 (Ref. 11), I expected the RP to provide a suitable and sufficient mechanical design justification that accounts for:

- Fuel handling and loading
- Power variations
- Temperature gradients
- Hydraulic forces, induced by the core flow and hold-down forces required to maintain core geometry
- Irradiation
- Creep deformation
- External events such as earthquakes
- Postulated faults such as a LOCA

211. The RP has submitted comprehensive mechanical analysis of the fuel assembly design in Ref. 69. This report addresses the hold-down system, the top and bottom nozzles, the guide tubes, the grids and the internal connections. It presents results covering both normal operation and fault conditions. I have verified that the PCSR (Ref. 3) and Ref. 69 list and consider similar mechanical loading phenomena to those listed in paragraph 210; I am therefore satisfied that the relevant phenomena have all been identified by the RP.

212. To gain confidence in the adequacy of the mechanical justification, I have sampled further in two main areas of high safety significance: evidence that fuel assembly bow will not impede RCCA insertion and evidence that structural integrity will be maintained in the most onerous design basis conditions. I considered these most onerous conditions to be the design basis LOCA, a seismic event and an RCP overspeed. Given that the fuel design used in UK HPR1000 is similar to those familiar to ONR from previous assessments, I consider this sample sufficient to gain confidence in the wider mechanical justification for UK HPR1000 fuel.

Fuel Assembly Bow

213. Fuel assembly bow results from irradiation creep of the guide tubes when under load due to axial growth or excessive hold-down force. If excessive bow of the guide tubes occurs, this has the potential to impair RCCA insertion or cause minor damage to the fuel assemblies during core loading and unloading. The RP has justified the design using OpEx submitted in the AFA 3GAA OpEx report (Ref. 31). The OpEx shows that the vast majority of fuel assemblies of this type suffer bow amplitudes that are small enough not to cause problems during refuelling. It states that to date, no RCCA drop anomaly due to assembly bow has been observed for any AFA 3GAA fuel assemblies of similar length to those used in UK HPR1000. Longer fuel assemblies are generally more likely to suffer from larger bow amplitudes.

214. However, fuel assembly bow even at these small amplitudes, although not posing a risk to RCCA insertion, can cause anomalies in thermal performance due to local perturbations of the neutron flux and coolant mass flow rate. This topic was the subject of RO-UKHPR1000-0045 (Ref. 72), which I discuss in subsection 4.6.1.4. As a result of the consideration given to thermal performance effects, the RP has submitted an assessment of the water gap distribution (Ref. 73) providing predictions of fuel assembly bow and the resulting inter-assembly water gap distribution for UK HPR1000. The RP has stated in its over-arching safety justification for the thermal performance effects of fuel assembly bow (Ref. 74) that “the bow behaviour of fuel assembly will be inspected during refuelling periods in the future, and a surveillance plan which specifies the items and ID of fuel assemblies to be inspected will be made in site license stage.”

215. Due to the large amount of OpEx with AFA 3GAA fuel, supported by the predictive analysis and the proposed surveillance scheme for UK HPR1000, I judge that the RP

has provided adequate evidence for GDA to demonstrate that excessive fuel assembly bow will not impact the fuel assembly design's safety functional requirement associated with RCCA insertion.

Fuel Assembly Structural Integrity in a combined Seismic Event and LOCA

216. Seismic events and LOCA apply significant forces to fuel assemblies, which can result in damage to fuel rods, a loss of coolable geometry and prevention of RCCA insertion. SSG-52 (Ref. 19) advises that analysis of the combined forces can give confidence in the integrity of the fuel assemblies in less onerous accidents. For UK HPR1000, the only fuel assembly structural integrity analysis provided for infrequent fault conditions is for the combination of a LOCA and seismic event. I have sampled this analysis to provide me with confidence in fuel assembly structural integrity in a wider range of conditions, informed by the guidance in Ref. 19.

217. The RP has submitted the analysis method and acceptance criteria to be used in a combined seismic event and LOCA in a methodology document (Ref. 75). This describes the method for the structural integrity fault analysis for the bottom nozzle, top nozzle, guide thimbles and the structural and mixing grids. The purpose of the maximum load criteria is to demonstrate that the fuel assembly will not bow and the impact forces do not result in component buckling. Ref. 75 states that these criteria are derived from the American Society of Mechanical Engineers (ASME) III and AFCEN RCC-M design codes.

218. The fault analysis is a calculation of the impact of the mechanical forces on the fuel assembly's components and a comparison against their acceptance criteria. The analysis code CASAC has been used to carry out these calculations and Framatome has submitted a qualification summary report for the code (Ref. 76). The mechanical forces from the fault are an input into this analysis from CGN.

219. I have not assessed the detail of these acceptance criteria, assessment method or analysis code in GDA. I have observed that these are similar to those ONR has previously assessed, apart from the input conditions derived for the UK HPR1000 plant. I have therefore focused my attention upon the analysis results, sensitivity to the inputs used and any predicted impact on the UK HPR1000 fuel assembly design.

220. The analysis results are presented in the fuel assembly mechanical design report (Ref. 69). The assumed seismic event is a peak ground acceleration of 0.3 g and the assumed LOCA is the bounding break of either the surge, safety injection or residual heat removal lines. This is an Intermediate Break LOCA (IB-LOCA) within the UK HPR1000 safety case. Within Ref. 69 the mechanical forces from the seismic event and IB-LOCA are combined using a quadratic method, which Ref. 75 states is in line with the US NRC methodology.

221. The results of the analysis show significant margins to mechanical acceptance criteria for most of the fuel assembly components. For example, the margins are at least a factor of [redacted] to buckling criteria for the guide tubes and the bottom nozzle. The mid span mixing grids at the edge of the core have the least margin. In most cases these margins are greater than [redacted]% but there is one case at the end of life where the margin is approximately [redacted]%.

222. Due to the limited margin for some locations, I considered potential sources of sensitivity in the analysis. The main sources I identified were the size and speed of pipe break assumed in the LOCA, the magnitude of seismic event assumed, and the way in which the LOCA and seismic loads have been combined.

223. The most conservative pipe break to assume would be a double ended break of the RCS main coolant line pipework, an LB-LOCA. However, the RP has removed this

fault from the design basis (as recorded in the design condition list, Ref. 63) following classification of the pipework as a High Integrity Component (HIC). Fault Studies inspectors have assessed the classification of the fault in their Assessment Report (Ref. 7) and judged that it need not be included in the design basis fault list, using SAP FA.5.

224. I have therefore assessed the analysis of LB-LOCA (in subsection 4.8 of this report) using guidance for faults outside of the design basis. I am satisfied that the assumption of an IB-LOCA (the next most onerous LOCA fault) combined with a seismic event is adequate for fuel system design basis purposes. In response to RQ-UKHPR1000-1761 (Ref. 33), the RP has clarified that breaks of the surge line (hot leg), safety injection line (cold leg) and reactor heat removal line (hot leg) have all been considered as the source of the IB-LOCA, which ONR Fault Studies specialists confirmed to be a conservative approach.

225. The break opening time assumed in Ref. 69 is not stated. This is an important assumption because a shorter break opening time would increase the mechanical loads on the fuel. The RP has argued that its mechanical analysis method follows the US NRC methodology. The conservative requirements from the US NRC's NUREG-609 guidance (Ref. 77) normally expects a break opening time of between 1 and 10 ms. In response to RQ-UKHPR1000-1761, the RP has clarified that the analysis in Ref. 69 assumes a break opening time of 1 ms. As well as being consistent with US NRC guidance, this is also consistent with the assumption made in some previous GDA. Although no justification has been provided specifically for UK HPR1000, I therefore judge that the RP has used an adequately conservative assumption for break opening time in this analysis.

226. The seismic event assumed in the analysis has a peak ground acceleration of 0.3g. ONR's External Hazards inspector confirmed 0.3g to be a conservative acceleration for this purpose.

227. The RP has not explained or justified why the loads on the fuel from the LOCA and seismic event have been combined quadratically rather than additively, other than to state that this is current practice and is based on US NRC guidance. The approach may be adequate if it is not credible for the loads to occur together, in phase with each other when the LOCA is caused by the seismic event. I observe that a similar approach to combining the loads quadratically was applied in a previous GDA.

228. My overall judgement is that the input assumptions discussed above are likely to be conservative but have not been adequately justified by the RP in the UK HPR1000 safety case. Noting the limited margin to load limits for the mid-span mixing grids at the edge of the core, I anticipate that relatively little change in input assumptions could cause predicted local grid buckling. I have therefore considered the potential consequences of local grid buckling in my decision-making. However, given the significant margin for the guide tubes and bottom nozzle, I expect these components are unlikely to be affected. I also note that the UK HPR1000 nuclear design contains no RCCAs at the edge of the core, which means limited buckling of the guide tubes in those locations would not prevent RCCA insertion.

229. Buckling of mixing grid(s) could cause local partial coolant flow blockage. NS-TAST-GD-075 (Ref. 11) suggests that in the case of LB-LOCA, limited crushing of grids in the fuel assemblies at the edge of the core where power density is low may be acceptable for cooling purposes. However, in my opinion these arguments should not be extended to DBA assuming an IB-LOCA (and combined seismic event) which is intended to bound a range of other faults. Ref. 69 states that the methodology used for UK HPR1000 "consists in verifying that the grid dimensional stability conditions are complied" and effectively claims the load limits are all complied with.

230. On balance, I therefore judge that further evidence should be presented to underly the assumptions and methods used in this analysis. However, I have gained sufficient confidence in the conservative nature of key assumptions that I judge it proportionate to allow this work to be completed by the licensee once site-specific ground motions are available. I have raised the following Assessment Finding.

> AF-UKHPR1000-0001 – The licensee shall demonstrate that the combined design basis loss of coolant accident and site-specific seismic event analysed according to the fuel assembly mechanical design basis do not challenge the structural integrity of the fuel assemblies. Justification should be provided for the methods and assumptions used in the analysis.

231. The RP's response to RQ-UKHPR1000-1527 (Ref. 33) states that it would be possible to improve the crush strength of the mid span mixing grids to some degree by [redacted] without an impact on thermal hydraulic performance. This potential improvement should be considered as part of the resolution to Assessment Finding AF-UKHPR1000-0001.

Fuel Assembly Structural Integrity in a Reactor Coolant Pump Overspeed Fault

232. An RCP overspeed has the potential to place excessive hydraulic loads on the fuel assembly hold-down springs. This could potentially lead to the fuel assemblies lifting off the core support plate if the springs fail to fulfil their function. This lift-off has the potential to challenge the fuel assemblies' coolable geometry and may impede RCCA insertion.

233. The RP has submitted analysis of the most onerous loading conditions for the fuel assembly hold-down springs in Ref. 69. The report shows that the analysis has considered an adequate range of operating conditions and burnups to ensure a conservative outcome. Cases are presented for the maximum hydraulic loads in cold, hot and 120% RCP overspeed conditions. In response to RQ-UKHPR1000-1765 (Ref. 33) the RP has provided arguments and evidence that 120% is bounding of the maximum overspeed that could occur during operation. This takes credit for the correct functioning of turbine overspeed protection functions, which are further considered in the Fault Studies Assessment Report (Ref. 7). For the fuel analysis in GDA I am satisfied that 120% overspeed is a conservative assumption.

234. The results of the hold-down analysis show significant margin to the spring's yield stress in all cases. I consider that this is adequate evidence to demonstrate that the fuel assemblies will remain held down in the most onerous conditions. This meets the expectations set by ERC.1 and ERL.1.

4.4.1.6 RCCA Design Expectations and Substantiation

235. RCCAs are part of the shutdown system. SAP ERC.2 advises that reactor shutdown should not be inhibited by mechanical failure, distortion, erosion or corrosion of plant components, or by the physical behaviour of the reactor coolant, under normal operation or design basis fault conditions. SAP EAD.2 is also relevant, stating that adequate margins should exist throughout the life of a facility to allow for the effects of materials ageing and degradation processes on SSCs.

236. In PCSR Chapter 5 (Ref. 3), the RP identify that the safety functional requirements of the RCCA are to provide control of core reactivity and ensure that the nuclear chain reaction can be stopped. To do this, I consider that the design of the RCCA must allow for their insertion in all DBC and DEC-A, allowing for relevant degradation mechanisms such as those listed above.

237. At principle level, I judge that the identified safety functional requirements meet the expectations of EKP.4 and ERC.2. To allow me to judge whether the design meets the expectations set by EAD.2, I have sampled the RP's detailed justification. I have focused my assessment on irradiation-induced swelling of the RCCAs in normal operation and on the integrity of the RCCAs in the bounding fault analysis. Irradiation-induced swelling is an important degradation mechanism for RCCAs containing silver-indium-cadmium material, which can potentially lead to mechanical failure preventing RCCA insertion.

238. The RP has submitted its mechanical justification of the RCCA design in Ref. 79. This includes mechanical component analysis and justification of the absorber rod behaviour in the reactor. Within the assessment of the absorber rod behaviour, Ref. 79 considers normal operation scenarios and fault conditions. I have not sampled the RP's RCCA analysis method or its acceptance criteria in GDA but take some confidence from previous work done by ONR, in which a HARMONI RCCA design was also assessed. Ref. 79 and a separate report submitted by the RP to present OpEx with the HARMONI RCCA (Ref. 80) also demonstrate an extensive amount of OpEx with the HARMONI RCCA product: over 25 years' experience including in the UK. These submissions also describe how feedback from OpEx has been used to help improve the design, addressing the challenge of absorber swelling and creep by reducing the absorber diameter at the bottom of the rod and back-filling the rod with helium to reduce absorber temperature. I consider Ref. 80 to be an example of good practice in using operating feedback to improve the HARMONI design and reduce risk. In response to RQ-UKHPR1000-1386 (Ref. 33), the RP has clarified that no HARMONI model RCCA is known to have failed to insert due to absorber swelling.

239. SSG-52 (Ref. 19) advises that control rods should be replaced or exchanged to limit irradiation-induced swelling (as well as depletion of the absorber material). The frequency at which such replacement should occur is clearly dependent on the time the plant spends operating at low power with RCCAs inserted, which cannot be fully defined in GDA. The RP's RCCA mechanical design and OpEx reports (Ref. 79 and Ref. 80) identify the need for a surveillance scheme on the UK HPR1000 plant, consisting of periodic RCCA inspections and the definition of an appropriate swelling rejection criterion by the licensee. Ref. 79 describes current fuel vendor practice as follows: "the latest surveillance program defines the acceptable maximum absorber rod diameter at the date of the inspection, which takes into account an evaluation of the swelling kinetics in order to guarantee that the RCCA geometry will remain compatible with the guide thimble geometry during the irradiation period till the next RCCA inspection." Ref. 80 also recommends that RCCAs be shuffled in and out of high duty locations. I am satisfied that development of a detailed RCCA surveillance scheme and associated operating rules is an activity for the licensee and therefore the submitted information is adequate in GDA, although in my opinion the requirement of the safety case for such a surveillance scheme could be made clearer. This is not the only shortfall I have identified of this type, so in subsection 4.12 I have listed all those I have identified and recommended a means by which they can be addressed.

240. Overall, informed by the relevant SAPs, I consider that the RP has presented adequate evidence to justify the RCCA design in respect of absorber swelling.

241. The RP has submitted fault analysis for the RCCA in a combined LOCA and seismic event. The analysis considers the lateral deflections and pressure differences that could impair RCCA insertion, including loads due to deformation of the Control Rod Guide Assemblies (CRGA) above the upper core plate and due to differential pressures across the absorber rods. The maximum differential pressure is assumed and according to information provided in response to RQ-UKHPR1000-1386 (Ref. 33), the analysis has assumed a CRGA deflection that is more than double that predicted for UK HPR1000 based on experiments from the CPR1000 fleet. The results show

approximately 25% margin to bending and membrane collapse. Therefore, it seems highly likely that in a bounding fault, RCCA mechanical deformation will not impair their insertion. Due to the extent of margin presented I have chosen not to sample the full underlying evidence for the CRGA deflection.

242. The RCCA absorber material has a relatively low melting temperature of 790ºC. I have observed that the fault analysis has assessed the potential for absorber melt in the bounding DBC-2 conditions and shows significant margin but does not appear to cover DBC-3 or DBC-4 conditions.

243. I consider that the absence of a bounding design basis fault analysis for RCCA absorber melt is a gap in the safety justification. This component has the lowest melting temperature in the fuel system. With insufficient cooling, there is a potential for this absorber to melt from gamma heating. However, there are several things that mitigate the impact of this gap.

244. Firstly, as argued by the RP in response to RQ-UKHPR1000-1761 (Ref. 33), I am satisfied that there is no risk of the absorber melting prior to or during the reactor trip in a design basis fault. This is because there is insufficient time and heat load to increase the absorber temperature whilst not fully inserted in the core.

245. Secondly, the most likely fault to challenge the absorber melt temperature is a LOCA causing core uncovery. This is because there will be a significant reduction in cooling and an increase in gamma heating because of the loss of water shielding. However, this also causes a significant reduction in moderation, meaning that reactor shutdown requires less neutron absorber. Furthermore, the emergency cooling system will inject highly boronated water during the refilling. Therefore, there is likely to be some tolerance to reduced RCCA bank shutdown reactivity worth in a LOCA causing core uncovery.

246. Thirdly, while the absorber material has a lower melting temperature, the cladding material does not. The RCCA absorber clad’s material has a significantly higher melting temperature than the fuel clad technical acceptance criterion in a LOCA (1204˚C). This indicates that the absorber clad is unlikely to fail in any LOCA that meets the technical acceptance criteria. The absorber material should therefore be retained in place.

247. As a result, I consider this gap to be a minor shortfall. Otherwise, after sampling evidence to justify the RCCA design for absorber swelling and for mechanical deformation in bounding faults, I consider that the RP has provided adequate evidence that the RCCAs will fulfil their safety functional requirements and that the expectations set by SAPs ERC.2 and EAD.2 are met.

4.4.1.7 Compatibility of Fuel Assembly and RCCA Designs

248. As part of my assessment of the fuel system design, I undertook a detailed review of the fuel assembly and RCCA design descriptions (Ref. 44 and Ref. 45) as well as sampling the design substantiation. This review uncovered an apparent difference between (1) the maximum insertion depth of the RCCA absorber rods below the fuel assembly top nozzle adaptor plate and (2) the vertical distance between the fuel assembly top nozzle adaptor plate and the bottom of the active length of the fuel rods. The result of this is that in the UK HPR1000 core, a length of active fuel equal to approximately 118 mm will be exposed below the lower end of the RCCA absorber rods at the bottom of the core when the RCCAs are fully inserted. To my knowledge this is a novel configuration in the UK. I considered that it may concentrate neutron flux at the lower end of the core and make SDM harder to predict due to difficulty in properly representing the geometry in the physics models. Recognising that SAP AV.1 expects theoretical models to adequately represent the facility, I therefore raised a

series of questions through RQ-UKHPR1000-0543 and RQ-UKHPR1000-0613 (Ref. 33) to gain confidence that the issue was understood, properly accounted for in the safety case and did not pose a challenge to the claim that the design reduced risks ALARP.

249. The RP has explained that this aspect of the design is required to avoid interference in all conditions between the bottom of the RCCA absorber rod and the fuel assembly guide tube screwed connection. I am satisfied that this requirement prevents the absorber rods from inserting any deeper in to the fuel assembly. However, in my opinion it would be possible to shorten the active length of the fuel such that it was fully covered when the RCCAs are inserted. That would be in line with some other fuel designs that have a longer unfuelled section at the bottom of the fuel rod. The detriment of shortening the active fuel length in the AFA 3GAA design would be a reduction in the total mass of UO_2 fuel in the core, requiring a re-design of the UK HPR1000 fuel management scheme (Ref. 29). I anticipate that higher average uranium enrichments could then become desirable to achieve similar fuel cycle performance and higher power density could be desirable in order to achieve the same total power output. Such changes could have significant impacts on both power generation economics and other aspects of nuclear safety, as well as moving the UK HPR1000 core design away from that used in the reference HPR1000 plant at Fangchenggang Unit 3.

250. The RP has argued, with supporting evidence, that its nuclear analysis is able to adequately model this aspect of the design and that neutronic predictions, including SDM and rod insertion curves, are not compromised. I have discussed this further as part of my assessment of the COCO code's validity in subsection 4.13. I am satisfied that the evidence is adequate for GDA to satisfy my expectations derived from SAP AV.1 and therefore provide confidence in the analysis of this aspect of the design.

251. The other potential impact of the active fuel being closer to the bottom of the fuel rod when compared to other designs is enhanced irradiation damage of the lower core plate. However, ONR Structural Integrity inspectors, whose assessment is recorded in Ref. 81, have advised me that the integrity of the UK HPR1000 lower core plate is not sensitive to an increase in irradiation damage.

252. On balance, I therefore judge that although the presence of an exposed length of active fuel below the RCCA absorber rods when fully inserted is different from other PWR fuel designs assessed in the UK, it does not compromise the RP's argument that risks have been reduced ALARP.

4.4.2 Strengths

253. Following my assessment of the UK HPR1000 fuel system design I have identified the following strengths:

- At a principle level, I consider that the fuel system design for the UK HPR1000 meets the expectations of EKP.4, ERC.1, ERC.2, EAD.2 and ERL.1.
- The RP has considered the relevant phenomena that can cause fuel failures during normal operations and provided adequate justification for the design in respect of these.
- The RP has submitted extensive relevant OpEx for the fuel assembly and RCCA designs, together with evidence that learning from OpEx has resulted in improvements to the designs.
- Structural mechanical analysis shows significant margins for most of the fuel system's components in limiting faults.

4.4.3 Outcomes

254. Following my assessment of the UK HPR1000 fuel system design I have identified the following outcome:

- I have raised an Assessment Finding requiring the licensee to demonstrate that the combined design basis LOCA and site-specific seismic event analysed according to the fuel assembly mechanical design basis do not challenge the structural integrity of the fuel assemblies, providing justification for the methods and assumptions used.

255. I also observed some minor shortfalls during my assessment of this part of the safety case.

4.4.4 Conclusion

256. Based on the outcome of my assessment, I have concluded that the UK HPR1000 fuel system design and substantiation are adequate for the purposes of GDA. I have sampled the substantiation for key aspects of the design and have raised one Assessment Finding. Otherwise, I am satisfied that the design meets the expectations set by SAPs EKP.4, ERC.1, ERC.2, ERL.1 and EAD.2 and is compliant with RGP in Ref. 17.

4.5 Fuel Design Criteria

4.5.1 Assessment

4.5.1.1 Technical Acceptance Criteria for Fuel in Fault Analysis

257. DBA uses technical acceptance criteria for nuclear fuel to give confidence in the integrity of the fuel's barriers to the release of radioactive material. The criteria are usually specific to the design of fuel and often the fault sequences being considered. As per guidance in NS-TAST-GD-006 (Ref. 13) their purpose is that if fault analysis can show that the criteria are met then there can be confidence that fuel integrity is maintained and the extent to which radiological targets need to be demonstrated can often be significantly reduced.

258. This subsection of my report contains my assessment of the UK HPR1000 technical acceptance criteria for nuclear fuel in DBA and the evidence underlying them. It excludes my assessment of the acceptance criteria for fuel in a LOCA, which is reported separately in subsection 4.8, and of the fuel criteria for both normal operations and faults in dry storage, which are reported in subsection 4.5.1.2.

259. The SAPs that I have used in this part of my assessment are ERC.1, FA.7, SC.4 and AV.3. In summary, I expect that: the technical acceptance criteria are able to provide sufficient confidence that the fuel will continue to meet its fundamental safety function to confine radioactive material; the criteria are set conservatively and therefore enable conservative fault analyses; the range of criteria provided is demonstrably complete; and the criteria are supported by relevant experimental evidence with applicability to the AFA 3GAA fuel design. NS-TAST-GD-075 (Ref. 11) and SSG-52 (Ref. 19) provide guidance to inspectors on specific types of design criteria that should be applied for fuel in fault analysis, which I have used to inform my judgements about the completeness of the RP's criteria.

260. Ref. 13 provides generic guidance about technical acceptance criteria for DBA in the context of SAP FA.7. In accordance with this guidance, I recognise that it can be acceptable to use more relaxed technical acceptance criteria for fuel in infrequent faults than those used in frequent faults. Using the guidance in Ref. 13, I expect that

the technical acceptance criteria used for infrequent faults may allow some limited loss of fuel integrity to occur, but the extent to which this occurs (for example, the number of fuel rods predicted to fail) should be limited and should be quantified in order that a demonstration can be made that the consequences have been minimised and the risks reduced ALARP.

261. The OECD has produced a technical review of fuel safety criteria (Ref. 21). I have used this to inform my technical judgements on the completeness of the RP's set of criteria and the adequacy of its justification for individual criteria.

262. The RP has consolidated in its design condition list and acceptance criteria report (Ref. 63) all of its technical acceptance criteria for fuel used in design basis fault analyses that are presented in PCSR Chapter 12 and its sub-references. However, it has not presented criteria in Ref. 63 for some fuel design limits such as fission gas pressure. This is because the RP does not use these for the DBA presented in PCSR Chapter 12. Instead, this subset of criteria have been analysed for bounding frequent fault cases and reported in the UK HPR1000 fuel rod design report (Ref. 71) which I have assessed as part of my assessment of the fuel's design in subsection 4.4. The RP has also consolidated these fuel design criteria and a summary of their evaluation in to the final revision of PCSR Chapter 5 in order that they are adequately visible to the licensee.

263. The RP's technical acceptance criteria for fuel in Ref. 63 can be summarised as follows:

- a minimum DNBR limit in frequent faults and a maximum proportion of fuel rods allowed to undergo DNB in infrequent faults;
- a maximum fuel temperature to avoid fuel melt in frequent faults and a maximum proportion of fuel rods allowed to suffer limited centreline melt in infrequent faults;
- additional criteria to avoid fuel failures in frequent faults, including due to PCI and Pellet Clad Mechanical Interaction (PCMI);
- a maximum clad temperature criterion to limit radiological release in infrequent faults if the limiting channels undergo DNB;
- additional criteria to avoid or limit radiological release from the fuel in an RCCA ejection accident, including due to PCMI, fine pellet fragmentation and clad burst; and
- specific criteria to avoid or limit radiological release from the fuel in a LOCA, discussed in subsection 4.8.

264. The RP has also submitted a supporting report towards showing there will be no fuel failures in frequent faults (Ref. 67), which consolidates the technical acceptance criteria applied in frequent faults, explains their origin and provides reference to subsidiary reports in which quantitative values are derived for the various criteria.

265. I am satisfied that all relevant phenomena identified in Ref. 11, Ref. 19 and Ref. 21 have been addressed by the RP with technical acceptance criteria either in Ref. 63 or the fuel design documentation and I have identified arguments in the safety case to justify all of these criteria. I am therefore satisfied that the RP's selection of technical acceptance criteria for fuel in faults is complete. My assessment of the adequacy of the quantitative values for the technical acceptance criteria used and the evidence underlying them is contained in the following paragraphs.

DNBR and Fuel Temperature

266. I discuss the DNBR and fuel temperature criteria applied by the RP in frequent faults in detail in subsection 4.6 of this report as part of my thermal hydraulic design assessment. The DNBR criteria are set to ensure that there is at least a 95%

probability that DNB does not occur on the limiting fuel rods at a 95% confidence level. The fuel temperature criterion is set to ensure that the peak fuel pellet temperature remains below the melting temperature. Use of criteria of this type aligns with RGP (Ref. 11, Ref. 19).

267. For infrequent faults, the RP's criteria specify that the amount of fuel rods experiencing DNB must remain lower than 10% and that the fuel pellet melting at the hotspot (the peak axial location in the hottest rod) must remain less than 10% by volume. I am satisfied that these criteria meet my expectations derived above. However, a specific ALARP demonstration is still required for faults in which a limited loss of fuel integrity is predicted. The assessment of whether fault analysis shows that risks have been reduced ALARP for UK HPR1000 is presented in the Fault Studies Assessment Report (Ref. 7). I have also assessed the implications of such work for the UK HPR1000 reactor core design in subsection 4.14 of this report.

Pellet Clad Interaction

268. The PCI phenomenon can lead to fuel failures in the absence of DNB and fuel melt during reactivity faults. In this scenario, failure results from a combination of pellet thermal expansion and the release of chemically aggressive fission products between the pellet and clad, which can give rise to Stress Corrosion Cracking (SCC) of the clad. This can lead to a localised cladding failure below the clad strain limit (which is discussed below). The phenomenon is sometimes referred to as PCI-SCC.

269. The RP's technical acceptance criterion for PCI-SCC in frequent faults is a Strain Energy Density (SED) limit of [redacted] MPa, reported in the PCI technological limit report (Ref. 82). This is derived empirically from reactivity ramp experiments, with irradiated fuel rods, that define a safe SED zone for the clad material. The limit is set below the maximum SED of all the failed rods and the experimental data covers an adequate burnup range to bound the predicted maximum burnup in UK HPR1000 fuel. I have observed that a similar limit was proposed for similar fuel in previous GDA. However, because there is limited M5 clad ramp test data for burnup between [redacted] GWd/Te and no failed specimens, the PCI-SCC limit is conservatively set to the highest SED of an intact fuel sample in that burnup range. This limit is significantly lower than the SED of fuel samples that failed (which were all at burnup in the range [redacted] GWd/Te) and lower than a number of other samples that did not fail at burnups above [redacted] GWd/Te. In my opinion setting the limit in this way is a conservative approach.

270. Informed by SAPs SC.4 and AV.3, I judge that the RP has submitted adequate evidence to justify its SED limit for PCI-SCC analysis with AFA 3GAA fuel. This limit appears to be conservative. Use of a criterion that precludes PCI-SCC fuel failures during reactivity faults aligns with RGP (Ref. 11, Ref. 19). This criterion will ensure that fuel clad failures are avoided in reactivity faults that could fail despite meeting the DNBR, fuel temperature and clad strain criteria. Informed by SAPs ERC.1 and FA.7, I therefore consider the PCI-SCC criterion used to be adequate.

Clad Strain

271. The purpose of a radial clad strain criterion is to ensure that the fuel cladding does not fail due to high stress induced by direct pellet/clad contact and pellet expansion during fast reactivity transients, often termed PCMI. According to the OECD/NEA nuclear fuel safety criteria technical review (Ref. 21), usual practice is to define a clad strain limit to give confidence that fuel clad will not suffer mechanical failures.

272. The RP has listed a clad strain limit as a criterion to demonstrate that there will be no fuel failures in frequent faults (Ref. 67). The RP argues that this limit ensures that the fuel will not fail from PCMI in frequent reactivity faults. A limit of [redacted] clad strain is

presented and analysed in the fuel rod design report (Ref. 71) and supporting methodology report (Ref. 83). There is limited justification of this limit in these reports and I observed it is greater than the traditional 1% clad strain limit (Ref. 21) that was also proposed in response to RQ-UKHPR1000-0055 (Ref. 33) earlier in GDA, at a time when a different fuel design was proposed. As a result, I requested further justification of the [redacted] clad strain limit in RQ-UKHPR1000-1689 (Ref. 33).

273. In the RP's response, it was claimed that the French nuclear safety regulator has agreed to relax the clad strain limit to [redacted] based on new experimental data. This takes credit for M5 clad's resistance to hydriding. To support this statement, a published Framatome paper was attached to the RQ response that explains the basis for the proposed [redacted] strain limit. This paper shows significant margin between the [redacted] clad strain limit and the strain level that caused failure in fuel clad fixed-end expansion-due-to-compression tests. However, the paper does not show that the French nuclear safety regulator has accepted the [redacted] strain limit and I judge that it does not contain clear or sufficient evidence that the experimental data is applicable to the fuel used in UK HPR1000.

274. I therefore judge that the RP's justification of the [redacted] clad strain limit contains a shortfall because complete evidence has not been submitted in GDA to justify a less conservative clad strain limit of [redacted] as opposed to the traditional 1% limit (Ref. 21). In mitigation, the analysis results in the fuel rod design report (Ref. 71) show that there is enough margin in the fuel rod transient analysis for the generic UK HPR1000 design to allow a lower clad strain limit of 1% to be met, were it to be applied. As a result, this shortfall is unlikely to challenge the safety of the design and I consider it to be a minor shortfall in the safety case.

Cladding Burst

275. Fuel failures have been observed to occur because of excessive clad expansion (ballooning). A pressure differential across the clad provides a driving force and at high temperature, a phase change in the clad material can lead to super-plasticity. In these scenarios, the OECD/NEA state-of-the-art report on nuclear fuel behaviour in a LOCA (Ref. 84) suggests that the fuel clad will expand until it exceeds the strain limit and fails.

276. The RP has not proposed a specific technical acceptance criterion to preclude fuel failure by clad burst. It argues in Ref. 67 that if the DNBR limit is met (a criterion for frequent faults) then this will limit the clad temperature to below the level where phase change occurs that could lead to clad burst. I am satisfied that this argument is valid for intact circuit faults. I have also observed that for infrequent faults in which some fuel rods undergo DNB, those fuel rods' cladding is assumed to fail through burst in the radiological consequences ALARP assessment (Ref. 85). This means that the consequences of clad burst are conservatively predicted. A separate technical acceptance criterion to preclude or limit clad burst is therefore not necessary.

277. Additional consideration of clad ballooning is important for analysis of LOCA faults in which high clad temperatures are reached and primary pressure drops substantially; I have discussed this topic in subsection 4.8 of this report.

Peak Clad Temperature

278. In faults where fuel experiences DNB the fuel clad can experience very high temperatures. It is usually assumed that the clad will fail locally and this is the assumption made by the RP in the UK HPR1000 safety case. However, the RP apply a Peak Clad Temperature (PCT) limit of 1482ºC for infrequent faults (excluding LOCA) as an additional technical acceptance criterion. The purpose is to ensure that fuel rods that are assumed to fail from DNB do not release more significant quantities of

radiological material into the coolant or cause a loss of coolable geometry. In practice, this means that the clad should not become brittle or reach the melting temperature.

279. In principle, I consider that a PCT criterion that helps ensure there is limited release from the fuel in low frequency faults meets the expectations of ERC.1 and FA.7, as it will minimise radiological consequences from faults where the protection systems cannot prevent fuel failures. In practice the PCT criterion gives confidence that the fuel's radiological material release is limited to gaseous release from fuel failures and does not increase due to fuel fragmentation, melt or dispersal in to the coolant. This type of limit aligns with that suggested in Ref. 21.

280. The RP has submitted Ref. 86 to justify the applicability of the PCT criterion for the UK HPR1000 fuel design. The report presents evidence based on Framatome's experiments with the fuel's M5 cladding material. These experiments show that the clad oxidation is dependent on time and temperature. They show that the clad will experience extensive oxidisation and become brittle if it exceeds the 1482ºC PCT limit. Therefore, the RP argues that remaining below this limit will give confidence that the fuel clad will not suffer brittle failure from quenching following a return to nucleate boiling post DNB.

281. Informed by SAPs SC.4 and AV.3, I consider that the RP has submitted adequate experimental evidence to justify the PCT limit. The limit bounds the relevant experimental data. However, the RP's justification of UK HPR1000 PCT criteria shows that PCT limit is not the only relevant criterion to avoid brittle failure. I observed that the PCT experiments reported in Ref. 86 show that clad can become embrittled at lower temperatures than the PCT limit if the time at temperature is sufficient. The RP has not included the time aspect in its PCT technical acceptance criteria or fault analysis. The RP showed that only two design basis intact circuit faults are predicted to cause limited DNB, the RCCA ejection (Ref. 41) and RCP Locked Rotor (Ref. 38), and that those transients meet both the time and temperature criteria necessary to prevent brittle clad failure. This is because the duration for which fuel undergoes DNB in both transients is less than the time needed for significant oxidation below the PCT limit. The RP has subsequently updated the design condition list and acceptance criteria report (Ref. 63) to clarify that in the transients for which the PCT limit of 1482ºC is applied, the duration for which DNB occurs is less than [redacted]. I am satisfied that this adequately qualifies the PCT criterion for GDA.

282. The UK HPR1000 PCT technical acceptance criterion is based upon relevant experimentation. Meeting this criterion in faults where fuel experiences DNB gives confidence that the fuel clad will not melt and that the fuel clad will not become brittle. I have observed that the technical acceptance criteria do not account for additional oxidation from clad burst. However, I consider that this is a minor shortfall because the fault analysis shows there is margin to temperatures where significant oxidation could occur. As a result, informed by the relevant SAPs I consider that this technical acceptance criteria and its justification to be adequate.

RCCA Ejection Limits

283. The RCCA ejection transient is an infrequent (DBC-4) fault that causes a short-term significant increase in reactivity. This increased reactivity rapidly increases the power and temperature in the fuel pellets in rods near to the ejected RCCA. These effects can cause fuel to fragment, clad to fail through PCMI, clad to become embrittled and fuel fragments to disperse in the coolant. Ultimately, these phenomena can lead to significant releases of radiological material from the fuel into the coolant.

284. The RP presented specific technical acceptance criteria for this fault in Ref. 63 with supporting arguments and evidence in Ref. 87. These criteria are limits on Radially

Averaged Fuel Pellet Enthalpy (RAFPE) and the PCT, fuel temperature and DNBR criteria discussed above for infrequent faults.

285. Use of a RAFPE limit for RCCA ejection faults is consistent with advice contained in Ref. 11 and Ref. 19. Using this advice, I expect the limit to be set to prevent or minimise the potential for PCMI fuel failures as well as prevent a loss of core coolability due to fuel fragmentation and dispersal. In principle I am satisfied that an adequately justified RAFPE limit, when combined with the other criteria identified above, can meet my expectations derived from SAPs ERC.1 and FA.7.

286. Ref. 87 explains that the objectives of the RP's RAFPE limits are to reduce the risks from RCCA ejection faults undermining core coolability and causing fuel fragmentation. It presents the results from RCCA ejection simulation experiments as evidence to support the applicability of these criteria for the UK HPR1000 fuel design. There are two different RAFPE limits in Ref. 87, applied to fuel at different burnup levels:

- For fuel up to [redacted] GWd/Te, a limit of [redacted] on peak RAFPE; and
- For fuel between [redacted] GWd/Te and [redacted] GWd/Te, a limit of [redacted] on the increase in RAFPE during the transient.

287. The RP has presented the results of experiments to support the limit on peak RAPFE of [redacted]. This data shows that this peak RAFPE limit is bounding of the majority of the fuel fragmentation experiments. The RP argues that the fuel samples that failed below this limit were either water-logged or did not experience significant fuel loss. Therefore, it argues that there is sufficient experimental evidence to show that the fuel will not fragment below this peak RAFPE limit. I consider that the RP has presented adequate evidence to demonstrate this.

288. I observed that this peak RAFPE limit is not intended to demonstrate that the fuel clad will not fail due to PCMI. Experiments have shown that fuel will fail due to PCMI at lower RAFPE than it will fragment and that this can occur at relatively low burnup (Ref. 21). I therefore judge that the RP's RAFPE limit in this burnup range (up to 33 GWd/Te) does not meet my expectation that fuel failures through PCMI be minimised or prevented.

289. The RP's limit on the increase in RAFPE during the transient for fuel between [redacted] GWd/Te and [redacted] GWd/Te is intended to prevent PCMI fuel failures (and therefore also fuel fragmentation and dispersal). It is based on experimental data, but I observed that Ref. 87 only presents three data points for M5 clad, showing that the fuel survived a RAFPE increase up to [redacted]. To justify the limit of [redacted], additional data from VVER fuel with E110 clad material is used. A significant proportion of other available experimental data has been excluded by the RP in deriving the presented limit, but not all of these exclusions are justified in Ref. 87.

290. Informed by SAP AV.3, I expect that the limits of applicability of the available experimental data should be identified and that extrapolation beyond these limits should not be used unless justified. I am therefore not satisfied with the RP's justification for its limit on the increase in RAFPE during the transient for fuel between [redacted] GWd/Te and [redacted] GWd/Te.

291. I have been able to compare both the low and high burnup limits presented by the RP in Ref. 87 with RAFPE limits that were justified in previous GDA and with equivalent limits prescribed by the NRC for RCCA ejection faults in Regulatory Guide 1.236 (Ref. 88), which were developed using the same range of experimental data. Using this information, I judge that any necessary reduction in the RP's limits to achieve an adequate safety case should be relatively modest. I have also observed that the most onerous case in the RP's fault analysis for the RCCA ejection (Ref. 39) shows safety

margin to RAFPE limits from other sources as well as the limits currently applied by the RP.

292. I am therefore able to judge that the shortfalls I have identified in Ref. 87 do not present a risk to the UK HPR1000 design in GDA. However, it is important that limits are adequately justified by the licensee to support the operational safety case and ensure safety margins are properly understood. I also recognise that fuel behaviour following RCCA ejection is a subject of active experimental research, for example through the CABRI programme (Ref. 89), the likes of which should be reviewed by the licensee to determine any relevant emerging information. As a result, I have raised the following Assessment Finding.

AF-UKHPR1000-0002 – The licensee shall justify technical acceptance criteria to prevent fuel fragmentation and minimise pellet-clad mechanical interaction induced fuel failures at all fuel burnups in a Rod Cluster Control Assembly ejection fault. The selection of underpinning experimental data should be justified.

Summary of Assessment for Fuel Technical Acceptance Criteria in Fault Analysis

293. I am satisfied that the technical acceptance criteria used in frequent faults provide sufficient confidence that the fuel will continue to meet its fundamental safety function to confine radioactive material.

294. I am satisfied that the technical acceptance criteria used in infrequent faults provide sufficient confidence that radiological release from the fuel will be limited and quantified in a way that will allow a demonstration to be made that consequences are minimised and risks reduced ALARP.

295. I have identified minor shortfalls associated with the clad strain and PCT criteria and have raised an Assessment Finding associated with the RCCA ejection criteria. I am satisfied that once this Assessment Finding is addressed the technical assessment criteria will be supported by sufficient experimental evidence with applicability to the AFA 3GAA fuel design and will enable conservative fault analyses.

4.5.1.2 Fuel Integrity Criteria for Spent Fuel Interim Storage

Background and Expectations

296. The RP has submitted documents outlining a concept design for a future Spent Fuel Interim Storage (SFIS) facility for UK HPR1000. The intention is that spent fuel will leave the spent fuel pool and be stored in the SFIS facility until a future Geological Disposal Facility is available.

297. The concept design presented by the RP (Ref. 90) is to store the fuel dry in welded canisters, inside a dedicated SFIS facility with an assumed design life of 100 years. The fuel will be removed from the spent fuel pool, dried and loaded in to canisters in the fuel building, before transit to the SFIS facility. A preliminary safety evaluation for the concept SFIS design has been submitted in Ref. 91.

298. Prior to Step 3 of GDA, ONR and the EA provided clarification to the RP via a letter (Ref. 92) on the regulatory expectations for the concept design of the SFIS facility and the maturity of the safety case in GDA, based upon what the regulators considered to be necessary to undertake a meaningful assessment.

299. The assessment of the SFIS facility design and safety case in GDA, including the overall demonstration that risks are reduced ALARP, is reported by ONR's Nuclear Liability Regulations inspector in their Assessment Report (Ref. 93), with support from

other specialists including Fault Studies inspectors. The scope of my own assessment is limited to the fuel criteria that the RP has specified to demonstrate that fuel integrity will be maintained in normal operation and faults after fuel leaves the spent fuel pool, during processing, transit and storage in SFIS. For brevity, in the remainder of this subsection I refer to all of these activities as 'SFIS operations'.

300. Based on Ref. 92, I do not expect the fuel criteria for SFIS operations to be fully detailed in every respect or to be finalised in GDA. However, I do expect the key criteria to be identified with suitable underlying evidence. I also expect there to be confidence that any likely future change in the criteria can be accommodated by the generic design without significant change. The main purpose of my assessment of this topic was to gain confidence that is the case for UK HPR1000.

301. The SAPs most relevant to my assessment of this topic are ERC.1, ENM.6 and EAD.2. In the case of design basis fault conditions, FA.7 is also relevant because the setting of appropriate criteria is necessary to predict when fuel failures will occur and therefore allow a conservative demonstration that the consequences of faults are reduced ALARP. NS-TAST-GD-075 (Ref. 11) and NS-TAST-GD-081 (Ref. 15) also provide detailed advice to inspectors for assessment of whether passive safety has been achieved and on the aspects of fuel and core design that need to account for the planned means of spent fuel storage. Guided by these sources of RGP, my expectations for this topic are:

- the numerical values of safety-related parameters at which physical barriers to release (in this case the fuel clad) are challenged should be specified;
- operating limits should be set, with suitable and sufficient margins, to retain the integrity of the cladding as a barrier to fission product release during SFIS operations;
- the limits should be set with consideration of all relevant degradation mechanisms, including clad corrosion, clad creep and potential embrittlement due to hydride reorientation; and
- where different limits are used in normal SFIS operations and design basis fault analyses, the differences should be explained and justified, with reference to the values of these parameters at which clad integrity will be challenged.

302. The only fuel criteria relevant to SFIS operations that I have not considered during GDA are mechanical load limits. The RP has submitted mechanical load limits for fuel handling purposes in its report on fuel failure mechanisms in the fuel route (Ref. 66). These are a function of the fuel design, which has been operated and handled at many PWRs internationally and is in the same family as that to be used in the EPR in the UK. I chose to focus my assessment on the other criteria necessary for dry storage, which are the result of complex phenomena and have been the subject of recent research for M5 clad.

303. The RP's preliminary safety evaluation (Ref. 91) states fuel design criteria for use in SFIS operations safety evaluations, which appear to be based on US NRC recommendations for fuel with Zircaloy-4 clad. These include limits on clad temperature of [redacted] in normal conditions and [redacted] in fault conditions, with an additional limit on time-at-temperature to be defined for fault conditions in the detailed design phase post-GDA. The response to RQ-UKHPR1000-1474 (Ref. 33) states that Ref. 91 also includes a clad hoop stress criterion of [redacted], but this is not stated anywhere in the formal submission. As well as the limits in Ref. 91, the SFIS design report (Ref. 90) identifies the need for the fuel not to exceed a hydrogen content limit (to be set by the fuel supplier) and suggests that an option exists to replace the [redacted] normal operation limit in detailed design (for the purposes of limiting clad creep) with a [redacted] strain criterion, a slightly wider temperature range and a [redacted] hoop stress limit.

304. The RP has also submitted the Framatome document Long Term Storage of Spent Fuel - Design Criteria (Ref. 66) to present and substantiate fuel design criteria specifically for the AFA 3GAA fuel design with M5 clad, for both wet and dry spent fuel storage. This report identifies the potential degradation mechanisms that can affect fuel integrity during dry storage. The most important are identified to be (1) clad creep, (2) hydrogen effects including the potential for hydride reorientation to lead to clad embrittlement and (3) clad corrosion. I am satisfied that the relevant degradation mechanisms in dry storage have been identified.

Corrosion and Creep in Normal SFIS Operations

305. Ref. 66 presents arguments to provide confidence that significant corrosion can be precluded in storage by control of the chemical environment. Detailed chemical and environmental specifications will need to be developed to support the detailed design phase of the UK HPR1000 SFIS facility post GDA, but I consider this to be normal business for the licensee, so the arguments presented are adequate for GDA.

306. Ref. 66 provides evidence supporting the application of a [redacted] creep strain failure criterion for M5 clad in a temperature range bounding the [redacted] limit stated in Ref. 91 and at clad hoop stress up to approximately [redacted]. Furthermore, it provides some evidence that at temperatures bounding that limit and hoop stress bounding the limits that need to be imposed due to considerations of hydride reorientation (discussed below), the creep strain limit will not be approached over a long time period. I am therefore satisfied with the adequacy of the clad temperature limit imposed in Ref. 91 for the purposes of preventing clad failure due to creep in normal SFIS operations. I consider the lack of a stated limit on creep strain and/or clad hoop stress in Ref. 91 to be a shortfall, but which is bounded by the considerations on hydride reorientation discussed below.

Clad Embrittlement due to Hydride Reorientation in Normal SFIS Operations

307. Ref. 66 identifies that hydrides can form within the fuel cladding material during power operation due to cladding corrosion and can significantly impact the mechanical properties of the cladding. It explains that during fuel transfer to dry storage all the hydrides in M5 clad will be dissolved in the temperature range approximately [redacted] and that a fraction of the dissolved hydrogen might reprecipitate in the form of radially oriented hydrides in clad with hoop stress above [redacted]. However, Ref. 66 references the results of long term creep tests at [redacted] that "have emphasized a significant propensity of $M5_{Framatome}$ to maintain high ductility thanks to irradiation damage recovery" and concludes that the embrittlement of irradiated M5 cladding due to hydride reorientation appears unlikely. I observed that these tests were undertaken with clad hydrogen concentrations below the limiting values which might be expected to occur in UK HPR1000, which in my opinion limits the confidence that can be taken from them.

308. No limits on clad temperature, hoop stress or hydrogen concentration are explicitly proposed to prevent embrittlement due to hydride reorientation in Ref. 66. However, following my interactions with the RP during GDA, a brief summary has been provided in Ref. 66 of additional hydride reorientation tests that have been undertaken, with a hydrogen content bounding that which is expected for UK HPR1000 and a hoop stress up to [redacted]. The results are stated to show that the clad retained very good ductility after cooling from those conditions, but the underlying evidence has not been provided by Framatome due to its proprietary nature.

309. Overall I judge that the criteria proposed by the RP to preclude embrittlement of M5 clad material due to hydride reorientation in normal SFIS operations are not yet substantiated by the available evidence.

Limits in Faults Occurring During SFIS Operations

310. For faults during SFIS operations, a higher clad temperature criterion of [redacted] has been set by the RP in Ref. 91. The [redacted]% strain criterion that has been shown to prevent creep failure of M5 clad is only, according to the evidence provided in Ref. 66, valid in a temperature range up to [redacted]. Ref. 66 states that at temperatures in the range [redacted], the time at temperature is also important to preclude clad failure due to creep and the most appropriate criteria to set may be either in the form of a strain criterion or in the form of criteria on stress and the rate of temperature rise. Some evidence is provided from creep tests to indicate the approximate time to clad failure at temperatures of [redacted], but this is also dependent on fuel rod internal pressure. Within Ref. 91, the RP has recognised that its proposed [redacted] clad temperature limit in faults must have an associated time limit and state that this will be determined during detailed design.

311. In my opinion, therefore, the limits proposed by the RP in Ref. 91 to preclude clad failure in faults during SFIS operations are not yet complete, and the temperature limit that exists is not yet substantiated by the available evidence.

Versatility of the Generic Design

312. As I am not yet fully satisfied by the evidence underlying the proposed fuel integrity criteria in SFIS operations, I have considered the likelihood that any future change in the criteria necessary to reduce risks ALARP can be accommodated by the UK HPR1000 generic design without significant change.

313. In my opinion the aspects of the generic design that will most strongly influence the behaviour of the fuel entering SFIS, particularly the phenomena discussed in the preceding paragraphs, are fuel decay heat, fission gas pressure, clad oxidation and hydrogen uptake, and the available cooling time before transfer to SFIS. The first three parameters are mainly driven by fuel enrichment, discharge burnup and clad material properties, while the fourth is driven by the fuel management design and capacity of the spent fuel pool.

314. With respect to the clad material, I recognise that OpEx shows evidence of relatively low oxidation and hydrogen uptake in M5 clad as a function of burnup (see sub-section 4.4.1.4).

315. To inform my judgement about the versatility of the rest of the UK HPR1000 generic design, I have compared the fuel enrichment, limiting discharge burnup and spent fuel pool capacity with those of another nuclear power plant. This other design uses similar fuel with the same clad material and includes a similar dry fuel storage design to the UK HPR1000 SFIS concept.

316. The peak fuel rod average burnup predicted for UK HPR1000 at the end of the equilibrium cycle (Ref. 29) is 54.8 GWd/Te. This is lower than the maximum fuel rod burnup predicted for the other nuclear power plant. The maximum fuel enrichment proposed in UK HPR1000 is also lower than at the other nuclear power plant.

317. The UK HPR1000 spent fuel pool capacity as reported in the Fuel Handling and Storage System Design Manual (Ref. 94) is sufficient to allow a cooling time for UK HPR1000 fuel of at least 10 refuelling cycles, or around 15 years. I understand this to be longer than the cooling time that will be available for fuel at the other nuclear power plant. The RP claims in response to RQ-UKHPR1000-1474 (Ref. 33) that by investigating international experience, it has determined that the minimum cooling time before entry to SFIS is usually in the range of 3-7 years, which imply that the UK HPR1000 spent fuel pool capacity provides substantial margin.

318. The relatively low discharge burnup, fuel enrichment and longer available cooling time should mean that UK HPR1000 fuel is able, if necessary, to enter SFIS operations with relatively low decay heat, fission gas pressure and hydrogen concentration. I am therefore satisfied that the UK HPR1000 generic design is sufficiently versatile to accommodate changes in the proposed fuel criteria for SFIS operations, if necessary to reduce risk to ALARP during SFIS detailed design.

Summary of Assessment for SFIS Fuel Criteria

319. Overall, I judge that the RP's quantification of key limits to prevent fuel failures in SFIS operations and its provision of evidence to support these limits are incomplete. The RP has identified the correct parameters that need to be controlled, but limits have not been specified for some of those parameters and the evidence needed to substantiate the currently proposed limits has not all been provided.

320. However, I judge that the generic design of UK HPR1000 is versatile enough to accommodate further development of fuel criteria for entry to SFIS operations in the detailed design stage. The risk posed to the generic design by the incomplete set of fuel criteria and underlying evidence is low. To ensure that this matter is fully addressed by the licensee before detailed design of the fuel building and finalisation of the fuel management design are complete, I have raised the following Assessment Finding.

AF-UKHPR1000-0003 – The licensee shall, as part of detailed design and before the fuel management design is finalised, demonstrate that the fuel criteria required to ensure clad integrity during spent fuel interim storage operations can be met for limiting fuel rods, from the point of fuel leaving the UK HPR1000 spent fuel pool. Evidence should be provided to substantiate the selected criteria.

321. I note that a related Assessment Finding has been raised by ONR's Nuclear Liability Regulations inspector in Ref. 93, focusing on the adequacy of equipment in the fuel building to ensure that fuel does not overheat during export to SFIS.

4.5.2 Strengths

322. Following my assessment of the UK HPR1000 fuel technical acceptance criteria I have identified the following strengths:

- The RP has submitted a comprehensive set of fuel technical acceptance criteria that address all phenomena that can cause fuel failures in reactor faults.
- The technical acceptance criteria for frequent reactor faults meet my expectations informed by SAPs ERC.1 and FA.7. I am satisfied that if these criteria are met in faults then the fault should not result in a release of radiological material from fuel.
- The UK HPR1000 design in GDA is sufficiently versatile to accommodate changes in fuel criteria for SFIS operations, due to its moderate fuel discharge burnup and the capacity specified within the spent fuel pool.

4.5.3 Outcomes

323. Following my assessment of the UK HPR1000 fuel technical acceptance criteria I have identified the following outcomes:

- I have raised an Assessment Finding to ensure the licensee justifies technical acceptance criteria to prevent fuel fragmentation and minimise PCMI-induced fuel failures at all fuel burnups in an RCCA ejection fault. The selection of underpinning experimental data should be justified.

- I have raised an Assessment Finding to ensure the licensee demonstrates that the fuel criteria required to ensure clad integrity during SFIS operations can be met for limiting fuel rods, from the point of fuel leaving the UK HPR1000 spent fuel pool. Evidence should be provided to substantiate the selected criteria.

324. I also observed some minor shortfalls during my assessment of this part of the safety case.

4.5.4 Conclusion

325. Based on the outcome of my assessment, considering SAPs ERC.1, EAD.2, ENM.6 and FA.7, I have concluded that the UK HPR1000 fuel technical acceptance criteria meet my expectations adequately in GDA. They will maintain the integrity of fuel clad or, should clad integrity be challenged, retain the majority of radiological material in the fuel. Subject to the licensee adequately resolving the Assessment Findings I have identified, I am satisfied that the criteria will help to ensure that radiological consequences from faults are predicted conservatively and allow a demonstration that the consequences are ALARP.

4.6 Thermal Hydraulic Design and Criteria

4.6.1 Assessment

326. Thermal performance analysis is required to demonstrate that the fundamental safety function to provide control of heat removal from the core is delivered with an appropriate degree of confidence, in accordance with SAP ERC.1.

327. To ensure adequate heat removal from the core, I expect to see thermal hydraulic design criteria defined that include limits on fuel temperature, DNBR or alternative criteria to avoid reaching Critical Heat Flux (CHF) and, informed by SAP ERC.3, criteria to avoid hydrodynamic instability. The occurrence of localised DNB or fuel melt may not always lead directly to clad failure. However, the use of fuel temperature and DNBR criteria allow fuel failure to be precluded without the need to model the complex thermo-mechanical phenomena that are like to cause clad failure if DNB or/and fuel melt occur. The purpose of avoiding hydrodynamic instability is to ensure the validity of the other thermal hydraulic calculations undertaken. Instability can lead to oscillating coolant flow in the core, which most thermal hydraulic performance codes (including those used for UK HPR1000) are not validated to model.

328. My assessment of the DNBR and fuel temperature criteria in this sub-section builds on that of sub-section 4.5, examining the quantification of the criteria and underlying evidence in greater depth. Additional thermal fuel criteria are specified for some specific fault types; I have assessed the validity of these for intact circuit faults in subsection 4.5 and for LOCAs in subsection 4.8.

329. Informed by SAP AV.3, I expect the thermal design criteria applied to be justified for the UK HPR1000 fuel design and relevant operating conditions, generally with reference to test data. Where relevant, in accordance with SAP EAD.2 I expect degradation of the fuel and plant to be accounted for (for example, corrosion of fuel clad surfaces, or bowing of fuel rods and assemblies). SAP ERC.1 guidance states that there should be suitable and sufficient margins between the normal operational values of safety-related parameters and the values at which the physical barriers to release of fission products are challenged. I expect thermal hydraulic design criteria to be defined with this in mind.

330. The core thermal hydraulic analysis is integrated with whole-plant analysis for the purposes of DBA, so I consider SAP FA.7 to be important for my assessment, which identifies the need for a conservative approach for design basis fault sequences. The

assessment of the UK HPR1000 Fault Studies analysis itself is reported in Ref. 7. In general, I expect that fuel failures predicted by analysis due to a breach of any of the criteria identified above should be avoided in all normal conditions and frequent faults and should be reduced ALARP for infrequent faults. These expectations are informed by NS-TAST-GD-075 (Ref. 11), which provides advice on the nature of RGP to meet expectations for defence-in-depth set by EKP.3.

331. Relevant parts of the RP's safety case are summarised within the PCSR (Ref. 3) and the Thermal Hydraulic Design report (Ref. 95). In order to satisfy thermal hydraulic safety functional requirements, these reports define design basis requirements for DNBR, fuel temperature, core flow (accounting for flow bypasses within the RPV) and hydrodynamic stability.

332. DNBR correlations and limits have been derived by the RP to predict when DNB will occur as a function of local conditions in the core including power, coolant mass flux and enthalpy. As I observed in sub-section 4.5, the correlations have been derived statistically using experimental data, to provide at least a 95% probability that DNB will not occur on the limiting fuel rods at a 95% confidence level.

333. Ref. 95 is well-structured with most of the 'design evaluation' sub-chapters corresponding to a topic that is detailed further in supporting references. I have satisfied myself that important aspects of core thermal hydraulic design have been addressed by the RP, through a comparison against guidance in Ref. 11 and SSG-52 (Ref. 19). I have sampled the following parts of the safety case in more depth, either because they are particularly important to reduction of risk or because additional information was required to form a judgment on the argument being made in the safety case:

- The DNBR design limits
- The fuel temperature limit
- Hydrodynamic instability predictions
- Allowances made for through-life degradation mechanisms
- The adequacy of reactor trip settings
- Bypass flow and pressure drop predictions

334. Thermal performance has been predicted for UK HPR1000 using a sub-channel computer code called LINDEN, for which the RP has submitted a set of supporting validation evidence. My assessment of the validity of LINDEN for application in UK HPR1000 sub-channel analysis is described in subsection 4.13 of this report.

4.6.1.1 DNBR Design Limits

335. The RP has submitted a report, DNBR Design Limit (Ref. 96), to provide the DNBR design limits to be applied for UK HPR1000 fault analysis with two different CHF correlations. Ref. 96 includes information about the limits' derivation and reference to supporting evidence. The two CHF correlations used are the FC2000 (a proprietary Framatome correlation) and W3 (a correlation originally developed by Westinghouse but now in the public domain). The test sections used to derive the FC2000 correlation incorporated mixing grids and the RP has therefore limited its application of the FC2000 correlation to predict DNBR downstream of the first mixing grid in UK HPR1000 fuel assemblies. The RP has applied the W3 correlation to predict DNBR upstream of the first mixing grid in UK HPR1000 fuel assemblies.

336. Both correlations are established within the nuclear industry and have undergone significant regulatory attention in the past, so I have not chosen to assess the evidence behind their derivation in detail. However, I have sampled elements of the correlation reports submitted by the RP (Ref. 97 for FC2000 and Ref. 98 for W3) to satisfy myself that the design limits calculated for use with the correlations specifically for UK

HPR1000 are adequately justified. The following paragraphs present my assessment of Ref. 96, Ref. 97 and Ref. 98.

DNBR Limits used with the FC2000 CHF Correlation

337. Ref. 97 shows that the test database used to validate the FC2000 CHF correlation is extensive, uses tests that are adequately representative of the UK HPR1000 fuel design and is treated with robust statistical procedures to determine a correlation limit.

338. Ref. 96 describes two different approaches to application of the FC2000 correlation within the UK HPR1000 safety case, for which different DNBR limits are derived. The first is a 'statistical thermal design procedure' in which the uncertainties associated with plant operating parameters are convolved with the correlation uncertainties derived from the test programme in Ref. 97. This approach is described thoroughly in Ref. 96. I am satisfied that the calculation procedure and uncertainties applied are consistent with international practice. The second approach is a 'deterministic' procedure in which the uncertainties derived from the test programme in Ref. 97 are still treated statistically but plant operating parameters are treated conservatively by specifying bounding inputs to DNBR calculations. This approach is used when plant parameters predicted by fault transient analysis are outside the range assumed by the statistical thermal design procedure (for example, when plant pressure is particularly low). Again, I am satisfied that the calculation procedure used for this second approach is consistent with international practice. Furthermore, the deterministic correlation limit derived in Ref. 97 has had an additional [redacted]% margin added to it by Framatome, which will lead to more conservative fault analysis results.

339. Before application in the UK HPR1000 safety case, the RP has applied two additional factors to both the DNBR limits derived for use with the statistical and deterministic procedures with the FC2000 correlation. These are a factor to account for the effect of fuel rod bow on DNBR and an additional flat margin of [redacted]%. In principle this [redacted]% margin should provide for a very conservative analysis. However, through my assessment I have uncovered two additional sources of uncertainty in the analysis which are not explicitly catered for in Ref. 96. The first of these is the phenomenon of fuel assembly bow.

340. The phenomena of fuel rod bow and fuel assembly bow are distinct. Fuel rod bow can occur due to differential expansion of cladding material on different sides of a fuel rod when there is a temperature gradient across it due to local power variations in the core. Fuel assembly bow can occur due to irradiation creep of the guide tube material, under stress in normal operation from both the assembly hold-down force and hydraulic loads. I discuss the margins available to account for both phenomena in the RP's DNBR analysis in sub-section 4.6.1.4.

341. The second source of uncertainty not explicitly catered for in Ref. 96 is the use of more than one sub-channel code in the safety case. The FC2000 CHF correlation is applied in UK HPR1000 fault analysis calculations together with the LINDEN code. FC2000 is a 'local conditions' correlation, meaning that test data is used to correlate the CHF prediction against the local conditions in the channel at the location of CHF. A sub-channel code is used to predict what local conditions at the location of CHF actually were, given the known test conditions at the time. If the same sub-channel code is later used with the CHF correlation in the fault analysis, then any error introduced in the CHF correlation by the calculation of local conditions in the test should effectively be cancelled out. However, for UK HPR1000 Ref. 97 states that the FLICA-3F code, a different sub-channel analysis code from LINDEN, has been used to predict local conditions in the test section for the purposes of FC2000 correlation development and validation. This introduces an additional error in the CHF correlation for application with

LINDEN, due to potential differences in predictions between FLICA-3F and LINDEN of the local conditions at the point of CHF in the test.

342. The RP clarified during Step 4 that the [redacted]% margin added to both FC2000 design limits is intended to allow [redacted]% margin to bound the uncertainty due to the use of two different codes, with [redacted]% remaining as a future design provision. I considered the incorporation of this design provision to be good practice but judged that further evidence was necessary to demonstrate that the total margin was sufficient to bound the uncertainties I observed.

343. Based on the information available in the LINDEN validation report (Ref. 99) and following consultation with my TSC to provide independent advice (reported in Ref. 100 and also discussed in subsection 4.13.1.6) my judgment is that a [redacted]% margin is highly likely to be sufficient to cater for the uncertainty introduced due to the use of these two different sub-channel codes. I have gained further confidence in this judgement through a meeting with CGN [redacted]. For the purposes of GDA I am therefore satisfied that this issue presents little risk to the UK HPR1000 design or fault analysis. However, [redacted] are not part of the UK HPR1000 safety case and the exact nature of the uncertainty introduced by this issue is unknown, so I consider this to be a potential shortfall against SAP AV.3. I have raised the following Assessment Finding to ensure that the licensee addresses this topic:

AF-UKHPR1000-0004 – The licensee shall validate the critical heat flux correlation limits and statistical parameters for the chosen critical heat flux correlation(s) in the sub-channel analysis code used to undertake fault analysis calculations. This should include an analysis of the underlying experimental data using that same code.

Turbulent Diffusion Coefficient

344. The RP has submitted evidence in its Turbulence Diffusion Coefficients report (Ref. 101) to underly the selection of turbulent diffusion coefficient for application in UK HPR1000 sub-channel analyses. This coefficient is used to vary the amount of turbulent mixing between channels in the sub-channel code and can impact DNBR predictions. The RP's application of the experimental evidence in Ref. 101 suffers from the same shortfall as that discussed in paragraph 341, associated with the use of two different sub-channel codes.

345. However, the coefficient value recommended by Ref. 101 for use in fault analysis is the minimum from all presented experiments, was from a 14-foot fuel assembly and is significantly lower than values derived from the experiments on 12-foot fuel assemblies like the AFA 3GAA. The value recommended for (and used in) CHF correlation development with FLICA-3F is also conservative, being the maximum of all the presented experiments. I judge that the conservatism inherent in the coefficients derived with FLICA-3F is likely to be sufficient to counteract the effect of the difference between codes. I have also obtained sensitivity analysis through RQ-UKHPR1000-0835 (Ref. 33) to show that the impact of changes in turbulent diffusion coefficient on calculated hot channel fluid properties using LINDEN is relatively small. Accounting for all these factors, I consider the use of different codes to determine and apply turbulent diffusion coefficients to be only a minor shortfall.

DNBR Limits used with the W3 CHF Correlation

346. The W3 CHF correlation is being applied in the UK HPR1000 fault analysis only to predict DNBR upstream of the first mixing grid, due to the limits on applicability of the FC2000 correlation described in paragraph 337. The design limits presented for use with the W3 correlation in Ref. 98 are [redacted] in the pressure range [redacted] MPa, or

[redacted] in the pressure range [redacted] MPa, with an additional margin of [redacted] then applied to both design limits by the RP as a rod bow penalty. These limits (excluding the rod bow penalty, which I have discussed in 4.6.1.4) are widely used in industry rather than having been derived specifically for UK HPR1000. Ref. 98 presents the results of three test programmes intended to demonstrate that use of the correlation with these limits is adequate for the AFA 3GAA fuel used in UK HPR1000. However, for just over 10% of the relevant data points, the results show that DNB occurred in the test at lower heat flux than was predicted by the W3 correlation with the selected design limit. These outlying data points are justified by the RP on the basis that the majority occurred with a combination of low mass velocity, low pressure and either slightly negative or positive quality, conditions which the RP argues will not occur upstream of the first mixing grid in the fuel assembly. In this context, 'quality' is a measure of the difference between the mixture enthalpy and the saturated liquid enthalpy, indicating how sub-cooled or super-heated the mixture is.

347. To obtain evidence on this point I reviewed the full range of design basis transient analysis reports submitted for UK HPR1000 in PCSR Chapter 12 to determine where the W3 correlation had been applied in the safety case. I determined that the only relevant faults are an RCCA bank withdrawal from low power/shutdown (Ref. 53) and a steam system piping break from zero power (Ref. 43). For the RCCA bank withdrawal fault, plant pressure is in the normal range so I am satisfied that the RP's justification of the W3 correlation is adequate. For the steam system piping break fault this is not the case, and the RP has submitted a dedicated report (Ref. 102) to justify the use of the W3 correlation for the relevant plant conditions in this specific fault. I have assessed this in conjunction with the Fault Studies inspector, whose holistic assessment of the steam system piping break fault is captured in Ref. 7.

348. Ref. 102 uses a subset of the experiments reported in Ref. 98 to present a linear extrapolation of the Measured-to-Predicted (M/P) CHF ratio as a function of pressure, using data points measured between [redacted] MPa and [redacted] MPa to justify the correlation limit for application at a pressure below [redacted] MPa. The RP claims there is margin between the M/P that the extrapolation predicts at that pressure and the M/P that is required to ensure the proposed design limit is adequate. However, I am not fully satisfied that this extrapolation is valid and moreover, even if a linear extrapolation is used, the variance of the M/P results in Ref. 102 appears in my opinion to be large enough that the low pressure correlation limit without rod bow of [redacted] may not be high enough to give a 95% probability that DNB will not occur at a 95% confidence level. I therefore judge the experimental data shows that the W3 DNBR design limit recommended at low pressures could be inadequate to provide a conservative prediction of DNBR to a high confidence level in UK HPR1000 design basis fault analyses.

349. However, I recognise that the main steam line pipework for UK HPR1000 is classified as a HIC and that the frequency of a large steam system piping break should therefore be very low. NS-TAST-GD-042 (Ref. 14) provides advice on application of a graded approach to validation of computer codes and methods (which I consider to include the CHF correlation used), stating that "the required level of validation rigor should be proportionate to the risk significance of the analysis". Given the very low fault frequency, the large steam system piping break is no longer identified by the RP as a design basis fault within the design condition list and acceptance criteria report (Ref. 63). I therefore consider that the same levels of statistical confidence are not necessary for this fault analysis as for DBA.

350. The fault analysis for a large steam system piping break (Ref. 43) predicts that no fuel rods will undergo DNB in the transient if the W3 correlation is used with the recommended limit. Separate analysis shows that hydrodynamic instability, a potential root cause of CHF correlation inaccuracy at low pressure, should not occur (see subsection 4.6.1.3). Independent confirmatory analysis commissioned by ONR (Ref.

22) concludes that the RP's overall modelling of the transient is conservative and predicts that the fault acceptance criteria to avoid loss of fuel integrity are met with wide margin. The consequences of a small non-conservatism in the DNBR correlation of this fault would therefore be limited. On balance, having assessed this topic in conjunction with Fault Studies inspectors I consider this matter to be a minor shortfall.

DNBR Limits Summary

351. I have assessed whether the UK HPR1000 CHF correlations and associated DNBR design limits have been derived sufficiently conservatively using experimental data with representative geometry and operating conditions, in accordance with the expectations of the relevant SAPs described in paragraphs 327-330. I am satisfied that they are adequate for the purpose of GDA but have identified some shortfalls and raised one Assessment Finding.

4.6.1.2 Fuel Temperature Limits

352. The RP submitted Ref. 103 to define the melting point of UO_2 fuel pellets for the purposes of fault analysis, as a function of burnup. I observe that a large selection of data has been considered from a variety of available sources, including the output of an IAEA research project. The melting point derived for unirradiated fuel is bounding of that in the IAEA reference and the assumed reduction in fuel melting point with burnup is more conservative than that which was applied in a previous safety case ONR have assessed. I therefore view the output of the document to be conservative.

353. The actual design basis fuel temperature limit used by the RP in the thermal hydraulic design report (Ref. 95) and in UK HPR1000 fault analyses to avoid fuel melting is lower than that recommended in Ref. 103 for the limiting UK HPR1000 fuel rod average burnup. In my opinion the design basis fuel temperature limit for UO_2 fuel pellets in UK HPR1000 is therefore conservative, particularly so for fresh fuel.

354. For UO_2-GdO_3 fuel pellets, Ref. 103 recommends that the same melting point is assumed as for UO_2 fuel pellets. In my opinion the evidence supporting this is less conclusive, particularly at high burnups. However, I note that the core design of the UK HPR1000 requires all gadolinia-doped fuel pellets to contain lower enrichments of uranium-235 than UO_2 fuel pellets do. This means that (1) these fuel rods are less likely to be limiting in a transient and importantly (2) the maximum burnup reached by gadolinia-doped rods is significantly lower than for UO_2 rods, making the treatment of melting point reduction with burnup even more conservative. I therefore judge that the design basis fuel temperature limit for UK HPR1000 takes adequate account of degradation effects and uncertainties. I consider it to be conservative for the purposes of UK HPR1000 safety analyses for all fuel types in the UK HPR1000 core.

4.6.1.3 Hydrodynamic Instability

355. The RP has submitted Ref. 104 to provide a justification against hydrodynamic instability for the UK HPR1000 core. Both dynamic (density wave oscillation) instability and static (Ledinegg) instability have been considered by the RP.

356. The method presented to justify the plant against dynamic instability is based on published literature. It was developed for parallel closed-channel systems and although simplistic, I judge it to be adequately conservative for an open-channel fuel assembly design in which energy transfer takes place between channels. The results present a large power margin to onset of dynamic instability for the range of conditions analysed. However, the design basis for UK HPR1000 (Ref. 95) only requires that hydrodynamic instability is prevented for normal operating conditions and the most frequent faults. I judged a demonstration to be necessary across a wider range of fault conditions because the presence of hydrodynamic instability would invalidate any DNBR analysis

undertaken with the chosen subchannel code and CHF correlations, which are not validated to predict CHF under oscillating-flow conditions. I therefore raised RQ-UKHPR1000-0683 (Ref. 33) to request more evidence that margin exists to dynamic instability in infrequent fault conditions and in faults occurring during shutdown conditions when the plant could be at lower pressures.

357. The response showed that there is positive margin to dynamic instability in the UK HPR1000 core across a wide range of design basis fault and DEC-A conditions. The limiting case is a large steam system piping break due to the low pressure conditions caused by over-cooling in the RCS, for which small but positive margin is shown. I am satisfied that the response adequately demonstrates that unstable conditions which could invalidate DNBR analyses will not exist for the vast majority of possible faults. However, I observed that the analysis does not include the case of a boron dilution fault when the plant is shut down and at low pressure. For that fault, the RP has submitted transient analysis in Ref. 105, in which it argues that the high neutron flux (source range) reactor trip will provide protection before any boiling occurs in the core. I am satisfied that if no boiling occurs then no dynamic instability will occur. However, Fault Studies inspectors have reported in the Fault Studies Assessment Report (Ref. 7) that the evidence supporting this argument is brief, based primarily on OpEx. An Assessment Finding has been raised in Ref. 7 to ensure that the licensee demonstrates that the high neutron flux (source range) trip setting is adequate to protect against the fault. I anticipate that the licensee's resolution of that Assessment Finding will provide improved evidence to support the argument that boiling will not occur.

358. Static (Ledinegg) instability can theoretically occur in the RCS of a nuclear plant if a small reduction in coolant flow-rate would lead to a significant increase in voidage in the core. The RP argue in Ref. 104 that under DBC-1 and DBC-2 conditions the pressure drop / flow rate curve for the UK HPR1000 RCS is positive, that is, conditions do not exist under which a small reduction in flow would cause enough voidage to increase the pressure drop. Although direct evidence of this is not presented in Ref. 104, I have confidence in the validity of the argument because the core does not have a particularly high power density and a limit has been placed on hot channel void fraction of 5% (Ref. 34), which is lower than the void fraction at some other operational PWRs.

359. Overall, I am therefore satisfied that the core will be hydrodynamically stable in normal operations in accordance with my expectations derived from SAP ERC.3 and satisfied that unstable conditions which could invalidate DNBR predictions in fault analyses will not exist in design basis faults.

4.6.1.4 Allowances made for Through-life Degradation Mechanisms

360. SAP EAD.2 states that adequate margins should exist throughout the life of a facility to allow for the effects of materials ageing and degradation processes on SSCs. I have sampled the way in which the RP's DNBR analysis accounts for two different forms of ageing and degradation in fuel materials: fuel rod bow and fuel assembly bow. I chose these areas to sample because my initial review of the documentation identified potential shortfalls.

361. Both of these phenomena can change the size of the coolant channel gap between adjacent fuel rods, which changes local coolant flow and can also change local power. Allowances need to be made for this in DNBR analysis.

Impact of Fuel Rod Bow

362. The RP has quantified the impact of fuel rod bow as a function of burnup using an empirical relationship defined in its fuel/reactor design interface data report (Ref. 106)

and used this to apply a penalty to the DNBR design limits (Ref. 96). I am familiar with the empirical relationship from previous GDA and have seen evidence that it is conservative. However, the penalty has been applied for UK HPR1000 as a fixed value, using rod bow data for fuel at a burnup of [redacted], which is well below the fuel rod burnup limit for the UK HPR1000 core. In order to justify this approach, the RP has submitted a dedicated report (Ref. 107) to show that the effect of increasing the rod bow penalty at higher burnups is off-set for UK HPR1000 by reduced F_{dH} values in high burnup fuel (which are not otherwise credited in the analysis). An 'iso-DNBR' curve has been produced in Ref. 107 to show the amount by which F_{dH} must reduce in order to offset the effect of the higher rod bow penalty on DNBR analysis for fuel at the UK HPR1000 fuel rod burnup limit. Evidence has then been produced to show that all fuel assemblies in the core at burnups higher than [redacted] have F_{dH} lower than the limit defined by the iso-DNBR curve.

363. The magnitude of F_{dH} reduction required by the iso-DNBR curve is broadly as I would expect given the size of the rod bow penalty defined in Ref. 106 for higher burnup fuel. I observe in Ref. 107 that all fuel assemblies in the current UK HPR1000 equilibrium core design have F_{dH} substantially below the iso-DNBR curve limit once they reach a burnup of [redacted] such that the rod bow penalty applied will result in a conservative DNBR limit. No fuel assemblies reach this burnup in the first cycle. I am therefore satisfied that the way in which fuel rod bow has been accounted for in the DNBR analysis is adequate for the currently-defined UK HPR1000 fuel cycles. However, the RP has recognised during our interactions in Step 4 that a significant change in core design by the licensee could invalidate the demonstration that high burnup fuel assemblies have sufficiently low F_{dH}. No operating rule or clear safety case assumption has been identified to prompt the licensee to verify that the safety case assumptions about F_{dH} in high burnup fuel assemblies remain valid for new core designs. The assumption should be made clearer and more traceable so that the licensee can record an appropriate operating rule. This is not the only shortfall I have identified of this type, so in subsection 4.12 I have listed all those I have identified and recommended a means by which they can be addressed.

Impact of Fuel Assembly Bow on Thermal Hydraulic Performance at the Fuel Assembly Edge

364. As noted previously, the thermal hydraulic design report (Ref. 95) does not describe any means to explicitly account for the effects of fuel assembly bow on thermal hydraulic performance. As outlined in NS-TAST-GD-075 (Ref. 11), two separate effects should be addressed, either of which could reduce the local CHF:

- An increase in local power at the edge of a fuel assembly due to an increased fuel assembly gap, which occurs due to fuel assembly bowing under irradiation.
- A reduction in local flow at the edge of the fuel assembly due to a reduced fuel assembly gap, which occurs due to fuel assembly bowing under irradiation and needs to be considered together with the effect of the local mixing grid geometry.

365. I raised RO-UKHPR1000-0045 (Ref. 72) to ensure this topic was resolved for UK HPR1000 during GDA. In summary, the RP's approach has been to:

- predict the range of modified water gaps that will occur in the UK HPR1000 core due to fuel assembly bow, using predictive models verified with data from existing plants;
- predict the impact of the modified water gaps on local power peaking at the edge of the fuel assembly using the PINE and COCO nuclear design codes;
- predict the impact of the modified water gaps on local mass flow rates and pressure loss coefficients at the edge of the fuel assembly using CFD

techniques, particularly for the case where the gap is 'closed' and mixing grids in adjacent fuel assemblies are touching;

- predict the impact of the local flow disturbances and local power peaking on DNBR at the fuel assembly edge using the LINDEN sub-channel code;
- produce equivalent DNBR predictions for the interior of the fuel assembly using the usual fault analysis power distribution assumptions, ignoring the effects of fuel assembly bow; and
- compare the DNBR predictions to show that the case using the usual fault analysis assumptions remains bounding and that the fault analyses in the existing safety case are therefore conservative even though fuel assembly bow effects are ignored.

366. I have assessed all the reports submitted by the RP in response to the RO (Ref. 73, Ref. 74, Ref. 108 and Ref. 109). Responses to RQs I raised have subsequently been consolidated in to these references. My expectations were primarily informed by SAPs ERC.1, EAD.2, FA.7, AV.1, AV.2, Ref. 11 and Ref. 14. My full assessment of the response to this RO is further detailed in Ref. 110. In summary:

- Informed by SAPs EAD.2, AV.1 and AV.2, I am satisfied that credible estimates of fuel assembly edge water gaps have been developed in Ref. 73. The effects of materials ageing and degradation processes have been adequately considered in the analysis and sufficient evidence of the method's validity has been provided for GDA. I judge that a surveillance scheme should be specified for UK HPR1000 to verify that real assembly bow measurements on the plant are in accordance with the assumptions of the safety case.
- Informed by SAP FA.7, I am satisfied the calculated water gap distributions have been applied conservatively using the PINE and COCO neutronic codes in Ref. 74 to produce modified power distributions resulting from fuel assembly bow. I am satisfied that uncertainties in the calculation have been accounted for in a conservative manner.
- Informed by the AV SAPs, I judge that adequate evidence has been provided in Ref. 109 to support application of Framatome's 3D CFD code and modelling methods to predict the effect of assembly edge geometry on local coolant flow and pressure loss coefficients. I am also satisfied with the arguments and experimental evidence presented in Ref. 108 and Ref. 109 to support the way in which the calculated local pressure loss coefficients have been applied in a sub-channel code to produce DNBR predictions at the assembly edge.
- Informed by the AV SAPs and FA.7, I judge that Framatome have submitted sufficient experimental evidence in Ref. 108 to show that use of the FC2000 CHF correlation with its existing UK HPR1000 design limit and statistical parameters remain conservative for prediction of DNBR at the assembly edge, including in the case where the gap between assemblies is closed due to assembly bow.
- Informed by SAP ERC.1, I am satisfied that the analysis used in Ref. 74 to predict DNBR at the assembly edge, using the data derived by the other means, covers all relevant core operating states.
- I am satisfied that the DNBR analysis results in Ref. 74 show that the radial power distributions used in the existing UK HPR1000 fault analyses in PCSR Chapter 12 are sufficiently conservative that the DNBR predictions will bound the DNBR that would be predicted at the fuel assembly edge with the increased local power or reduced flow resulting from the water gap distributions predicted in Ref. 73.

367. Overall, I judge that the submissions provide adequate evidence that DNBR predictions used in UK HPR1000 fault analyses are conservative when allowing for assembly bow in all relevant permitted modes of operation. This conclusion is dependent on the adequacy of water gap predictions produced for the AFA 3GAA fuel

assemblies in the UK HPR1000 core design in Ref. 73. I judge it is important that a surveillance scheme be specified for UK HPR1000 to verify that real assembly bow measurements on the plant (and therefore water gap distributions) are in accordance with the assumptions of the safety case. The RP's concluding report for the effect of fuel assembly bow on DNBR (Ref. 74) states that a surveillance scheme will be developed by the licensee. I am satisfied with this approach but the requirement for a surveillance scheme should be made clearer in the safety case. This is not the only shortfall I have identified of this type, so in subsection 4.12 I have listed all those I have identified and recommended a means by which they can be addressed.

4.6.1.5 Adequacy of Reactor Trip Settings

368. The adequacy of reactor trip functions and the associated setpoints is primarily substantiated by successful design basis fault analyses, which have been assessed in the Fault Studies Assessment Report (Ref. 7). However, informed by SAP ERC.1 I also expect a demonstration to be provided that adequate safety margin is retained to fuel integrity criteria at the limits of the operating zone, were a condition to occur in which the trip setpoint were almost but not quite reached. I have sampled the RP's demonstration that its reactor trip setpoints will prevent breach of DNBR and fuel temperature criteria in these circumstances.

369. The RP has described two variable trips known as the over-temperature ΔT trip and the over-power ΔT trip (Ref. 111). Both of these form part of the F-SC1 classified RPS. They trigger a reactor trip if the temperature difference across the core reaches a particular value that is calculated as a function of measured average coolant temperature, measured pump speed, measured axial offset and, for the over-temperature ΔT trip, measured pressurizer pressure. Use of trip functions of this type is common to other modern PWR designs, so I am satisfied they can provide adequate protection against fuel failures due to inadequate heat removal if the setpoints are set correctly.

370. Within the Power Capability Analysis Report (Ref. 112) the RP has submitted a summary of the work done to show that fuel integrity will not be lost under conditions just prior to the over-temperature ΔT and over-power ΔT trip setpoints being reached. This includes analysing three transients that cause over-power conditions of different types – an RCCA bank withdrawal fault, a boron dilution fault and a secondary excessive load increase fault – and showing that design limits are not breached at operating points prior to reactor trip.

371. I have sufficient confidence to judge that the fuel temperature limit will not be breached under conditions in which the over-power ΔT trip setpoint has not been reached. However, I sought further evidence during GDA that adequate DNBR margin would be retained at the limits of the operating zone before either the over-temperature ΔT or over-power ΔT trip set-points are reached.

372. The 95% probability and confidence level inherent in the RP's DNBR limits allow for the possibility that a small number of fuel rods in the core will undergo DNB at conditions just before the DNBR limit is reached. The RP has submitted new analysis in Ref. 113 to show that at the limits of the operating zone prior to either over-temperature ΔT or over-power ΔT trip set-points being reached, the number of fuel rods statistically predicted to be in DNB is < 1.0, which is a more stringent requirement than that the DNBR limit be met. This addressed feedback provided during Step 3 of GDA and I consider the approach to be good practice for this purpose.

373. The method used by the RP in Ref. 113 is to calculate for a particular set of conditions (1) the distribution of fuel rods in the core as a function of relative power level and (2) the statistical probability to a 95% confidence level that a fuel rod at a given relative power level will be in DNB. The RP has combined the two calculation results to

determine a statistical prediction of the number of fuel rods in DNB under those conditions. The conditions used for the analysis correspond to the intersection of the over-temperature ΔT and over-power ΔT trip setpoints when plotted on a graph of ΔT as a function of core average temperature and also bound the power setpoint of the high flux neutron trip (which is also part of the RPS). In my opinion they represent an appropriate set of limiting conditions for which to undertake this analysis. Ref. 113 describes the assumptions and uncertainty margins used in the DNBR analysis, which I am satisfied are adequately conservative for this application. The results show a statistical prediction that < 1.0 fuel rods will be in DNB in this set of core conditions. Overall, I consider this to be a good demonstration that safety margin exists at core conditions immediately below the ΔT trip setpoints and that no fuel rods should enter DNB in those conditions.

374. The over-power ΔT trip is required to protect against loss of fuel integrity due to PCI failures as well as breaches of DNBR or fuel temperature limits. However, the likelihood of PCI failure depends on the magnitude and gradient of a power ramp as well as the limiting conditions that are reached, so can only be assessed through specific fault transient analyses. PCI transient analyses have been conducted for UK HPR1000 and I assess them in subsection 4.7 of this report. Margin shortfalls originally discovered by the analysis in over-cooling faults led to a change to the over-power ΔT trip setpoint during GDA, which is implemented in Ref. 111 such that reactor trip will occur at slightly lower ΔT if the core has a strongly negative axial offset. This change has allowed the RP to demonstrate that the over-power ΔT trip provides adequate protection against PCI failures.

375. I am therefore satisfied with the RP’s demonstration that adequate safety margin is retained to fuel integrity criteria at the limits of the operating zone before a reactor trip occurs. This satisfies my expectations on this topic derived from SAP ERC.1. From a Fuel and Core perspective, I consider the setpoints for the over-temperature ΔT trip and the over-power ΔT trip to be adequate.

4.6.1.6 Pressure Drop and Bypass Flow Predictions

376. Pressure drop and bypass flow data for the reactor core and RPV internals, used in a range of fault analyses, have been reported by the RP in Ref. 114. The purpose of this subsection of my report is to assess the adequacy of this data. The data are generated using a simple method that relies on input data in the form of pressure loss coefficients. For fuel and other core components, these data have been provided by the fuel designer and the RP has submitted a report synthesising pressure loss measurements undertaken for AFA 3GAA fuel assemblies (Ref. 115), providing evidence that fuel data are derived from test programmes using representative geometry under appropriate conditions. The source of pressure loss coefficients for other RPV internal components is not explained in Ref. 114, but the thermal hydraulic design report (Ref. 95) states that they were obtained from hydraulic test data, which is in accordance with my expectations.

377. The bypass flow paths considered by the RP match my expectations for this design of reactor vessel internals, and estimates are presented with wide uncertainty bands covering uncertainty in the operating parameters and/or loss coefficients. The total bypass flow of 6.5 % assumed in fault analysis is a bounding figure after all the predicted uncertainties are conservatively summed up and is a similar value to that I which have observed for another similar PWR design. The total pressure drop predictions will be validated in future as normal business during plant commissioning. I therefore chose not to target the non-fuel component pressure loss coefficients and underlying test data for more detailed assessment in GDA. From my assessment of the overall approach used and the underlying data for the fuel, informed by SAPs FA.7 and AV.3 I am satisfied that the pressure drop and bypass flow data provided is adequate.

4.6.2 Strengths

378. Following my assessment of the UK HPR1000 thermal hydraulic design I have identified the following strengths:

- This part of the safety case is well-structured and I have satisfied myself that important aspects of core thermal hydraulic design have all been addressed by the RP through a comparison against guidance in NS-TAST-GD-075 (Ref. 11) and SSG-52 (Ref. 19).
- The design basis fuel temperature limit for UO_2 fuel pellets in UK HPR1000 is conservative, particularly so for fresh fuel.
- I am satisfied that the core will be hydrodynamically stable in normal operations and that unstable conditions which could invalidate DNBR predictions will not exist in design basis faults.
- The RP has provided evidence that DNBR predictions used in UK HPR1000 fault analyses are conservative when allowing for assembly bow in all relevant permitted modes of operation.
- The RP has provided a demonstration that at the limits of the operating zone, prior to either over-temperature ΔT or over-power ΔT trip set-points being reached, the number of fuel rods statistically predicted to be in DNB is < 1.0, which I consider to be good practice.

4.6.3 Outcomes

379. Following my assessment of the UK HPR1000 thermal hydraulic design I have identified the following outcome:

- I have raised an Assessment Finding to ensure that the licensee validates the implementation of the chosen CHF correlation(s) in the sub-channel analysis code used to undertake fault analysis calculations.

380. I also observed three minor shortfalls during my assessment of this part of the safety case.

4.6.4 Conclusion

381. Based on the outcome of my assessment, I have concluded that the thermal hydraulic design of UK HPR1000 is adequate. I am satisfied that the expectations set in this context by SAPs ERC.1, ERC.3, EAD.2, FA.7 are all met. I am satisfied that the expectations set by AV.3 will also be met following resolution of the Assessment Finding I have raised.

4.7 Protection Against Pellet Clad Interaction

4.7.1 Assessment

382. PCI-SCC fuel clad failures are a result of the combined effects of fuel pellet expansion and the presence of a corrosive fission product environment. In PWR, these fuel failures are usually associated with reactivity faults because the expanding pellet generates a clad stress and releases additional fission products. The fuel is usually most susceptible to faults occurring after a period of low power operation because it becomes ‘de-conditioned’ (described further in paragraph 397).

383. Reactor designers usually protect against PCI-SCC fuel failures by defining safe operating power ranges or optimising the protection system designs to limit the maximum linear power increase that the fuel experiences during frequent faults. I consider that ONR SAPs ERC.1 and FA.7 are relevant to the UK HPR1000’s protection against PCI-SCC fuel failures.

384. NS-TAST-GD-075 (Ref. 11) advises that the safety case should present adequate operating limits and automated protection to prevent fuel failures by PCI-SCC in normal operation and frequent faults. Using guidance in the SAPs and Ref. 11 my expectations are therefore that the UK HPR1000 safety case should demonstrate adequate protection to prevent PCI-SCC fuel failures in all permitted power ranges in normal operation and for frequent faults.

385. The RP's safety case claims that the generic UK HPR1000 design will ensure that there are no fuel failures in faults with a frequency greater than 10^{-3} per year (Ref. 67). To substantiate this claim, the RP has submitted evidence for normal operation and frequent faults in the following reports:

- PCI Technological Limit for Fuel Rods with M5 Framatome Cladding (Ref. 82)
- PCI Transient Analysis (Ref. 117)
- PCI Thermal-Mechanical Analysis (Ref. 118)
- PCI Thermal-Mechanical Analysis in Frequent Fault (Ref. 119)

386. CGN identified and analysed the bounding frequent reactivity faults for the purposes of PCI analysis, then supplied the output from this reactivity fault analysis to Framatome. Framatome used this data to carry out the PCI-SCC thermal mechanical analysis of the fuel against the technical acceptance criteria (Ref. 82). The RP submitted Ref. 117 to report CGN's parts of this work and Ref. 118 and Ref. 119 to report Framatome's parts of the work.

387. I have assessed Ref. 82 as part of the set of technical acceptance criteria specified for the UK HPR1000 fuel in sub-section 4.5.

388. Some reactor designs include an independent, dedicated, but often lower safety classification system to protect against PCI-SCC fuel failures, using in-core detectors and a dedicated Control and Instrumentation (C&I) function to predict margin to PCI failure risk. International OpEx suggests that plant designs that lack this additional protection may need additional limitations to be imposed on their flexible electrical output operations at the end of their fuel cycles (Ref. 120). The UK HPR1000 does not incorporate such a system, instead providing protection against PCI-SCC fuel failures using the F-SC1 classified RPS (primarily the overpower ΔT trip). Therefore, I sampled the transient analysis and PCI-SCC analysis in detail to gain confidence that the generic UK HPR1000 design has adequate protection in this area.

389. In subsections 4.7.1.1 and 4.7.1.2 I cover my assessment of the PCI safety case for the UK HPR1000 design in GDA. In sub-section 4.7.1.3 I cover my assessment of the impact on the PCI safety case of some potential post GDA design modifications that have been proposed by the RP to improve compliance with the UK grid code.

4.7.1.1 CGN PCI-SCC Transient Analysis

390. The RP submitted Ref. 117 to demonstrate that CGN has selected the relevant frequent faults for Framatome's PCI-SCC analysis. This document describes the method for selecting the bounding faults and conservatively analysing them. The selected frequent faults are:

- Uncontrolled RCCA Bank Withdrawal at Power
- RCCA Misalignment up to Rod Drop
- Decrease in Boron Concentration in Reactor Coolant
- Excessive Increase in Secondary Steam Flow
- Inadvertent Opening of One SG Relief Train or of One Safety Valve
- Uncontrolled Single RCCA Withdrawal

391. Ref. 117 argues that these frequent faults are bounding of faults not analysed because they have the potential for the greatest increase in the fuel's linear power density.

392. The transient analysis uses the computer codes PINE, COCO, POPLAR and GINKGO. I noted that these are the same computer codes as used other deterministic fault analysis. My assessment of the adequacy of PINE, COCO and POPLAR is reported in subsection 4.13 of this report and that of GINKGO is captured in the Fault Studies assessment (Ref. 7).

393. I have compared the list of frequent faults in the PCI-SCC Transient Analysis Report to the design condition list (Ref. 59). I consider that it contains the relevant frequent reactivity faults and an adequate justification for the identification and use of a selection of those faults as the bounding faults. I sampled against the RCCA withdrawal transients and confirmed that the RP has selected the most onerous case. On that basis I am satisfied that bounding transients have been selected for PCI-SCC analysis.

394. I also examined the input assumptions used in Ref. 117. I observed that the analysis considers three different burnup points, five different power levels, the maximum and minimum possible RCCA bank insertion level at each power and the two extremes of core axial offset that are allowed during operation. On that basis I am satisfied that the analysis adequately accounts for all permitted operating modes of the reactor when at power.

395. I judge that CGN's PCI-SCC transient analysis is adequate to provide limiting transient data for input to Framatome's PCI thermal mechanical analysis of the fuel.

4.7.1.2 Framatome PCI-SCC Thermal Mechanical Analysis

396. The results of Framatome's analysis are presented in two reports. Ref. 118 presents the results for limiting DBC-1 and DBC-2 conditions while Ref. 119 presents the results for limiting DBC-3 conditions that are considered frequent faults because they have a frequency greater than 10^{-3} per year. In each of the two reports, the PCI-SCC thermal mechanical results are presented separately for (1) reactivity faults during routine operations and (2) reactivity faults occurring after a period of ELPO.

397. When power in a reactor is reduced the fuel pellet shrinks slightly and during a subsequent period of ELPO the clad creeps down under irradiation, re-closing the pellet clad gap. When power is subsequently increased, the expanding pellet causes additional tensile stresses in the clad that make it more susceptible to PCI-SCC failure for a period of time until further irradiation creep in the clad relaxes the tensile stresses. As a result of these phenomena (sometimes referred to as conditioning and de-conditioning of the fuel), the clad is usually most susceptible to PCI-SCC failure if a transient occurs just after the end of a period of ELPO.

398. The PCI-SCC thermal mechanical analysis uses a quantity known as SED, which is a time-integrated function of the clad's inner hoop stress and strain during a transient. The analysis method simulates an increasing power transient and calculates the maximum SED for each part of the fuel clad's axial mesh as the transient progresses. The calculated axial mesh's SED will increase until it exceeds the SED limit. As reported in subsection 4.5, I am satisfied that the SED limit defined to prevent PCI-SCC failures for UK HPR1000 is conservative.

399. The maximum allowable linear power density for each axial mesh in the fuel model is determined by the point in the transient at which the SED limit is reached in that location. To complete the analysis, the local axial mesh's maximum allowable linear power density is compared against the linear power density from Ref. 117. A positive

margin between the maximum linear power density predicted in the fault and the maximum allowable linear power density should preclude PCI-SCC failure.

400. I am satisfied that the PCI-SCC thermal mechanical analysis method aligns with RGP in SSG-52 (Ref. 19) and is similar to a method previously assessed by ONR. Therefore, at a principle level I judge that this method is adequate to assess the risk of PCI-SCC failures. The analysis uses the COPERNIC computer code. To justify the adequacy of this code, the RP has submitted a V&V report (Ref. 32). ONR has experience of both the overall thermal mechanical analysis method and the COPERNIC code through our GDA of the UK EPR. Therefore, I chose not to sample the evidence underlying their adequacy in this GDA.

401. The analysis originally submitted predicted that fuel would fail in the limiting frequent fault, either at EOC during routine operations or at both MOC and EOC following a 30 day period of ELPO. The limiting fault was the inadvertent opening of an SG relief valve train. The RP reviewed potential options to improve the generic UK HPR1000 design (Ref. 121). These options included modifications to the reactor's secondary system, restricting reduced power operation and modifying the RPS. The RP decided that the option to reduce risk ALARP was a modification to the RPS, M88 (Ref. 116). Specifically, the modification is a change to the setpoint of the F-SC1 classified over-power ΔT trip, such that reactor trip will occur at slightly lower ΔT if the core has a strongly negative axial offset. This has most benefit to safety margins in over-cooling faults because the power increase is most pronounced low in the core and axial offset becomes more negative.

402. Following modification M88, CGN and Framatome repeated their analyses, reported in Ref. 117, 118 and 119. These results show adequate margins to the PCI-SCC criteria for the bounding frequent reactivity faults during routine operations. For the ELPO cases, there is adequate margin following up to 30 days ELPO between BOC and MOC, and following 15 days of ELPO through to EOC. I consider that following modification M88, the RP has demonstrated that the design has adequate protection against PCI-SCC fuel clad failures in frequent faults during any permitted power state. I am satisfied that the modification selected is the most appropriate solution of the options presented in Ref. 121 to reduce the risk of PCI-SCC failures because it does not require any major physical modification to the plant and does not cause detriment to any other aspect of the safety case.

4.7.1.3 Operating Rules and Impact of PCI Limitations on Grid Code Compliance

403. As previously discussed in sub-section 4.3.1.5, the RP has submitted Ref. 42 to present an analysis of potential gaps in compliance of the generic UK HPR1000 design with UK grid code requirements.

404. Ref. 42 aims to demonstrate the feasibility of potential post-GDA design modifications to enable additional operating modes and close the identified gaps in grid code compliance. The reactor core safety case needs to demonstrate that the fundamental safety functions will be delivered with sufficient confidence in all permitted operating modes, as set out by SAP ERC.1. In sub-section 4.3.1.5, I considered whether the impact of the potential design modifications on core neutronic and kinetic data have been analysed and considered by the RP in sufficient depth to support the conclusions of the feasibility study. In this sub-section, I consider the impact of the potential design modifications on PCI fault analysis.

405. I have also assessed whether the flexible operation capability claimed for the UK HPR1000 GDA design in Ref. 42 is supported by the PCI-SCC analysis discussed in the previous subsections of this report. I particularly looked at the claims made in Ref. 42 about ELPO operations, because the PCI-SCC margin in a fault following a period of ELPO is sensitive to both the ELPO duration and power level.

406. Ref. 42 presents limitations on allowable periods of ELPO due to the need to maintain margin to PCI limits. The stated allowable periods are 30 days between BOC and MOC and 15 days between MOC and EOC, which I observed to be consistent with the ELPO periods for which the PCI-SCC analysis discussed in the previous sub-section showed that fuel would be protected in frequent faults. Ref. 42 also presents a minimum stable operating level of 65% full power. I am satisfied that this limit is conservatively bounded by the PCI-SCC analysis discussed above, which analysed ELPO cases with powers from 30% full power upwards.

407. The PCI-SCC analysis discussed in the previous subsections is limited to consideration of a single period of ELPO in a cycle. In Ref. 42, the RP argues that PCI margin could be recovered during a period of full power operation after a period of ELPO, such that further periods of ELPO could be undertaken up to EOC whilst retaining sufficient PCI margin. Specifically the RP claims that based on OpEx in China, PCI margin can be fully recovered (from a limiting ELPO transient) after a period of [redacted] days at full power, and also that sufficient PCI margin could be maintained during a continuous cycle of 2 days ELPO followed by 5 days at full power, for a complete operating cycle. I have no reason to doubt the feasibility of these modes of operation, but I observe that the PCI-SCC analysis submitted in GDA does not allow for them. The licensee will need to define a clear set of operating rules for low power operations with the UK HPR1000 plant and complete further safety analyses to show that all safety margins are adequate in all permitted operating modes.

408. The PCI-SCC analysis discussed in the previous subsections is also limited to consideration of a PFC response capability of +/- 3% full power from the primary plant. In the PCI-SCC thermal mechanical analysis in Ref. 118 and 119, the RP has applied a penalty (or ‘bias’) to the PCI SED acceptance criterion to allow for the effect of cumulative fatigue (from small power changes in PFC mode) on fuel clad integrity during a PCI transient. However, Ref. 42 explains that the UK grid code may require a PFC response capability of up to +/- 10% full power. It presents arguments that the penalty applied to the PCI SED acceptance criterion in the extant safety case will remain conservative if a plant design modification is made to allow PFC response capability of up to +/- 10% full power to be implemented. Supporting evidence is presented in Ref. 122. I have assessed this with consideration of SAPs ERC.1 and FA.7, whilst recognising that the intent of Ref. 42 is to present a feasibility study rather than a full safety case.

409. Ref. 122 describes the method used to verify the penalty and the different irradiation histories considered. It shows that the penalty has been derived to bound the observed impact of cumulative fatigue on fuel SED at a range of local power levels in the fuel. The RP conclude in Ref. 122 that the penalty applied to the PCI-SCC analysis discussed in the previous subsections of this report is sufficient to allow for the effect of cumulative fatigue for both the +/- 3% and +/- 10% power PFC profiles.

410. I am satisfied that on the basis of the information provided in Ref. 122, the method described should ensure that the penalty applied to the PCI SED criterion is valid for the full range of permitted operating modes and should provide for conservative fault analysis results, in accordance with the expectations of SAPs ERC.1 and FA.7. There are some pieces of supporting evidence that have not been submitted in GDA. However, I observed in my assessment of the SED limit for PCI-SCC (subsection 4.5, paragraph 269) that the limit is set in a conservative manner from the available ramp test data. In my opinion this additional conservatism provides confidence that any remaining uncertainty in the SED penalty due to operation in PFC mode should not compromise the conservative nature of the SED criterion applied in faults.

411. In conclusion, I am satisfied that the work reported in Ref. 42 and Ref. 122 is adequate to demonstrate that it is feasible to retain adequate PCI-SCC safety margin if the UK

HPR1000 plant is operated with a PFC response capability of +/- 10% full power and the maximum ELPO durations stated. Any future design modification(s) made to implement this change for the purposes of grid code compliance will be tracked through Assessment Finding AF-UKHPR1000-0020, which has been raised in the Electrical Engineering Assessment Report (Ref. 57). As part of the resolution to that Assessment Finding, if PFC or ELPO capability are extended beyond that of the GDA design and safety case then I would expect the licensee to extend the PCI-SCC analysis accordingly, including all the necessary operating rules and underlying evidence to support it. However, this does not constitute a shortfall in the existing safety case from a Fuel and Core perspective.

4.7.2 Strengths

412. Following my assessment of the UK HPR1000 safety case for protection against PCI I have identified the following strength:

- UK HPR1000 has adequate protection against PCI-SCC fuel failures in frequent faults while maintaining a level of operational flexibility.

4.7.3 Outcomes

413. Following my assessment of the UK HPR1000 safety case for protection against PCI in GDA I have identified no Assessment Findings or minor shortfalls. Shortfalls that I originally identified in the safety case were addressed by the RP during GDA.

414. The implementation of any future design modification(s) made for the purposes of grid code compliance will be tracked through Assessment Finding AF-UKHPR1000-0020, which has been raised in the Electrical Engineering Assessment Report (Ref. 57). As part of the resolution to that Assessment Finding, if PFC or ELPO capability are extended beyond that of the GDA design and safety case then I would expect the licensee to extend the PCI-SCC analysis accordingly, including all the necessary operating rules and underlying evidence to support it. However, this does not constitute a shortfall in the existing safety case from a Fuel and Core perspective.

4.7.4 Conclusion

415. Based on the outcome of my assessment, I have concluded that the RP has improved its safety justification and implemented a reasonably practicable improvement to the RPS during GDA. This ensures that there is adequate protection against PCI-SCC failures in frequent faults. Therefore, I judge that the generic UK HPR1000 design and the RP’s safety case has met my expectations derived from SAPs ERC. 1 and FA.7 in this area.

4.8 Fuel Behaviour in a Loss of Coolant Accident

4.8.1 Assessment

416. This section records my assessment of the fuel and core aspects of the UK HPR1000 safety case for LOCA. Other aspects of the LOCA safety case have been considered by a range of ONR specialists and discussed in their respective assessment reports, notably the Fault Studies Assessment Report (Ref. 7).

417. The RP has adopted the same technical acceptance criteria for the fuel in all LOCA faults. I have focused my assessment on the LB-LOCA, the most onerous LOCA for the fuel, to give me confidence in the substantiation of the safety case for less onerous LOCA faults.

418. My assessment of LOCA has focused on the UK HPR1000 safety case claims and arguments for:

- the fuel's LB-LOCA technical acceptance criteria, equivalent to my assessment of the criteria used for other faults in sub-section 4.5; and
- the fault analysis methods for modelling fuel behaviour in LB-LOCA.

419. The LB-LOCA is a complex fault. To provide context, I have included a summary of the fault in this sub-section. I have used this background to inform my expectations and areas of assessment.

4.8.1.1 LB-LOCA Background

420. Generally, for a PWR, the limiting LB-LOCA is a double-ended guillotine break of the RCS pipeline between the RCP and RPV. This break causes a rapid depressurisation of the RPV and loss of coolant inventory. I have briefly described the main aspects of the transient below.

421. The LB-LOCA causes the RPV water to flash to steam. The increased voiding together with reactor trip causes a neutronic shutdown. When combined with the loss of coolant inventory the decreasing RPV water level leads to uncovery of the fuel. The combination of reduced power and reduced cooling flattens the fuel rod's radial temperature profile, with an increase in fuel clad temperatures. This phase is often called the blowdown phase.

422. In most PWR designs, the emergency cooling systems will inject boronated water into the RCS to counteract the LOCA. While some of this water may bypass the RPV, the water that does not will begin to refill the RPV. However, during this refilling phase the core has a stationary steam atmosphere. This has a poor heat removal capability, so fuel rod temperatures rise significantly. This can cause a material phase change in the clad that promotes super-plasticity and expansion driven by the internal fission gas pressure. The phenomenon is known as clad ballooning. In the LB-LOCA, ballooning can cause the higher rated fuel rods to burst and can lead to cooling flow blockages. The phase of a LOCA where the water level increases above the bottom of fuel is known as the reflood phase.

423. During the reflood, the rising water level quenches the fuel rods. As these rods cool, they experience film boiling before transitioning to nucleate boiling. The fuel clad can undergo significant oxidation during this process. Excessive clad oxidation can lead to clad embrittlement and brittle fracturing. Together, clad burst, loss of local cooling due to blockages and brittle fracturing due to oxidation can lead to significant releases of radiological material into the coolant and subsequently containment (Ref. 123).

424. Historically, many international reactor vendors and regulators have followed the US NRC's 10 CFR 50.46 (Ref. 124) requirements to demonstrate that their designs have adequate protection against significant radiological release for the limiting LOCA. These regulatory requirements are intended to demonstrate, with high confidence, that a reactor's emergency cooling systems can prevent the limiting LOCA from releasing excessive radiological material from the fuel into containment.

425. The US NRC's 10 CFR 50.46 prescribes the fault analysis methods and acceptance criteria for LOCA for use in the context of its regulatory regime. This involves the US NRC approving the analysis method. Licensees can use the analysis rules from 10 CFR 50 Appendix K to treat relevant phenomena conservatively, or use approved best estimate alternatives. The acceptance criteria used with these approved methods and rules are:

- PCT remains less than 1204ºC;

- clad oxidation remains less than 17% of the total clad thickness;
- hydrogen generation remains less than 1% of the potential maximum;
- the changes in core geometry do not prevent cooling; and
- the emergency cooling systems will preserve long term decay heat removal.

426. As I discuss further in the following sub-sections, the RP have used these criteria within their LOCA safety case. The RP has submitted a specific report to justify the adequacy of these criteria for the UK HPR1000 fuel design (Ref. 125).

427. The PCT limit is intended to demonstrate that the highest fuel clad temperature in the core is below the threshold for a self-sustaining oxidation reaction. The clad oxidisation limit is based on a time at temperature correlation. Together, these two criteria are intended to give confidence that the fuel clad will maintain sufficient ductility so that it will not suffer brittle failure during the reflood stage of a LOCA.

428. The coolable geometry criteria is an overall objective that consolidates the PCT and clad oxidation criteria with other fuel damaging phenomena. For example, fuel melt, mechanical damage or flow blockages. The objective of this criteria is to ensure that the fuel can continue to be cooled and therefore retains sufficient radiological material following the LOCA.

429. 10 CFR 50.46 has two further criteria for hydrogen generation and maintaining long term cooling. These criteria have different objectives associated with protecting the containment building and ensuring the provision of long-term decay heat removal, which are beyond the scope of this report.

4.8.1.2 Assessment Expectations and Strategy

430. My regulatory expectations for this topic are cognisant of the context set by the UK HPR1000 safety case claims and arguments for the LB-LOCA, as well as information from the ONR Fault Studies and PSA assessment activities. The key aspects of the UK HPR1000 safety case that are relevant to my regulatory expectations are:

- the RP has designated the LB-LOCA as a beyond design basis fault for deterministic safety analysis (Ref. 126);
- the UK HPR1000 PSA risk analysis (Ref. 127) considers the LB-LOCA as a low frequency initiating event and shows that LB-LOCA sequences make a small contribution to overall plant risk; and
- the RP has submitted deterministic analysis for the LB-LOCA transient against thermal acceptance criteria (Ref. 128) and mechanical damage criteria (Ref. 126).

431. The reason the RP do not classify LB-LOCA as a design basis fault is that it has designated the RCS pipework as HIC and claims a consequent reduction in IEF. Fault studies inspectors have assessed the validity of this approach in Ref. 7 against the expectations established by SAP FA.5 and are satisfied it is reasonable.

432. The UK HPR1000 PSA modelling claims that successful operation of the emergency cooling systems following LB-LOCA will protect the core and prevent extensive core damage. The low IEF and redundancy in the emergency cooling systems mean that the LB-LOCA is less than 1% of the overall core damage frequency. ONR's PSA inspector has assessed the PSA modelling of this fault (Ref. 129) and found it to be sensible and logical, providing an adequate representation of the evolution of the accident sequence.

433. The RP's safety case clarification and core assessment report for LB-LOCA (Ref. 126) presents the deterministic fault analysis as a special case, used to show defence in depth and to determine the sizes of relevant safety systems. Based on the thermal and

mechanical analyses, the RP has concluded that the technical acceptance criteria will be met and the LB-LOCA fault will not result in a loss of coolable geometry or prevent RCCA insertion.

434. ONR’s expectations as set out in SAP FA.15 (and consistent with international guidance such as Ref. 17 and Ref. 18) are that initiating events outside of the design basis should be analysed with a best-estimate approach to demonstrate there is no sudden escalation of consequences (for example an escalation to a severe accident). Consistent with this, NS-TAST-GD-075 (Ref. 11) states that the level of conservatism included in DEC-A calculations may be reduced, and judgements on what is ALARP may be different, when compared to the approach for design basis faults. I have applied this guidance within my assessment. I also expect that the analysis methods used for such faults should be adequately validated, so I have applied SAPs AV.1 and AV.2. However, informed by guidance in IAEA SSG-2 (Ref. 18), I recognise that the level of confidence provided by the validation should be appropriate to the type of analysis and again, may be lower for a fault outside of the design basis.

435. Informed by guidance in Ref. 11 and IAEA SSG-52 (Ref. 19), I have assessed whether the RP’s acceptance criteria and analysis adequately account for the following fuel phenomena that can challenge fuel integrity and/or coolability in LB-LOCA:

- Clad and Fuel Pellet Melt
- Clad Ballooning and Flow Blockage
- Clad Oxidation and Embrittlement
- Fuel Fragmentation, Relocation and Dispersal (FFRD)
- Fuel Assembly Structural Deformation

4.8.1.3 Clad and Fuel Pellet Melt

436. Melting of the fuel pellet and clad can occur following LB-LOCA when there is insufficient heat removal. The clad has a lower melt temperature than the fuel so it is likely to melt first. The consequences of clad melt are that the fuel loses its coolable geometry and releases radiological material to the coolant.

437. The RP argues that its PCT limit of 1204ºC from 10 CFR 50.46 is below the melting temperature for the fuel and clad. The evidence in Ref. 125 supports this for the AFA 3GAA fuel design. Therefore, the RP argues that because the LB-LOCA fault analysis shows that the PCT limit is met (Ref. 128), the fuel or clad will not melt.

438. I consider that this argument is adequate for GDA and I am satisfied that the RP’s LOCA PCT limit should, if met, prevent clad or fuel pellet melt from occurring. Therefore, I judge that the RP has an adequately justified technical acceptance criterion to address fuel clad or pellet melt in a LOCA.

4.8.1.4 Clad Ballooning and Flow Blockage

439. Fuel clad is at risk of rapid expansion during the LB-LOCA transient due to the internal fuel rod pressure, low coolant pressure and a zirconium alloy phase change that occurs at high clad temperature. This effect has been studied extensively and has been observed to occur in fuel accident conditions. For most fuel cladding materials, temperatures exceeding approximately 800°C will cause the cladding to balloon and often burst (Ref. 84).

440. High circumferential strain due to clad ballooning reduces the available coolant flow area. If the strain is sufficient that contact between adjacent fuel rods occurs, this also reduces the clad surface area available for cooling of those fuel rods that have suffered contact. If multiple fuel rods balloon at the same height or axial plane (‘co-planar’), then it can cause flow starvation downstream. This is often termed a flow

blockage. Severe clad ballooning can therefore challenge the criterion that a coolable geometry is maintained.

441. The extent of clad ballooning depends on the clad temperatures reached in a LOCA. The RP's fault analysis for the LB-LOCA predicts that the PCT is greater than 1100ºC and the core average clad temperature reaches approximately 900ºC (Ref. 128). On this evidence, I judge that the UK HPR1000 is likely to be impacted by clad ballooning in LB-LOCA.

442. The RP's fault analysis for design basis LOCAs, from smaller pipe breaks, shows that the PCT remains well below 800ºC (Ref. 130), the approximate threshold for a zirconium alloy phase change. This means that clad ballooning and flow blockage are unlikely to impact design basis LOCAs.

443. The RP has argued in response to RQ-UKHPR1000-0778, RQ-UKHPR1000-1101 and RQ-UKHPR1000-1630 (Ref. 33) that its LB-LOCA analysis method accounts for clad ballooning and its impact on flow blockage when it calculates the PCT. The RP therefore argues that because the PCT limit and other technical acceptance criteria are met (Ref. 128), a coolable geometry must have been maintained.

444. These arguments are dependent on the adequacy of the RP's analysis method. Therefore, I have assessed the capability of its analysis method using SAPs AV.1 and AV.2. I looked at:

- the V&V evidence for the analysis method; and
- the theoretical and physical basis for the method.

445. The RP's analysis method of the LB-LOCA uses the LOCUST code, for which V&V evidence is reported in Ref. 55. I have observed that this report does not include integral LOCA tests to validate the clad ballooning and flow blockage models. The integral tests that it presents are for unblocked fuel assemblies. I am aware that integrated LOCA clad ballooning and flow blockage experiments have been undertaken in the past (Ref. 84). I consider that the absence of integral LOCA tests for clad ballooning and flow blockage is a weakness in the RP's method validation.

446. Instead of integral tests, the RP has presented V&V data from separate effects tests. These separate effects tests may be appropriate with an adequate justification of the tests' coverage against robust theoretical and physical models. However, I consider that the RP has provided a limited justification for its separate effects tests for clad ballooning and flow blockage. These tests do not demonstrate LOCUST's capability of predicting the LOCA PCT in the presence of flow blockages for a single hot rod model or for complex geometries. The separate effects tests in Ref. 55 are limited to [redacted] for open geometries. I consider that this is a weakness against the expectations of SAP AV.2 for the LOCUST code validation.

447. Following my assessment of the LOCUST code validation for prediction of clad ballooning and blockage, I have assessed the adequacy of its theoretical and physical models. LOCUST uses a simplified model of the core with conservative boundary conditions to analyse the LB-LOCA. This approach focuses on showing that PCT and clad oxidation limits are met for a bounding single hot rod and cooling flow channel. In my opinion the limitations of this approach for analysing clad ballooning are:

- the single hot rod cannot simulate the impact of ballooning and flow blockage for bundles of fuel rods, fuel assemblies or across the core; and
- one-dimensional simulation of clad strain cannot account for two-dimensional effects if adjacent fuel rods touch.

448. Most adjacent fuel rods in the core operate under similar conditions to each other. Thus, fuel rods that are adjacent to fuel rods experiencing clad ballooning are likely to also experience clad ballooning. LOCA experiments have confirmed that this does occur (Ref. 84).

449. LOCUST's limitations mean that it cannot physically predict the expected interactions between adjacent rods that experience clad ballooning. However, the RP may be able to bound these phenomena with a conservative method. Therefore, I have considered whether the clad ballooning and flow blockage model in LOCUST can bound phenomena that it cannot physically simulate.

450. The RP described the LOCUST clad ballooning and flow blockage models in response to RQ-UKHPR1000-1728 and RQ-UKHPR1000-1742 (Ref. 33). The key points of these models are:

- clad ballooning is calculated using a correlation between clad strain and clad temperature, derived from experiments;
- the amount of flow blockage is calculated using a correlation between flow blockage and the predicted clad strain, which accounts for the specific fuel geometry; and
- the flow blockage impact on PCT is calculated by LOCUST's hot rod flow model.

451. The LOCUST clad ballooning model is based on temperature and burst strain correlations from experiments using M5 fuel clad that are reported in Ref. 131. The RP has submitted evidence to demonstrate that LOCUST uses the M5 clad correlation to predict the amount of clad ballooning. I consider that their use of this correlation to predict clad ballooning meets the expectations of AV.1 and AV.2.

452. LOCUST uses an empirical correlation to convert clad ballooning into a ratio of flow area blockage. This correlation is based on early research and assumes that co-planar clad ballooning and flow blockage will not occur. However, state-of-the-art reviews of other experiments have shown occurrences of co-planar blockage (Ref. 84 and Ref. 132). Therefore, I consider that an assumption excluding all co-planar blockage is not valid and the model used by LOCUST lacks sufficient underlying evidence.

453. In summary, I have uncovered a shortfall in LOCUST's predictions of flow blockage due to clad ballooning and a lack of validation data from integral LOCA tests. Accordingly, I am not fully satisfied with the evidence to support the RP's argument that if the PCT limit and other technical acceptance criteria are met, then a coolable geometry must have been maintained.

454. Noting my expectations of reduced conservatism in this analysis when compared to DBA, I have considered whether the RP's thermal analysis using LOCUST in Ref. 128 contains other conservatisms that, if relaxed, may reduce the impact of clad ballooning in the core. In particular, from a reactor core perspective I observed that:

- the analysis makes conservative assumptions about the initial hot rod power and power distributions in the core, accounting for uncertainties in the nuclear analysis methods;
- the analysis uses a conservative decay heat curve based on the US NRC's Appendix K methodology; and
- the analysis uses conservative assumptions for initial core flow rate and the core bypass flow ratio.

455. There may be further conservative assumptions made in the analysis about systems outside of the reactor core, such as the single failure assumption. These are within the scope of the Fault Studies assessment (Ref. 7). All these conservatisms act to

increase the peak temperatures predicted in the current analysis. I therefore consider that a more realistic LB-LOCA analysis may show that most fuel rods in the core do not experience temperatures that lead to clad ballooning, so a coolable geometry may be maintained. However, the licensee should present its own arguments and underlying evidence to substantiate the safety case claim that LB-LOCA will not result in a loss of coolable geometry.

456. I have raised an Assessment Finding to ensure that this matter is addressed by the licensee. The Assessment Finding is described in subsection 4.8.1.6 because I have identified related matters in that subsection.

4.8.1.5 Clad Oxidation

457. The RP's LOCA technical acceptance criteria (Ref. 125) are intended to avoid excessive clad oxidation that could cause brittle fracture of the clad. When Zirconium alloy fuel clad experiences high temperatures and a steam atmosphere it can rapidly oxidise. The reaction rate is increased with temperature and if the temperature exceeds approximately 1200ºC then it becomes self-sustaining. As the amount of clad oxidation increases, it will lead to the fuel clad becoming embrittled. This brittle clad has been observed to shatter when it is quenched during the LB-LOCA reflood phase. The US NRC 10 CFR 50.46 criterion used by the RP is that less than 17% of the initial clad thickness is oxidised.

458. The RP argues that the US NRC 10 CFR 50.46 criteria for PCT and clad oxidation ensure that fuel clad will not be embrittled during the LB-LOCA transient. Therefore, it will not suffer brittle failures during the reflood stage. The report submitted to justify the LOCA technical acceptance criteria (Ref. 125) provides an adequate justification for these two criteria for the UK HPR1000 fuel design. The document explains that in certain scenarios the 10 CFR 50.46 PCT and clad oxidation criteria are non-conservative. As a result, it concludes that the UK HPR1000 safety analysis should use the Baker-Just 'Equivalent Cladding Reacted' correlation to conservatively account for these phenomena and uncertainty.

459. I am satisfied with the demonstration in Ref. 125 that as long as the Baker-Just correlation is used then the RP's criterion for oxidation to prevent embrittlement is substantiated by experiments for M5 clad. In response to RQ-UKHPR1000-1742 (Ref. 33), the RP has confirmed that the oxidation correlation used in its LB-LOCA analysis is Baker-Just.

460. I consider that the RP has provided adequate evidence to demonstrate that the fuel clad will not experience significant embrittlement if the acceptance criterion is met. The RP has confirmed that the oxidation correlation in its fault analysis aligns with recent experimentation. Therefore, I consider the technical acceptance criteria to be adequate for clad oxidation and ensure fuel coolability.

4.8.1.6 Fuel Fragmentation, Relocation and Dispersal

461. FFRD covers a set of fuel phenomena that can occur during a LOCA. The OECD has been co-ordinating research in this area. Its state-of-the-art report details the current understanding of these phenomena (Ref. 133). The topic is also addressed in the US NRC's NUREG 2121 (Ref. 134).

462. I have briefly summarised the relevant three phenomena:

- Fragmentation involves the fuel pellets fragmenting into smaller pieces under irradiation. At higher burnup the fragments become increasingly fine.

- Relocation occurs during a LOCA when the fuel fragments relocate into the ballooned regions of the clad. This can cause local heating that is difficult to simulate.
- Dispersal is a combination of fine fragmentation and bursting clad that can result in a release of fine fuel pellet fragments into the coolant.

463. I recognise that this is an area of developing understanding. Furthermore, I consider that the relatively modest UK HPR1000 burnup limit (sub-section 4.2.1.1) will preclude some of the more onerous aspects of fine fragmentation and fuel pellet dispersal in LB-LOCA. However, I expected the RP to have considered the potential impact of these phenomena on the generic UK HPR1000 design.

464. The RP has not provided a safety submission that covers FFRD. In response to RQ-UKHPR1000-1742 (Ref. 33), it has provided a qualitative argument that the UK HPR1000 burn-up limits ensure that fragmentation and dispersal are unlikely to cause an impact. For relocation, they have carried out sensitivity analysis. This sensitivity analysis shows an increase of between 100ºC and 200ºC PCT in the reflood phase of the LB-LOCA. The RP argues that these results mean that there are adequate margins in its design to account for the phenomena

465. The RP's qualitative arguments associated with FFRD align with my expectations and the approach is equivalent to the approach taken for previous GDA. However, the sensitivity analysis, reported in response to RQ-UKHPR1000-1742, shows an increase in PCT due to fuel relocation and the RP has identified through a review of Ref. 133 that this is an area of active international research. I expect the licensee to develop its justification for these phenomena after GDA as international understanding improves.

466. Clad ballooning and FFRD phenomena are all affected by, and themselves affect, the peak fuel and clad temperatures reached in the LB-LOCA. It is therefore necessary that this matter is addressed together with that discussed in subsection 4.8.1.4. I have raised the following Assessment Finding.

AF-UKHPR1000-0005 – The licensee shall demonstrate that flow blockage due to clad ballooning will not result in a loss of coolable geometry in a large break loss of coolant accident. Fuel fragmentation, relocation and dispersal phenomena should also be addressed within the analysis, or their exclusion justified. Appropriate validation evidence should be provided for all analysis methods used. The analysis may use a reduced level of conservatism compared to that used for faults inside the UK HPR1000 design basis.

4.8.1.7 Fuel System Structural Damage

467. During LB-LOCA the mechanical forces due to the pipe break can challenge the structural integrity of the fuel assemblies. This can lead to direct damage to the fuel that can cause flow blockages and may prevent insertion of the RCCAs. As a result, NS-TAST-GD-075 (Ref. 11) and SSG-52 (Ref. 19) advise that safety cases should contain adequate analysis of the impact of these forces. The objective of this analysis is to demonstrate that any damage does not result in a loss of a coolable geometry or impair RCCA insertion.

468. The RP has presented evidence in its LB-LOCA summary report (Ref. 126) to give confidence that these forces will not prevent RCCA insertion or impair coolability. The analysis in the Ref. 126 uses some inputs which are best estimate. In accordance with SAP FA.15 and the advice in Ref. 11, I am satisfied that a less conservative analysis is adequate for LB-LOCA in UK HPR1000 than would be expected for a design basis fault.

469. Ref. 126 states that the conclusions of mechanical damage analysis for the reference plant, Fangchenggang Unit 3, show that damage is limited to one set of mid-span mixing grids for the peripheral fuel assemblies. These assemblies do not contain RCCAs, so the RP argues that the predicted damage will not impact the ability of the RCCAs to insert. The report also explains the equivalence of the Fangchenggang Unit 3 and UK HPR1000 plants in the context of this LB-LOCA analysis.

470. To demonstrate continued fuel coolability, the RP presented a LOCUST prediction of limiting flow blockage due to mechanical damage to the mid span mixing grid in an assembly at the core periphery and presented sensitivity results that show this blockage will not impact coolability of the fuel in that location.

471. Considering advice in Ref. 11, I judged these arguments to be adequate for GDA. However, I observed that only a summary of the analysis results was submitted for assessment, without the underlying evidence. [redacted] This gave me further confidence that this phenomenon is unlikely to challenge core coolability or RCCA insertion. However, the evidence was not provided specifically for UK HPR1000 and is not part of the GDA safety case. I have raised an Assessment Finding to ensure this matter is addressed by the licensee.

AF-UKHPR1000-0127 – The licensee shall demonstrate that core coolability and Rod Cluster Control Assembly insertion are not challenged due to mechanical damage to UK HPR1000 fuel assemblies and core components in a large break loss of coolant accident. The analysis may use a reduced level of conservatism compared to that used for faults inside the UK HPR1000 design basis.

4.8.1.8 Summary of Assessment Against Fuel Phenomena in a LOCA

472. I have assessed the adequacy of the RP's arguments and evidence to substantiate its claims that its application of the fuel technical acceptance criteria in Ref. 125 will ensure that LB-LOCA does not result in significant fuel degradation or a loss of core coolability. I have considered fuel damaging phenomena that can challenge fuel integrity and/or coolability in LB-LOCA, using guidance in Ref. 11 and Ref. 19.

473. I judge that the RP has adequately substantiated its arguments that the LOCA technical acceptance criteria limiting PCT and clad oxidation will prevent loss of coolability or significant fuel degradation in LB-LOCA due to fuel pellet melt, clad melt, clad oxidation and embrittlement.

474. The RP's technical acceptance criteria include a broad requirement that a coolable geometry be maintained. However, the RP's demonstration that this criterion is met depends on the adequacy of its analysis method, which I have assessed using SAPs AV.1 and AV.2. After considering the RP's arguments and evidence with respect to flow blockage due to clad ballooning, FFRD and fuel assembly structural deformation phenomena, I judge that the RP has not provided sufficient substantiation of its claim that a coolable geometry will be maintained in LB-LOCA.

475. Due to the lower temperatures reached in design basis LOCAs, clad ballooning and FFRD will not have as significant an effect as in the LB-LOCA. On balance, I judge that the technical acceptance criteria applied in design basis LOCAs (Ref. 125) are adequate to prevent a loss of core coolability or significant fuel degradation. I have discussed the mechanical justification of the fuel assembly for the most limiting design basis fault in sub-section 4.4 of this report.

4.8.2 Strengths

476. Following my assessment of the fuel aspects of the UK HPR1000 safety case for LOCA, I have identified the following strength:

- The RP has adequately substantiated its arguments that the LOCA technical acceptance criteria limiting PCT and clad oxidation will prevent loss of coolability or significant fuel degradation in LB-LOCA due to fuel pellet melt, clad melt, clad oxidation and embrittlement.

4.8.3 Outcomes

477. Following my assessment of the fuel aspects of the UK HPR1000 safety case for LOCA, I have identified the following outcomes:

- I have raised an Assessment Finding requiring the licensee to demonstrate that flow blockage due to clad ballooning will not result in a loss of coolable geometry in a LB-LOCA. FFRD phenomena should also be addressed within the analysis, or their exclusion justified. Appropriate validation evidence should be provided for all analysis methods used. The analysis may use a reduced level of conservatism compared to that used for faults inside the UK HPR1000 design basis.
- I have raised an Assessment Finding requiring the licensee to demonstrate that core coolability and RCCA insertion are not challenged due to mechanical damage to UK HPR1000 fuel assemblies and core components in a LB-LOCA. The analysis may use a reduced level of conservatism compared to that used for faults inside the UK HPR1000 design basis.

4.8.4 Conclusion

478. I have assessed the fuel aspects of the UK HPR1000 safety case for LOCA, including the claim that the LOCA technical acceptance criteria in Ref. 125 will prevent a loss of core coolability or significant fuel degradation. I also assessed aspects of the fuel modelling used to demonstrate that these criteria are met.

479. I am satisfied that the LOCA technical acceptance criteria I have assessed are adequate. However, using the AV SAPs and guidance in Ref. 11 and Ref. 19, I observed shortfalls in aspects of the RP's fuel modelling and judge that further work is required to fully substantiate the claim that a coolable geometry will be maintained in the unlikely event of a LB-LOCA. I have raised two Assessment Findings as a result.

480. In collaboration with other inspectors I have reached a judgement that these matters should not prevent issue of a DAC because the LB-LOCA fault is outside of the design basis and because I consider that a more realistic LB-LOCA analysis may allow it to be shown that a coolable geometry will be maintained. ONR's fault studies inspectors have considered the impact of my conclusion on the adequacy of the wider LB-LOCA safety case in Ref. 7 and report that they are content to judge that the safety case for LB-LOCA is adequate for GDA. However, the licensee will have to address these Assessment Findings before ONR can conclude that the overall risks associated with LB-LOCA are reduced ALARP. In sub-section 4.14 of this report I have considered whether my findings have any implications for the core design ALARP demonstration in GDA.

4.9 Fuel Deposits

4.9.1 Assessment

481. Corrosion products or ‘crud’ can deposit on the fuel during powered operation due to chemical and thermal hydraulic effects in the core, particularly in areas with high rates of sub-cooled boiling at the clad surface. As described in NS-TAST-GD-075 (Ref. 11), this has two potential effects on fuel safety:

- In the event that fuel deposits occur with boron trapped in the crud layer, nuclear safety parameters may be affected. Power peaking and neutronic parameters are most affected because of the un-planned reactivity suppression caused by the boron deposited in parts of the core; this phenomenon is known as Crud Induced Power Shift (CIPS). The immediate consequence is reduced safety margin in a variety of faults.
- Clad failure can occur due to enhanced corrosion in the presence of thick deposits of crud. The enhanced corrosion occurs due to a combination of reduced heat transfer and enhanced concentration of chemical species (for example lithium) at the clad surface; this phenomenon is known as Crud Induced Localised Corrosion (CILC). The immediate consequence is one or more localised fuel clad failures and the release of radioactivity in to the coolant.

482. No safety justification for operation with fuel deposits was originally provided for UK HPR1000. RO-UKHPR1000-0015 (Ref. 135) was raised by ONR’s specialist Chemistry inspectors during GDA Step 3 to prompt a demonstration that the risks associated with fuel deposits for UK HPR1000 are ALARP. I sampled the RP’s submissions in response to the RO in detail, to support closure of the RO and gain confidence that the shortfalls had been fully addressed from a Fuel and Core perspective. I have conducted my assessment of this topic in conjunction with the Chemistry inspectors, whose assessment is reported in the Chemistry Assessment Report (Ref. 136).

483. The primary SAPs of relevance to my assessment of this topic are ERC.1 and FA.7. I expect the fundamental safety functions (control of reactivity, removal of heat and confinement of radioactive material) to be delivered with an appropriate degree of confidence in the presence of potential crud deposits. I expect analysis of design basis fault sequences to be performed on a conservative basis, implying a demonstration that the presence of potential crud deposits on the fuel does not compromise fuel or core data supplied for use by that fault analysis.

484. Paragraph 552 of the SAPs under SAP ERC.3 is also relevant, stating “The design of the core and its components should take account of any identified safety-related factors, including…chemical and physical processes”. Ref. 11 advises specifically that inspectors should consider whether limits on the rate of sub-cooled boiling are required to restrict the rate of deposition of crud.

485. Key elements of the actions within RO-UKHPR1000-15 included:

- provision of an estimate of the fuel deposits expected for the different UK HPR1000 core designs including masses, thicknesses and distributions;
- identification of key assumptions and sensitivities in the estimates and provision of suitably robust evidence supporting the analysis;
- provision of a suitable description of the nature of the fuel deposits;
- provision of information on the allowable thermal and boiling parameters in the core designs;
- identification of other operational parameters that may affect crud deposition or behaviour;

- consideration of the impact of crud on fault conditions as well as normal operation; and
- demonstration that adequate measures, including limits and conditions, were in place to reduce the risks associated with fuel deposits ALARP.

486. The scope of my assessment does not include any impact of crud itself on coolant activity or associated source terms because that topic is assessed in Ref. 136. The holistic assessment of the RO, including the judgements made by ONR Chemistry inspectors and myself, is also captured in the RO closure note (Ref. 137).

4.9.1.1 Fuel Deposit Estimates

487. The RP's report Assessment of Fuel Crud for UK HPR1000 (Ref. 54) includes detailed predictions of total crud mass, crud thickness and deposited boron mass distributed around the core. Within Ref. 54 and in response to two RQs (RQ-UKHPR1000-1429 and RQ-UKHPR1000-1510, Ref. 33) the RP has adequately explained the complex trends seen in the results. Ref. 54 also provides a description of the methods used for the analysis, the assumptions made and a series of sensitivity analyses to key inputs.

Methods

488. The method used by the RP to provide quantitative fuel crud predictions involves a code called CAMPSIS. Validation evidence for CAMPSIS has been assessed by Chemistry inspectors in Ref. 136. CAMPSIS requires thermal hydraulic parameter inputs that are generated using the LINDEN sub-channel thermal hydraulic code. LINDEN itself requires power distribution inputs that are generated using the PINE and COCO nuclear design codes for the different core cycle designs as a function of fuel burnup.

489. My assessment of the validity of the LINDEN, PINE and COCO codes is captured in subsection 4.13 of this report. Although the validation evidence provided for LINDEN is focused on its application in sub-channel fault analysis and prediction of margin to CHF, I judge that it also supports the application of LINDEN in providing thermal hydraulic parameters to CAMPSIS for generation of fuel crud predictions.

Thermal Hydraulic Analysis and Sensitivity Studies

490. Ref. 54 references a supporting report on thermal and boiling parameters for UK HPR1000 (Ref. 35) for presentation of the key thermal hydraulic inputs to the CAMPSIS analysis. To enable a robust estimate of the fuel deposits expected for UK HPR1000 and the consequent risk of CIPS occurring in particular, it is my opinion that the thermal hydraulic analysis in Ref. 35 and Ref. 54 should constitute a reasonable best estimate analysis of relevant parameters (evaporation rates, boiling area) and not contain unnecessary conservatism that could produce misleading results. The SAP I consider most important to this particular analysis is AV.6, which states: "Studies should be carried out to determine the sensitivity of the analysis (and the conclusions drawn from it) to the assumptions made, the data used and the methods of calculation."

491. Ref. 35 presents sensitivity studies to a range of important inputs to the thermal hydraulic analysis and assumptions within the method, which affect the predictions of maximum local evaporation rate, total evaporation rate and total boiling area in the core, and therefore the fuel crud predictions. I am satisfied that the sensitivity analyses shown to variation in core power, mass flow rate, primary pressure and core inlet temperature have considered the correct uncertainties (consistent with those assumed in fault analysis) and have also been carried though to sensitivity studies in Ref. 54 showing the effect on actual crud deposition. Ref. 54 shows from these sensitivity studies that the crud mass deposited goes up with increased power, reduced flow,

reduced pressure or increased average temperature, but the maximum crud thickness shows the opposite tendency. This is because a greater area of the core starts to contain boiling and therefore the available corrosion products are spread more thinly. The sensitivity to these parameters is significant, which in my opinion highlights the need for an adequate surveillance scheme during operation.

492. Ref. 35 also presents sensitivity analyses to key variables within the LINDEN analysis for this application, including the turbulent mixing coefficient, the fidelity of modelling used (number of channels per fuel assembly) and the severity of axial and radial power distributions. The sensitivity to turbulent mixing coefficient and the fidelity of modelling used is relatively small compared to the parameters above, with the exception that modelling a greater number of sub-channels within the hottest fuel assembly does affect the predictions of local maximum evaporation rate. The RP argue that the local maximum mass evaporation rate on the fuel rod surface is not a key parameter for fuel crud analysis concerning the CIPS risk assessment and therefore do not carry this sensitivity analysis forward to Ref. 54. I am satisfied with this for the CIPS risk assessment, but I judge that it means the predictions of local peak crud thickness may be non-conservative for the purpose of CILC risk assessment. The RP has instead used OpEx to support its assessment of CILC risk; this is discussed in subsection 4.9.1.2.

493. The sensitivity analyses to conservative axial and radial power distributions show that they can have a large effect on the evaporation rates and boiling area. The RP has not carried these results forward to the analysis in Ref. 54 either and I agree with its assertion that to do so would be too conservative (and potentially misleading) for use in the fuel crud analysis. However, these power distributions are theoretically allowed by the UK HPR1000 operating rules and it is therefore clear that in some allowed operating conditions, the maximum and total evaporation rates are likely to be greater than those used in Ref. 54. What this means is that the behaviour of fuel deposits during operation will depend not only on the operating limits set to ensure compliance with the safety case, but also on the way in which the core is operated within those limits, such as the frequency of flexible operations undertaken or the particular R bank insertion used. I again judge that the most appropriate way to address this issue is through an adequate surveillance scheme in operation.

494. Overall, I am satisfied that the thermal hydraulic analysis presented in Ref. 35 is adequate to provide inputs to the fuel crud analysis in Ref. 54 and the sensitivity studies performed are sufficient to properly inform the conclusions of that work. The high sensitivity to some parameters raises the importance of an adequate surveillance scheme associated with fuel crud. I also observe that the maximum predicted crud thickness in UK HPR1000 occurs in the first cycle due to the initial higher corrosion product release rate and I judge that any such surveillance scheme should therefore include visual surveillance of fuel predicted to see the highest levels of crud deposition at the end of the first operating cycle. The topic of surveillance schemes is discussed in subsection 4.9.1.4.

Supporting Evidence from the CPR1000 Fleet

495. Ref. 54 also presents some OpEx data from the CPR1000 fleet in China and, by also generating fuel crud predictions for a CPR1000 plant using the UK HPR1000 methods, aims to provide some additional confidence in the conclusions drawn from the work.

496. The CPR1000 reactor core is smaller than the UK HPR1000 reactor core, with a very slightly higher nominal average coolant temperature and power density. Evaporation rates will therefore be slightly higher. Together with some differences in the primary circuit water chemistry between the two plant designs, intuitively, I would expect this to result in higher fuel crud deposition rates in CPR1000. The results presented in Ref. 54

bear this out, indicating that total crud mass, maximum crud thickness, deposited boron mass and deposited boron-10 mass are all predicted to be lower for UK HPR1000 than for CPR1000 with an 18-month fuel cycle. Furthermore, the RP states that the CIPS phenomenon has never been observed during CPR1000 operation and that the CIPS risk of UK HPR1000 is foreseen to be lower due to the nature of these fuel crud predictions. Ref. 54 presents evidence in the form of axial offset measurements taken on a wide range of CPR1000 plants, compared against the predicted axial offset values, to show that those plants have not experienced problems due to CIPS. I am satisfied that the axial offset discrepancies shown are bounded by margins catered for within safety analyses and that they provide a reasonable demonstration that CPR1000 plants have not suffered from CIPS. From this analysis I take some confidence that provided the plant is operated appropriately, CIPS should not occur for UK HPR1000; the topic is further discussed in subsection 4.9.1.3 of this report.

4.9.1.2 Application of OpEx

Nature of Fuel Deposits

497. The RP has submitted a report entitled Status on Crud Monitoring and Acceptability (Ref. 138), which provides a short description of the origin of crud, its potential impact on fuel rods and assemblies, its physical characteristics and how it can be influenced by the water chemistry. Ref. 138 then provides a ranking for different categories of crud seen on operating plants to help evaluate what crud formation is acceptable during operation. Different crud types are categorised as ‘Light’, ‘Medium’ or ‘Heavy’ and their nature is briefly described. This is supported by OpEx including photographs of observed crud deposition. The OpEx includes three examples of plants with “heavy crud” and for two of these, statements are made in Ref. 138 that imply the clad corrosion behaviour has been affected by the crud layer. Ref. 138 identifies the need for a surveillance scheme for fuel crud and recommends actions to be taken should each category of crud be observed. These actions appear to be proportionate to the potential severity of the crud’s impact. Ref. 138 states that observation of heavy crud requires immediate actions in order to mitigate the impact on the fuel cladding.

498. I am satisfied that the three categories of fuel crud described are consistent with the OpEx and sufficiently differentiated to enable an adequate surveillance scheme to be developed for UK HPR1000.

Identification of Key Operating Parameters and Qualitative Assessment of Risk

499. Although Ref. 54 presents quantified predictions of fuel crud deposition as described previously, it also explains that due to the uncertainties associated with these predictions, particularly for a new reactor design, the RP’s main assessment of fuel crud risk is reached qualitatively. This is achieved by identifying key operational parameters that may affect crud deposition or behaviour, undertaking a review of available international OpEx with fuel crud and comparing key UK HPR1000 plant parameters with the equivalent parameters for those plants for which OpEx has been reviewed. The quantified estimates of fuel crud support the OpEx analysis and allow for additional sensitivity assessments, as described previously. I judge that this strategy is sensible in order to provide a robust overall assessment of fuel crud risk, despite the uncertainties involved.

500. Ref. 54 describes the key parameters impacting crud formation in three groups: core design parameters, materials choices and chemical control (the latter two of which have been assessed by ONR’s Chemistry inspectors). From a core design perspective, Ref. 54 identifies the key parameters to be those impacting the evaporation rates and boiling area, because the presence of sub-cooled boiling (or, in areas of pre-existing crud deposition, wick boiling) enhances crud deposition rate. I am satisfied with the

approach taken. The conclusion that evaporation rate and boiling area are the most important factors is also consistent with guidance in NS-TAST-GD-075 (Ref. 11).

501. The OpEx detailed in Ref. 54 is further supplemented by data submitted by Framatome from its own international crud OpEx in Ref. 139. I am satisfied that Ref. 139 supports the key arguments made about OpEx in Ref. 54.

502. The RP's review of OpEx in Ref. 54 concludes with an expectation that the crud level of the UK HPR1000 will be lighter than most US PWR plants and similar to, or a little lower than the European (French / German) PWR and Chinese CPR1000 plants with the current core design, material choices and chemistry control. Ref. 54 states that in accordance with the categories of crud described in Ref. 138, light crud is expected in the UK HPR1000 plant. After assessing this topic in conjunction with Chemistry inspectors, my opinion is that the OpEx presented in Ref. 54 and Ref. 139 supports that conclusion.

4.9.1.3 Consequences of Predicted Fuel Crud Deposition for UK HPR1000

503. The conclusion of the OpEx review in Ref. 54 implies that the CILC phenomenon will not occur in the UK HPR1000 plant because only light crud is expected to be observed. The quantitative predictions of maximum crud thickness and associated sensitivity analyses suggest a relatively modest maximum crud thickness, so also support this conclusion. For the purposes of a generic safety case, I am satisfied with this position, but judge that an adequate surveillance scheme is necessary due to the high sensitivity of maximum crud thickness to operating parameters and modelling assumptions.

504. Within Ref. 54, the RP has evaluated the risk of CIPS occurring for UK HPR1000 by using the deposited mass of boron-10 as a surrogate indicator of the risk, rather than undertaking specific nuclear calculations to predict modified power shapes. I am satisfied this is adequate during GDA because of the dominance of the boron-10 neutron absorption cross-section when compared to those of other isotopes commonly found in fuel crud. As described previously in paragraph 497, Ref. 54 also shows that the deposited boron-10 mass is predicted to be lower for UK HPR1000 than CPR1000 plants and presents evidence to show that CPR1000 plants have not experienced problems due to CIPS. When combined with suitable surveillance schemes (see subsection 4.9.1.4), I judge this data provides appropriate confidence that CIPS will not occur for UK HPR1000.

505. Ref. 54 presents an assessment of the impact of fuel crud deposition on fault tolerance and the conservatism of the existing fault analysis. Arguments are provided to explain why key neutronic and kinetic parameters are either unaffected, or are only changed by very small amounts within the existing uncertainty bands. Explicit calculations have been conducted to show that fault analysis assumptions about SDM and the integral RCCA negative reactivity insertion curve are bounding of the effect that fuel crud could have, even if the axial offset moves to the extremes of the allowable operating zone due to a significant CIPS event. I judge this adequate to show that the neutronic and kinetic performance of the core in faults will not be significantly affected by the presence of crud.

506. Ref. 54 also presents an assessment of the impact of crud deposition on the consequences of individual faults due to changes in axial power distribution and possible increased thermal resistance, which primarily impacts the peak fuel and clad temperatures. The RP has used assumptions for crud deposition thickness based on its own CAMPSIS results and for crud conductivity from referenced sources. I am satisfied these parameters together should provide for a sufficiently conservative output without being overly onerous in the assumptions about the crud layer that might exist. The RP has presented a mixture of qualitative and quantitative studies for

individual faults, and Ref. 54 concludes that the impact of crud does not challenge the acceptance criteria.

507. For most faults a qualitative explanation is provided as to why the results are not sensitive whereas for a limited selection of faults, quantitative analysis is provided to demonstrate that although the results are affected, adequate margin still exists to fault acceptance criteria.

508. In the case of an RCCA ejection accident, quantitative analysis using the same assumptions as in the transient analysis report (Ref. 41) shows that if the fault occurs with crud pre-deposited on the limiting fuel rod, the peak fuel temperature criterion may be breached, and some limited fuel melting may occur. However, the RP has shown the results of sensitivity analyses in Ref. 54 to demonstrate that the fuel temperature limit is met if the initial fuel temperature assumed is slightly reduced. Arguments have been provided by the RP in response to RQ-UKHPR1000-1656 (Ref. 33) as to why this remains a conservative approach, stating that the lower initial temperature assumption in the sensitivity analysis still accounts for uncertainties and that only ‘additional provision’ has been reduced. This does not imply a change to the design or any actual operating parameters. I also observe that within the ALARP assessment for DNB analysis (Ref. 37), as discussed in subsection 4.14 of this report, the RP has been able to show that using a cycle-by-cycle set of core analysis assumptions also reduces the predicted consequences of an RCCA ejection accident, which is not taken credit for in Ref. 54. Overall, I am satisfied from a Fuel and Core perspective that the RP’s prediction that the fuel temperature limit will be met in an RCCA ejection accident remains conservative, and that my assessment of the ALARP case for this fault, discussed in subsection 4.14, is not changed by the potential presence of crud on the fuel.

4.9.1.4 Surveillance Schemes for Fuel Deposits

509. As discussed previously, due to the sensitivity of fuel crud predictions to the way in which the plant is operated within its operating rules, I expected an adequate surveillance scheme to be defined for UK HPR1000. Ref. 11 advises that “The licensee should undertake surveillance programmes to ensure that the state of the cladding is consistent with safety case assumptions. For fuel deposits, OpEx shows that surveillance of fuel assemblies can be appropriate even directly after the first operating cycle to verify the expected behaviour.” I therefore expect that the defined surveillance scheme should verify that fuel crud behaviour is similar to that predicted and initiate corrective action if certain categories of crud are observed, in accordance with Ref. 138. Due to the prediction in Ref. 54 that maximum crud thickness will occur during the first cycle of operation, I expect that this surveillance scheme should begin during the refuelling outage after that first cycle.

510. Ref. 54 states that four categories of surveillance will be undertaken for the UK HPR1000 that are associated with fuel crud: chemistry sampling and monitoring, axial offset monitoring, visual inspection of fuel and monitoring of core outlet temperatures.

511. From a Fuel and Core perspective, the axial offset monitoring is important to give early indication of a CIPS problem developing and I judge the proposed monitoring regime for UK HPR1000 to be adequate. In addition, the visual inspection of fuel is important to meet the objectives identified in paragraph 510 and therefore further reduce the risk of CILC occurring. Ref. 54 states that the visual surveillance of fuel to identify the presence of fuel crud will be undertaken on UK HPR1000 to confirm the expected crud deposition behaviour, during the core unload and re-load process. No details are given about the fuel assemblies to be inspected, nor whether these inspections will begin in the first outage. I judge this acceptable for GDA, as this is a matter to be addressed by the licensee as part of normal business. However, the safety case requirement for a

surveillance scheme should have been made clearer with the PCSR Chapter 5 documentation. This is not the only shortfall I have identified of this type, so in subsection 4.12 I have listed all those I have identified and recommended a means by which they can be addressed.

4.9.2 Strengths

512. Following my assessment of the UK HPR1000 safety case for fuel deposits I have identified the following strengths:

- The RP has developed quantitative estimates of fuel crud deposits for UK HPR1000 and used these, together with sensitivity analyses and international OpEx data, to provide a robust assessment of the range of fuel crud deposits that could occur for UK HPR1000.
- The RP has provided sufficient confidence that CIPS and CILC phenomena will not prevent the three fundamental safety functions (control of reactivity, removal of heat and confinement of radioactive material) from being met for the UK HPR1000 plant.

4.9.3 Outcomes

513. Following my assessment of the UK HPR1000 safety case for fuel deposits I have not identified any Assessment Findings. The development of a detailed fuel crud surveillance scheme is important, but I judge it to be a matter that can be addressed by the licensee as part of normal business.

4.9.4 Conclusion

514. Based on the outcome of my assessment, I have concluded that the expectations set by SAPs ERC.1, ERC.3, FA.7 and by guidance in NS-TAST-GD-075 (Ref. 11) are met in the context of fuel deposits. After undertaking my assessment of this topic in conjunction with Chemistry inspectors whose work is reported in Ref. 136, I am satisfied with the demonstration provided by the RP that the fuel and core related risks associated with fuel deposits for UK HPR1000 are ALARP.

4.10 Core Mis-loading and In-Core Neutron Detectors

4.10.1 Assessment

4.10.1.1 Risks associated with core mis-loading faults

515. Core mis-loads may have the potential to result in criticality prior to start-up, by placing more reactive fuel assemblies closer together in the core. They also have the potential to increase localised power peaking once the plant has returned to powered operation, thereby reducing safety margin in faults.

516. Due to the potential for high consequences in an uncontrolled criticality, I have sampled the RP’s safety case associated with core mis-loading. In accordance with SAP ECR.1 and advice in NS-TAST-GD-075 (Ref. 11) I expect that sufficient controls are in place to mitigate the risk of inadvertent criticality as fuel is loaded into the core. As an associated part of the safety case I have also sampled the fault analysis related to a mis-loading of the core that is undetected before return to full power. Ref. 11 advises that LWRs may require additional limits to provide sufficient margin to CHF, including margin for undetected core mis-loadings.

Risk of Inadvertent Criticality

517. Boron concentrations required to meet the UK HPR1000 nuclear design basis limits on neutron multiplication factor when shut down and during refuelling are presented in the nuclear design reports for first and equilibrium cycle cores (Ref. 27 and Ref. 28 respectively). During refuelling the concentration specified is high and this provides additional criticality safety margin in the case of a core mis-load. The Loading Sequence Analysis of Cycle 1 (Ref. 46) provides evidence that:

- the sensitivity of neutron flux monitoring equipment will be adequate to detect significant changes in flux throughout the core loading process; and
- even in a worst-case mis-loading scenario, the UK HPR1000 core will remain sub-critical and therefore an uncontrolled criticality cannot occur.

518. In my opinion this latter point is a strength of the design. Due to the high boron concentration during refuelling, a mis-load will not cause an inadvertent criticality. The analysis used to provide evidence of this in Ref. 46 assumes that the most reactive fuel assemblies in Cycle 1 are all placed together in the centre of the core. I consider this to be a very pessimistic assumption and therefore I am satisfied the analysis conclusions are conservative. As a result, I judge that the UK HPR1000 core design, given the correct boron concentration, is inherently safe against the hazard of uncontrolled criticality due to a core mis-loading fault because it cannot occur, in alignment with the principle of SAP EKP.1. Although evidence is only provided for the first cycle core design, having reviewed beginning of life reactivity data for different fuel assembly types presented in response to RQ-UKHPR1000-0574 (Ref. 33) I am satisfied that the most reactive fuel assemblies in the equilibrium cycle are no more reactive at BOC than those in the first cycle. I therefore take confidence that this conclusion can be read across to other cycles without requiring further evidence in GDA.

519. Safety measures are described to reduce the risks associated with a core mis-load in Ref. 46:

- underwater photography in the fuel pool before fuel loading and in the reactor core afterwards to verify that the correct fuel assembly is in each location, using an ID marked on each fuel assembly;
- isolation of systems with lower boron concentration than the RCS; and
- neutron flux monitoring in several different core locations using two different systems, with alarms and operating actions triggered by certain changes in neutron flux count rate.

520. These measures provide some additional defence in depth. Due to the inherent safety of the design I decided it would not be proportionate to request further detailed evidence underlying these measures during GDA for the purposes of my assessment of core mis-load faults. However, control of boron concentration is clearly important for the UK HPR1000 during refuel, as it is for other PWRs. Separate safety analyses have been provided to demonstrate that the risks associated with various boron dilution faults are ALARP. These have been assessed by Fault Studies specialists in Ref. 7. Chemistry specialists have assessed the control of boron concentration in normal operations in Ref. 136.

Risk of Fuel Damage due to Operating at Full Power with an Undetected Mis-load

521. In conjunction with ONR Fault Studies specialists, I also sampled the RP’s analysis of the consequences of an undetected core mis-load occurring that could disturb the power distribution following a return to power. In this case, “undetected” means a mis-load that would cause a power distribution discrepancy too small to be reliably observed using the in-core SPNDs at start-up. The consequences of more severe mis-

loads failing to be observed is considered within the categorisation of the in-core instrumentation functions, which I assess in subsection 4.10.1.2 of this report.

522. The fault analysis report submitted by the RP for an undetected mis-load (Ref. 140) shows that no fuel failures should occur directly as a result of the fault, with safety margin to the DNBR design limit reduced, but still positive. This conclusion is reached by calculating the limiting F_{dH} that would cause DNB to occur in steady state operation and then using the nuclear design codes to show that feasible undetected mis-loads will not cause F_{dH} to reach that magnitude, for each cycle. However, no account is made for the consequences if another fault were to occur during the operating cycle in which this undetected mis-load occurred. Due to the mis-load being undetected, appropriate recovery actions may not be taken and the fault tolerance of the plant could be reduced throughout the following operating cycle of approximately 18 months. I explained my expectations to the RP during GDA Step 4 that a safety case be presented to show that the risks associated with this potential fault sequence were ALARP, informed by SAPs EKP.2, ERC.4 and the FA range of SAPs. The RP subsequently submitted an additional document entitled Assessment of Core Misloading Preventing Strategy (Ref. 141) to present a more comprehensive safety justification and show that the relevant acceptance criteria will not be exceeded if a fault occurs in the cycle following an undetected mis-load.

523. Ref. 141 makes clear reference to other parts of the safety case such as Ref. 140 and Ref. 46 in order to present a holistic safety case for this topic, which I consider to be good practice.

524. The RP has calculated the IEF of a core mis-load, at the point that the RPV is closed and before physics tests are ready to begin, to be [redacted] per year. ONR's PSA inspector has confirmed they are satisfied with the way in which the IEF has been calculated. Measures have been identified in Ref. 141 by which a mis-load could be identified before reaching full power. Noting SAP ERC.4, I judge that the RP has taken all reasonably practicable steps to enable the detection of core mis-loads and allow appropriate recovery actions to be taken.

525. To enable a prediction of the frequency for a fault sequence in which a fault occurs in the operating cycle after an undetected mis-load, the predicted frequency of an undetected mis-load was necessary. To quantify this in Ref. 141, the RP have reduced the total core mis-loading IEF of [redacted] to [redacted] The RP has predicted that out of 519 total mis-load cases considered for the cycle 1 core, only [redacted] would be undetected in practice. On this basis I consider the factor of [redacted] reduction in frequency to be conservative. The RP's prediction takes credit for the detection of binary swap mis-loads that are predicted to cause a power discrepancy of greater than 10% from the nominal in a single fuel assembly. I judge this is also conservative on the basis of evidence provided in response to RQ-UKHPR1000-0914 (Ref. 33) that mis-loads causing discrepancies of this magnitude or greater will always result in significant asymmetry in flux rates measured at the opposite SPND locations.

526. Supported by Fault Studies inspectors I have assessed the way in which fault sequences have been developed in Ref. 141. The RP has assumed in Ref. 141 that a fault will occur during the 18-month operating cycle subsequent to the undetected mis-load, calculated the combined event sequence frequency and determined the appropriate fault acceptance criteria in accordance with its own design basis rules. All potential design basis faults occurring at power have been considered, resulting in a wide range of sequence frequencies. I am satisfied that the range of acceptance criteria applied by the RP to the selected fault sequences is adequate. Some of the sequences have frequencies below 10^{-7} per year, which is the typical design basis cut-off frequency suggested by paragraph 631 of the SAPs. Considering guidance

applicable to DEC-A conditions in Ref. 11, my expectations were that the RP's analysis of these fault sequences in Ref. 141 may be less conservative than their analysis of more frequent faults.

527. Three different mis-load cases predicted to be undetectable have been selected from the equilibrium cycle for analysis: the one causing the maximum F_{dH}, the one causing the maximum F_Q and the one causing the maximum power discrepancy in a single fuel assembly. Each of these three cases has been analysed in turn to determine the impact on fault progression. I am satisfied this is a conservative approach to selection of mis-load cases for analysis.

528. Five reactivity faults are considered that require the generation of fault-specific neutronic data. For these faults, the RP has presented the results of physics calculations to show that the main neutronic parameters driving the fault progression, accounting for each of the three selected mis-load cases discussed in paragraph 528, are still bounded by the neutronic parameters used in the analysis of the same faults without a core mis-load. For the two frequent faults considered, this demonstration has been made on a conservative basis. For three infrequent faults, the RP has removed some uncertainty allowances from the nuclear analysis to allow the same demonstration to be made. I judged this to be adequate due to the very low fault sequence frequencies, as per paragraph 527.

529. The existing transient analyses for all the other faults considered in Ref. 141 use neutronic data from the general nuclear data report (Ref. 48). The RP has demonstrated in Ref. 141 that this data is still bounding for cores with the three selected mis-load cases discussed in paragraph 528. The RP therefore claims that the predicted consequences of all these faults are unaffected by an undetected core mis-load. I was largely satisfied with these arguments. The RP identified a need to measure F_{dH} during flux-mapping exercises at start-up and verify that it is below a certain value in order for its arguments to be validated. I was not fully convinced that the F_{dH} criterion the RP proposed for start-up accounted for sufficient uncertainty, but I also observed that the three undetected mis-load cases analysed were all predicted to meet this criterion with wide margin. I see the refinement and justification of physics test criteria as an activity for the licensee.

530. I therefore judge that Ref. 141 presents an adequate demonstration of continued fault tolerance following an undetected core mis-load, meeting expectations for this topic that I have derived from SAPs EKP.2 and FA.4. I consider the overall outcome to be a strength of the safety case.

4.10.1.2 In-core instrumentation

Background and Expectations

531. The UK HPR1000 plant has both in-core and ex-core neutron flux detectors. Both systems provide measurements of neutron flux but the functions for which such measurements are required are different. The ex-core neutron detectors provide continuous monitoring of core power, axial power distribution (axial offset) and neutron flux rate of change in all core operating states using a combination of source range, intermediate range and power range detectors arrayed around the core. These measurements are used for various monitoring and alarm functions in the control room and automatic protection functions within the RPS, including reactor trip. The ex-core detector system is classified F-SC1. I view the functional requirements of the ex-core detectors (Ref. 142) to be similar to those for equivalent systems in other PWRs and the safety classification meets my expectations for a system of this type.

532. The in-core neutron flux detection system (part of the in-core instrumentation system, whose functional requirements are described by the RP in Ref. 143) has been newly

developed for the HPR1000 design and was originally assigned a classification of “NC” (not safety classified). I considered that this classification could potentially lead to inadequate control over the design or through-life management of the equipment (for example, potential exclusion from the plant maintenance schedule), so elected to sample the functional requirements for the system and the evidence underlying the system classification in some depth.

533. The system is described by the RP in Ref. 144 along with other in-core instruments (thermocouples and RPV level indications). It makes use of rhodium-based SPNDs. These are placed in the central tubes of 42 different fuel assemblies, in locations spread evenly and symmetrically throughout the core. In each of the 42 fuel assemblies, 7 SPNDs are stacked on top of each other covering the core height. In this way, a full 3D neutron flux distribution map of the core can be generated. Rhodium detectors provide a slightly delayed signal and Ref. 143 requires that the system output be updated at intervals of approximately 60 seconds during operation.

534. The SPNDs are each allocated to one of four groups. The wiring from every SPND in a group is combined in the upper plenum to exit through a common penetration in the RPV head. The signals from each SPND in a group are processed in to digital data by a common SPND cabinet. There are therefore four RPV head penetrations and four SPND cabinets, one for each SPND group. The digital signals from the SPND cabinets are transferred to a core surveillance cabinet where they are processed in to data suitable for use by operators. Another off-line computer back-up is available should the core surveillance cabinet fail. I hereafter refer to all of this equipment as the SPND sub-system.

535. The RP has submitted a generic process for categorisation of safety functions and classification of safety systems for UK HPR1000 in Ref. 145. SAP ECS.1 and the associated guidance (Ref. 2) sets expectations for such a process. The RP’s process has been assessed by Fault Studies specialists in Ref. 7 and determined to be adequate. My initial expectation in this assessment was therefore that in undertaking the safety function categorisation process for the SPND sub-system, the RP should have followed its own process for the UK HPR1000 project in Ref. 145.

536. Additionally, the ONR TAG for Categorisation and Classification, NS-TAST-GD-094 (Ref. 16) states that the classification of an SSC should consider the potential for a failure to initiate a fault or exacerbate the consequences of an existing fault, including situations where the failure affects the performance of another SSC. With that in mind, because the SPND sub-system is assigned a lower classification than the main RPS, I also sought confidence that:

- failure of the SPND sub-system cannot credibly result in operating the plant in a condition which would require a higher classification protection system to act to prevent fuel damage but in which that system does not act because the usual trip set-points have not been reached; and
- failure of the SPND sub-system cannot degrade the function of a higher classification protection system to the extent that a substantial increase in fuel damage or other identified consequences is credible in a fault.

537. I also expect that if the work completed shows that there is an impact on safety due to a postulated failure of the SPND sub-system, then a demonstration should be provided that the risks have been reduced ALARP.

Functional Categorisation

538. The RP submitted a Design Modification during GDA Step 4 (Ref. 146) to upgrade the categorisation of the SPND sub-system functions from NC to FC-3 (leading to a sub-system classification of F-SC3) within the framework of Ref. 145. The RP’s re-

assessment of the functions of the SPND sub-system and their safety categorisation is reported in Ref. 143. From Ref. 143 I interpret the main SPND sub-system functions to be:

- core monitoring (to identify any anomalous behaviour that occurs during operation);
- core prediction (to predict how core parameters will change for operators);
- flux mapping (a systematic comparison of the neutron flux maps with predictions, undertaken both at start-up and periodically at power); and
- calibration of ex-core detectors.

539. Specifically, the core monitoring and prediction functions are categorised NC whereas flux mapping (including for core mis-load detection, as discussed in subsection 4.10.1.1) and ex-core detector calibration functions are now classified FC-3. According to Ref. 146, the upgrade in categorisation of the system functions will result in the SPND cabinet undergoing additional qualification tests and the SPND sub-system software being designed to the relevant international standard for the new categorisation.

540. The RP has provided a justification for the newly assigned function categorisations in a new SPND categorisation report (Ref. 147), supported by a high-level Failure Modes and Effects Analysis (FMEA) for the SPND sub-system in Ref. 148. C&I Inspectors have raised Assessment Finding AF-UKHPR1000-0026 in Ref. 149 requiring the licensee to undertake comprehensive reliability analyses for UK HPR1000 C&I systems due to the lack of properly underpinned reliability studies in GDA. I have agreed with C&I inspectors that the FMEA provided for the SPND sub-system in Ref. 148 should fall within the scope of that Assessment Finding. However, I have been able to reach judgements about the adequacy of the SPND sub-system safety classification in GDA by assessing Ref. 147 and considering the consequences of failures at a functional level. As such, any shortfalls within Ref. 148 do not affect the outcome of my assessment.

541. Ref. 147 provides a justification for the categorisation of each of the four identified SPND sub-system functions in turn. The RP argues that the continuous core monitoring function and the prediction function are auxiliary functions and not safety functions because the monitoring signals are not used to trigger any protection or mitigation actions. I am satisfied this is true because all neutron flux signals used by the RPS at power are provided by the ex-core neutron detectors. The SPND sub-system does not therefore have the potential to degrade the function of those systems, or allow the core to operate outside of the envelope prescribed in the operating rules, through an unrevealed failure.

542. For the ex-core detector calibration function, the RP states in Ref. 147 that the SPND sub-system provides monitoring of the state of an FC-1 function (i.e., those functions fulfilled by the ex-core detectors) and that in accordance with Ref. 145 should therefore be classified FC-3. Ref. 147 also makes clear there is no hard-wired or network interface between the SPND sub-system and ex-core detectors. I have confirmed that Ref. 145 Appendix A does indeed propose a category of FC-3 for such a monitoring function. In considering my further expectations outlined above and reaching my judgement as to the adequacy of an FC-3 category for this specific sub-system function, I have also accounted for the following additional factors:

- the periodicity with which the ex-core detectors require re-calibration should be known to the licensee and set down in operating rules;
- the calibration activity need not be undertaken under time pressures and if unsuccessful then Ref. 147 states that power will be reduced to limit the impact of faults until the calibration failure can be rectified; and

- according to Ref. 147, in advance of the calibration activity being undertaken, a physical check of SPND insulator resistance is undertaken and a "usability test" is specified to verify that the SPND sub-system is operating correctly. This includes a check on signal stability from each SPND and a check that the signals from SPNDs in symmetrical positions in the core do not deviate from each other by more than a maximum of 10%.

543. In my view the described usability test provides significant additional assurance in the output of the SPND sub-system for the purposes of the calibration function. The layout of SPNDs in the core presented in Ref. 148 shows that for every detector in the core, there is another (or in most cases three others) that are in symmetrical positions and whose signals are processed through different SPND groups. For an erroneous SPND signal to not be discovered during the usability test, would require multiple SPNDs or their associated groups' SPND cabinets to have failed in similar ways at the same time such that the symmetry requirement was still met, or for the software to have failed such that the erroneous signal was not made visible to operators. The SPND symmetry and grouping means that the system contains a degree of redundancy for this application, in accordance with SAP EDR.2. Further, if such a failure had occurred, it would be very likely to have already been noticed by operators when using the SPND sub-system for core monitoring and/or flux mapping functions, because it would produce an anomaly in the indicated core power distribution.

544. In my judgement, with these additional measures in place, failure of the SPND sub-system to fulfil its calibration function is extremely unlikely to degrade the function of the higher classification RPS to the extent that a substantial increase in fuel damage or other identified consequences is credible in a fault. I am satisfied that the FC-3 function categorisation reached by the RP through application of the process in Ref. 145 is adequate.

545. For the flux mapping function, the RP has applied the process within Ref. 145 to determine the appropriate safety category for the function in Ref. 147, in the event that it is required to correctly identify a core mis-load fault before full power operation is reached. Within the framework of Ref. 145, a severity is assigned using the calculated IEF for a core mis-load and the conservatively predicted consequences if it went undetected. The consequences predicted due to occurrence of the worst identified binary swap mis-load are that [redacted]% of fuel rods in the core could enter DNB at full power; radiological consequences are predicted in Ref. 147 as a result. Using Ref. 145, the RP conclude that the function should be categorised FC-3.

546. I am not satisfied that the straightforward application of the Ref. 145 process in Ref. 147 has been done correctly because procedural checks during core load and core photography have both been claimed in the calculation of the IEF and then claimed again as alternative protective measures to the SPND sub-system in the function categorisation process. However, in reaching my judgement as to the adequacy of an FC-3 category for this specific SPND sub-system function, I have accounted for the same factors listed in paragraph 543. In addition:

- during start-up the flux mapping is to be undertaken [redacted], providing multiple opportunities to detect a mis-load;
- the calculated IEF accounts for any binary swap of two fuel assemblies in the core occurring and being undetected before the start-up process begins. However, only up to approximately [redacted]% of such mis-loads are predicted to be severe enough to cause a loss of fuel integrity due to DNB upon full power being reached, according to the response to RQ-UKHPR1000-1462 (Ref. 33). The true IEF of a core mis-load that causes the consequences described above is therefore substantially lower than the IEF assumed in Ref. 147; and

- Ref. 147 identifies other parameters measured during start-up physics tests that may identify a core mis-load independently of the SPND sub-system. In my view these could not be claimed as highly reliable measures in their own right for this purpose, but they would be more likely to detect a mis-load that caused a larger power distribution anomaly and therefore do provide some additional diversity.

547. For similar reasons to those described for the calibration function, in my view the usability test provides significant additional assurance in the output of the SPND sub-system for the purposes of flux mapping. If a core mis-load was present, it would cause some asymmetry in flux distribution across the core. I consider it very unlikely that this would be masked both during the usability test and during the subsequent flux mapping activity at multiple power levels, particularly because the power distribution changes during power raise. A systematic software error that prevented the operators from observing asymmetry in the SPND signals during the usability test would also likely cause anomalies in the measured power distribution when compared against nuclear design predictions during power raise.

548. In my opinion, accounting for the usability test and other factors listed in paragraph 547, a core mis-load followed by failure of the SPND sub-system cannot credibly result in operating the plant in a condition which would require the higher classification RPS to act to prevent fuel damage but in which that system does not act because the usual trip set-points have not been reached.

549. I am not satisfied that the straightforward application of the Ref. 145 process in Ref. 147 has been done correctly, and consider this to be a minor shortfall in the safety case. However, I am satisfied for the reasons given above that the FC-3 function categorisation for core mis-load detection is adequate.

Overall Sub-System Classification

550. To support an overall demonstration that risks have been reduced ALARP through the classification of the SPND sub-system, the RP has provided in Ref. 147 a comparison against the classification and functions of in-core neutron detector systems used in CPR1000, EPR and AP1000 PWR plants. It concludes that the FC-3 categorisation is appropriate for the SPND sub-system functions in UK HPR1000.

551. I consider the in-core detector systems used for the EPR and CPR1000 plants to be substantially different in both design and application to the SPND sub-system used for UK HPR1000. The ‘BEACON’ SPND-based system designed for AP1000 is more similar to that proposed for UK HPR1000. The AP1000 system is designated Class 3, and in my judgement provides a relevant comparator for assessment of the proposed classification for UK HPR1000.

552. I have also observed that the functions allocated to the SPND sub-system by the RP and the system classification are consistent with high level advice provided by IAEA in SSG-52 (Ref. 19).

553. Overall, following the design modification made in GDA (Ref. 146) I am satisfied that an F-SC3 classification for the UK HPR1000 SPND sub-system is adequate. I have reached this view because of its equivalency to the classification reached for similar plant in previous GDA and because I judge that risks associated with failure of the system have been reduced ALARP by the use of additional measures, especially the usability test. These additional measures should be captured within operating rules, a topic I have discussed further in subsection 4.12 of this report.

4.10.2 Strengths

554. Following my assessment of the UK HPR1000 safety case for core mis-loading and classification of the SPND sub-system, I have identified the following strengths:

- Due to the high boron concentration specified during refuelling, the core is inherently safe against the risk of inadvertent criticality from a mis-loading fault, which is a strength of this design.
- The safety case has been improved by the addition of a demonstration of continued fault tolerance following an undetected core mis-load.
- In response to interactions during GDA, the RP has upgraded the safety categorisation of the functions of the in-core SPNDs, which I now judge to be adequate. The design will be upgraded as a result.

4.10.3 Outcomes

555. Following my assessment of the UK HPR1000 safety case for core mis-loading and classification of the SPND sub-system, I have identified one minor shortfall.

4.10.4 Conclusion

556. Based on the outcome of my assessment, I have concluded that sufficient controls are in place to mitigate the risks due to a core mis-load, meeting expectations set by SAP ECR.1 and NS-TAST-GD-075 (Ref. 11). Noting SAP ERC.4, I judge that the RP has specified all reasonably practicable steps to detect core mis-loads and allow appropriate recovery actions to be taken. The core design is inherently safe against the risk of inadvertent criticality as long as correct boron concentration is maintained and adequate fault tolerance has been demonstrated against mis-loads that may be undetected before full power is reached, in accordance with expectations set by EKP.1, EKP.2 and the FA series of SAPs.

557. I am also satisfied that the SPND sub-system has been allocated an appropriate safety classification in accordance with the expectations set by SAP ECS.1.

4.11 Management of Failed Fuel in Operation

4.11.1 Assessment

558. Despite measures taken to improve the reliability of PWR fuel assemblies, failures of fuel cladding can and do occur, either in normal operation or because of faults. Common mechanisms have been discussed previously and are identified by IAEA in their review of fuel failures in water cooled reactors (2006-2015) (Ref. 62). As I have observed in subsection 4.4 of this report, if the historical average reliability of AFA 3GAA fuel is replicated for UK HPR1000 then approximately three fuel rods may be expected to lose their integrity through the life of the plant.

559. When a cladding failure occurs, fission products present in the pellet-clad gap will gradually be released to the coolant, increasing the RCS coolant activity. Water ingress through the failed cladding can occur and potentially lead to further degradation. Water ingress may also invalidate some of the technical acceptance criteria used for fuel in faults. For example, water-logged fuel can fail in an RCCA ejection fault below the usual RAFPE criterion (see paragraph 287). A strategy is therefore necessary to allow failures in normal operation to be detected and ensure that the risks associated with failed fuel are reduced ALARP.

560. I have derived my expectations for this topic from the SAPs and NS-TAST-GD-075 (Ref. 11). SAPs paragraph 542 under SAP ERC.1 states: "…the criteria and strategy

for dealing with fuel failures should be specified." Ref. 11 contains further detailed guidance to inspectors on this topic, which I have considered in my assessment.

561. My assessment of this topic also interfaces with those of other technical disciplines. Chemistry inspectors have discussed in their Assessment Report (Ref. 136) whether quantified coolant chemistry parameters and operating limits, including activity limits, have been adequately justified by the RP. Fault Studies inspectors have discussed in their Assessment Report (Ref. 7) whether the radiological consequences of design basis faults have been predicted conservatively in accordance with the expectations of SAP FA.7. The assumptions made about coolant activity are an important input to this analysis for many faults and are influenced by the strategy adopted to deal with fuel failures.

562. The Environment Agency also consider this topic because the strategy for dealing with failed fuel and the associated coolant activity limits can have an effect on radioactive discharges in normal operations.

563. My initial assessment of the RP's failed fuel strategy found a set of documentation that lacked coherence. Some of my expectations had been addressed in either PCSR Chapter 21 (Chemistry) or Pre-construction Environmental Report (PCER) Chapter 3 (Demonstration of Best Available Techniques (BAT)) submissions. However, there were areas of inconsistency between these two documents, in both the types of monitoring they stated were used to detect fuel failures and the actions that were described if activity limits were reached. It was not possible for me to determine whether the actions described in the PCSR or in the PCER would happen first in a real operating scenario. I also observed a lack of any explanation as to how the strategy would minimise the dispersal of nuclear material from the failed fuel rod to the coolant or limit further degradation of the cladding material following detection of a failure.

564. Consequently, I raised several queries in RQ-UKHPR1000-1176 (Ref. 33). In response, the RP has summarized a more coherent strategy that explains how the different measures previously described in PCSR Chapter 21 and PCER Chapter 3 will work in tandem, as follows:

- Failures are detected through sampling of coolant to monitor dose-equivalent iodine-131 activity, total noble gas activity, the ratio of xenon-133 to xenon-135 activity and iodine-134 activity. Dose-equivalent iodine-131 activity and total noble gas activity are the primary drivers of radiological consequences in the coolant activity if a fault were to occur. The ratio of xenon-133 to xenon-135 activity is intended to give early indication of fuel failures and the iodine-134 activity is monitored specifically as an indication of tramp uranium activity in the coolant.
- If abnormal conditions are observed, the frequency of sampling is increased. If one of the parameters above exceeds a particular value at the edge of the usual operating range (to be defined), appropriate actions will be taken to recover normal operating values within a specified time and maintain plant stability.
- If operating limits on dose-equivalent iodine-131 activity, total noble gas activity or iodine-134 activity are exceeded, the plant will shut down within a specified time.
- Increased coolant activity can also be detected using gamma-sensitive detectors in the Plant Radiation Monitoring System, which are located on the Chemical and Volume Control System (CVCS) let-down line and the Nuclear Sampling System (NSS) line from the RCS. These monitors are used to initiate two alarm levels, based on the usual operating range and the operating limit for dose-equivalent iodine-131 activity. If alarm level 1 is exceeded on either channel, the operator will closely monitor any increase in the measured value

and RCS charging and let-down is used to purify the coolant, aiming to return the plant to the operating window. If alarm level 2 is exceeded on both channels, containment isolation valves in the CVCS, NSS and vent and drain systems will be automatically closed.

- If fuel failures are suspected then following reactor shutdown, failed fuel assemblies will be identified by a combination of online and offline sipping tests, then stored in one of five dedicated failed fuel cells in the spent fuel pool.

565. The operating limits discussed above have been quantified and justified during GDA in the water chemistry specification and radiochemical parameters justification (Ref. 150 and Ref. 151), which are assessed within Ref. 136. The operating ranges for dose-equivalent iodine-131 activity and total noble gas activity remain to be defined by the licensee.

566. As part of the ALARP justification for a SG tube rupture fault (Ref. 152), the RP has suggested a potential additional operating limit for dose-equivalent iodine-131 that would lead to some further recovery action at a lower activity than the maximum limit specified. This additional limit has been quantified but there is inconsistency between Ref. 152 and Ref. 150 about what action may be initiated if it is reached. This topic is discussed in Ref. 136 and Ref. 7.

567. Notwithstanding the apparent inconsistency described in paragraph 567 I am now satisfied that the RP’s strategy is more coherent and consistent than it was previously.

568. With reference to the advice and good practice described in Ref. 11, I am satisfied that release of activity into the coolant from failed fuel should be detected and that limits and constraints on coolant activity are captured by the proposed operating rules. However, I do not see adequate evidence of measures to limit further degradation of the clad following a failure being identified, or to minimise the dispersal of material from the failed rod. The current strategy would allow the plant to continue to operate normally for the rest of the operating cycle with known fuel failures in the core if the coolant activity could be managed such that limits were not breached.

569. Some actions upon a parameter exceeding the usual operating range are left to be defined in future. This is reasonable because they relate to the licensee’s decision making processes and do not affect the design. However, these aspects need to be developed further to allow a demonstration that the expectations of Ref. 11 have been met.

570. Importantly, the strategy does not place a clear objective to recover the failed fuel assembly at the earliest practical opportunity and could instead allow further degradation of the clad and dispersal of material to occur. As a result, I judge that the RP’s strategy is not yet adequate to reduce risks ALARP. I do not have confidence that further development of the existing strategy as normal business will lead to an appropriate outcome, so I have raised the following Assessment Finding.

AF-UKHPR1000-0007 – The licensee shall, as part of their operating procedures, implement a strategy for decision making in the event of a potential fuel failure being identified that defines the actions to be taken to reduce relevant risks to as low as reasonably practicable. The strategy should minimise the dispersal of nuclear material and limit further degradation of the cladding material.

571. I would expect such a strategy to be commensurate with the principles set by SAP MS.3 and SAPs paragraph 69 on the topic of decision making (Ref. 2).

572. For the storage of failed fuel assemblies once recovered, I am satisfied that the provision of five dedicated failed fuel cells in the spent fuel pool is sufficient, based on

a comparison with RGP from other PWR and my review of historical reliability data for AFA 3GAA fuel.

4.11.2 Strengths

573. Following my assessment of the UK HPR1000 strategy for management of failed fuel in operation, the RP has improved its consistency and coherency, but I have not identified any particular strengths of the case.

4.11.3 Outcomes

574. Following my assessment of the UK HPR1000 strategy for management of failed fuel in operation, I have identified the following outcome:

- I have raised an Assessment Finding requiring the licensee to implement a strategy for decision making in the event of a potential fuel failure being identified that defines the actions to be taken to reduce relevant risks to ALARP. The strategy should minimise the dispersal of nuclear material and limit further degradation of the cladding material.

4.11.4 Conclusion

575. Based on the outcome of my assessment, I have concluded that the RP's strategy for management of failed fuel in operation contains shortfalls against the expectations of ONR guidance, but that these can be adequately dealt with by the licensee without impact on the reactor design.

4.12 Operating Limits and Conditions, Commissioning and EMIT

4.12.1 Assessment

Strategy and Expectations

576. Paragraph 100 in the SAPs, associated with SAP SC.4, provides a list of expectations for the content of a safety case. These include "…(e) identify all the limits and conditions necessary in the interests of safety (operating rules); and (f) identify any other requirements necessary to meet or maintain the safety case such as surveillance, maintenance and inspection.

577. The development of detailed operating procedures, commissioning plans or EMIT schedules for UK HPR1000 is for the licensee and outside the scope of GDA. However, where the safety case in GDA makes assumptions about, or sets requirements on these activities, this needs to be made clear. The GDA technical guidance (Ref. 10) states that ONR may choose to assess "How it will be ensured that assumptions, requirements and commitments made within the safety case documentation are transferred to the licensee to be included in operating rules, manuals, procedures, training requirements, commissioning tests, etc…"

578. SAP ERC.1 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. The related paragraph 541 of the SAPs states that there should be suitable and sufficient margins between the normal operational values of safety-related parameters and the values at which the physical barriers to release of radioactive materials are challenged. This sets an implicit expectation that operating rules associated with the fuel and core be set in the safety case to ensure that there is suitable and sufficient margin to the values at which physical barriers (including the fuel clad) are compromised.

579. SAPs ECM.1, EMT.1 and EMT.2 identify the need for a facility to undergo commissioning activities before operation, for SSCs to undergo regular and systematic through-life EMIT, and for these commissioning and EMIT activities to be identified in the safety case. NS-TAST-GD-075 (Ref. 11) provides further advice to inspectors on these topics with respect to nuclear fuel.

580. Accordingly, my strategy for assessment of these topics in GDA has been:

- to look for evidence that limits and conditions necessary in the interests of safety have been identified;
- to look for evidence that EMIT activities necessary in the interests of safety have been identified;
- to look for evidence that an appropriate strategy of core monitoring and physics testing is planned to confirm that the core (as built) operates within the performance envelope defined by the safety case; and
- to form a judgement on whether the limits and conditions, EMIT activities and commissioning activities I have identified in the safety case are sufficiently clear and traceable to enable transfer to the licensee.

581. I have reviewed limits and conditions, EMIT activities and the core monitoring strategy as integrated parts of my wider assessment in the preceding subsections in this report.

Physics Testing

582. The RP has presented a high-level summary of the content of different stages of commissioning planned for UK HPR1000 in PCSR Chapter 30 sub-chapter 30.5.3 (Ref. 153) and has identified high level commissioning requirements for the reactor core, including physics tests, in PCSR Chapter 5.8 (Ref. 3). I sought some more details of planned physics test activities through RQ-UKHPR1000-1287 (Ref. 33) to satisfy myself that the scope was appropriate.

583. In response, the RP has listed all the parameters it expects to be measured in both commissioning and periodic physics tests, the core operating state under which each measurement would be made, and the purpose of each measurement. It has also presented a separate set of information clarifying additional tests that are expected for the first cycle core. In my opinion the scope of physics testing appears appropriate because it includes checks on the important and measurable core neutronic parameters that influence the response of the core in faults and will allow any significant deviation from the core's predicted nuclear behaviour to be identified.

Clarity and Traceability of Requirements and Assumptions

584. In response to RO-UKHPR1000-0004 on the suitability and sufficiency of the safety case (Ref. 154), the RP has produced some new documentation and a coding system intended to demonstrate that 'specific requirements and assumptions' of the UK HPR1000 safety case are transferred to the licensee in a clear and traceable fashion. The term 'specific requirements and assumptions' in the RP's safety case encompasses all the information discussed above including limits and conditions, EMIT requirements and commissioning requirements, where they are necessary for the safety case. Another term used for these in the ONR TAG on the purpose, scope and content of safety cases (Ref. 155) is 'implementable requirements'. The adequacy of the RP's approach to this issue is assessed generically within the Cross-Cutting Step 4 Assessment Report (Ref. 156). From a Fuel and Core perspective, I have sampled the relevant parts of the safety case to form a judgement as to whether, at the conclusion of GDA, the necessary specific implementable requirements are now sufficiently clear and traceable.

585. The RP has submitted PCSR Chapter 31 on operational management (Ref. 157), which provides a summary of planned operating procedures, operating limits and conditions and EMIT.

586. The operating limits and conditions in Ref. 157 include a specific category termed 'core design requirements'. In the RP's Generic Limits and Conditions of Operation report (Ref. 158), these are expanded as a set of operating technical specification limits on parameters including boron concentration, MTC, axial offset, power distributions and RCCA bank positions. I am satisfied that this document captures the key parameters that must be monitored during power operation (in addition to those measured in start-up physics tests) to ensure the core is behaving as expected from a nuclear performance perspective. Ref. 158 also presents a set of reactor core safety limits, which correctly align with the thermal hydraulic design limits on DNBR and fuel temperature (see subsection 4.6). However, safety limits necessary to prevent fuel failure through other mechanisms such as PCI are not included in Ref. 158.

587. A particular set of operating rules that is omitted from Chapter 31 during GDA is the set of core design requirements that will be applied during design of each new re-load pattern during operation, to ensure that the assumptions of the generic safety case are met. These requirements are captured in Tier 2 documents within the Fuel and Core safety case, primarily the Nuclear Design Basis (Ref. 26). During my interactions with the RP in Step 4, I have been informed that they expect the link to be made between Ref. 26 and each new core design by way of two documents. Firstly, the Re-load Safety Analysis Checklist should record the list of generic and specific key safety parameters and their limiting values, against which new re-load core designs will be checked. Data in this document will be obtained from safety case reports including the Nuclear Design Basis. Secondly, the Core Operating Limits Report should record all core related operating limits for each new specific fuel cycle, providing the link between core design and the operators of the plant. I am satisfied that this is a reasonable high-level approach and appears similar to that applied at some operating PWRs, but it is not part of the safety case in GDA and the detail will be the responsibility of the licensee.

588. The EMIT section of Ref. 157 identifies the need for core physics tests but does not identify any need for fuel assembly inspections or surveillance. The RP has submitted some relevant information in PCSR Chapter 5.9 (Ref. 3), which does identify the need for some fuel surveillance. It states that during fuel unloading, the fuel assemblies will be required to undergo an online sipping test whenever abnormal radioactivity levels in the primary coolant are detected. I consider this requirement is sufficiently clear and aligns with the failed fuel strategy described to me (see subsection 4.11). PCSR Chapter 5.9 also states that visual inspection will be required to examine items including the fuel rod cladding surface and structural integrity of the fuel assembly grids. However, it does not provide any clarity on exactly what would be inspected, how frequently or why.

589. In summary, these top-level PCSR chapters and the Generic Limits and Conditions report do contain a selection of the more important operating limits and conditions and EMIT requirements for the UK HPR1000 fuel and core. However, they are not comprehensive and do not provide traceability to the source of the limits as implementable requirements in the lower-tier safety case reports. To identify the full set of implementable requirements that should eventually be captured by operating rules, it is currently necessary to read the full range of Tier 2 and Tier 3 documents under PCSR Chapter 5.

590. A selection of operating rules and EMIT requirements that I have identified from the fuel and core safety case during my assessment but which I judge are not yet identified

or managed as implementable requirements in a sufficiently clear and traceable fashion are listed below:

- the requirement for operating rules on the sequence of activities and RCCA bank insertions during start-up to ensure MTC is non-positive before power ascension begins (paragraphs 105-106);
- the requirement for an RCCA surveillance scheme to ensure RCCAs are ageing in the manner predicted and are replaced before absorber swelling or depletion become excessive (paragraph 239);
- the requirement on core re-load design for F_{dH} to drop later in a cycle in order for the rod bow penalty applied on DNBR to be valid (paragraphs 362-363);
- the requirement for a fuel surveillance scheme to monitor fuel assembly bow (paragraph 366);
- the requirements for limits on flexible operations to ensure that generic nuclear parameters supplied for fault analysis remain bounding and that PCI safety margins remain sufficient (subsections 4.3.1.5 and 4.7.1.3);
- the requirement for a fuel surveillance scheme to monitor crud deposition (sub-section 4.9.1.4); and
- the requirement for a ‘usability test’ to be undertaken before each use of the in-core SPNDs for a safety-categorised function, to ensure adequate reliability of the SPND sub-system (sub-section 4.10.1.2).

591. The list above is based on my sample and may not be comprehensive. Overall, I conclude that the fuel and core limits and conditions and EMIT activities in the safety case are not yet sufficiently clear and traceable. This is because a number of them are not captured in either the relevant sections of the PCSR Chapter 5 Tier 1 report, or anywhere within PCSR Chapter 31 (operational management) and its references. Rather, the reader must review all of the Tier 2 and Tier 3 documents under PCSR Chapter 5 to be able to identify what operating limits and conditions and EMIT activities are required by the fuel and core safety case.

592. For commissioning, the RP has not consolidated the detailed information about physics test plans provided in response to RQ-UKHPR1000-1287 (Ref. 33) within the safety case during GDA, stating that this is an activity for the licensee. However, it has captured commitments that this work be completed post-GDA within its commitment log (Ref. 40), with the intent that the detail in the RQ response is not lost.

593. Overall, I consider it necessary to raise the following Assessment Finding to ensure that this matter is resolved and tracked post GDA.

AF-UKHPR1000-0126 – The licensee shall ensure that relevant fuel and core related implementable requirements are included in site-specific operating documentation and underpinned by the safety case. This should include, but not be limited to, those requirements relating to operating rules, examination, maintenance, inspection and testing requirements, commissioning tests and rules for core reload design which feature in the generic safety case but which are not yet identified and managed as implementable requirements.

4.12.2 Strengths

594. Following my assessment of the operating limits and conditions, EMIT and commissioning requirements captured within the UK HPR1000 safety case, I have identified the following strengths:

- Fuel and core operating limits and conditions and EMIT necessary in the interests of safety are now captured within the parts of the fuel and core safety case that I have sampled.

- An appropriate strategy for physics testing and core monitoring has been developed by the RP, with the detail to be consolidated in the safety case by the licensee as normal business.

4.12.3 Outcomes

595. Following my assessment of the operating limits and conditions, EMIT and commissioning requirements captured within the UK HPR1000 safety case, I have raised an Assessment Finding requiring the licensee to ensure that relevant fuel and core related implementable requirements are included in site-specific operating documentation and underpinned by the safety case. This should include, but not be limited to, those requirements which feature in the generic safety case but which are not yet identified and managed as implementable requirements.

4.12.4 Conclusion

596. Based on the outcome of my assessment, I have concluded that safety case requirements for operating limits and conditions, EMIT and commissioning activities are captured adequately in a technical sense in Fuel and Core safety case documents. However, informed by the GDA technical guidance (Ref. 10) and SAP SC.4, I judge that further work is required post-GDA to ensure all such information is clear and traceable.

4.13 Computer Code Validity

4.13.1 Assessment

597. A number of different computer codes have been employed to predict aspects of fuel and core performance in the UK HPR1000 safety case submitted by the RP.

598. The Framatome COPERNIC code is used for analysis of fuel performance. This code has known pedigree with Framatome fuels and has been used on other projects with which ONR is involved, including GDA for the UK EPR. A brief review of the COPERNIC validation report (Ref. 32) has confirmed my expectation that it is a comprehensive document and presents extensive validation evidence for key predictions such as clad corrosion and fuel temperature. I have chosen not to sample this report in detail due to the regulatory attention it has received in the past for application to fuels of a similar type.

599. The Chinese-developed JMCT code is used by the RP for criticality analyses during core loading, which I considered in subsection 4.10.1.1. Validation of the JMCT code has undergone assessment by ONR’s criticality specialist and their TSC for application in criticality calculations for UK HPR1000 fuel in the spent fuel pool. Their work included both confirmatory calculations and code documentation reviews, as reported in the Radiation Protection and Criticality Assessment Report (Ref. 9). In that context and cognisant of the conservatism in the core mis-loading analysis, I considered it unnecessary in GDA to sample the evidence underlying the JMCT code for this specific core loading application.

600. The remaining Fuel and Core codes have been developed in-house by CGN and the UK HPR1000 GDA is their first application in a safety case context anywhere in the world. I have therefore focused on the validation of these codes within my assessment. It is important to note that the scope of my assessment of these codes is limited to their validity for application in the UK HPR1000 safety case submitted for GDA. My judgements, findings and conclusions should not be read across to any other applications of these codes. The codes are as follows:

- COCO – a 3D nuclear core design code

- PINE – a 2D lattice physics code
- POPLAR – a 1D core calculation code
- PALM – a depletion calculation code
- BIRCH – a fuel rod temperature analysis code
- LINDEN – a sub-channel analysis code

601. For each of these computer codes, the RP has submitted a code qualification report and subsequently a code V&V report that goes in to greater depth about the underlying evidence. The RP also submitted generic software quality assurance documentation and signed records for each code (partially in Chinese) to show that their quality assurance process was properly followed.

602. My expectations associated with validity of these codes are primarily drawn from the AV range of SAPs and the associated TAG, NS-TAST-GD-42 (Ref. 14).

603. The V&V reports submitted by the RP contain a wide range of evidence to support the validity of the codes' predictions and present this evidence in a clear, concise way. Except for guidelines to code users, all the information necessary to meet the expectations set by SAP AV.5 is present. I have not been able to assess code user guides during GDA, but I am content that this poses little risk to the generic UK HPR1000 design.

604. I have made use of two TSCs to aid me in reaching judgements on the adequacy of these codes and the associated documentation:

- firstly, to undertake confirmatory analysis of the UK HPR1000 safety case calculations with an independent code-set, to improve my confidence in the relevant CGN computer codes, particularly the physics codes COCO, PINE and POPLAR; and
- secondly, to undertake reviews of the code qualification and V&V reports against the expectations embodied in the AV SAPs and Ref. 14 for all six codes.

605. The code documentation review first consisted of my TSC reviewing the code qualification reports against the expectations of the AV SAPs and Ref. 14, raising questions through RQs when it identified potential shortfalls. The V&V reports were submitted for assessment by the RP after my TSC's initial review. My TSC then checked whether its questions and comments on each qualification report had been addressed within the corresponding V&V report and reported its findings. I have followed my TSC's findings up with the RP myself where I judged necessary. This process has led to the RP further updating the V&V report during GDA to the versions referenced in this report.

606. My TSC's documentation reviews were to a varying degree of depth, on my instruction. Some involved additional sampling of the V&V reports as well as checks that questions and comments on the qualification reports had been addressed. The LINDEN code documentation was reviewed in most detail because the code is important to the safety case and I could not gain insight in to its performance through my other TSC's confirmatory analysis, which did not use a dedicated sub-channel code. I have also undertaken targeted assessment of my own on some of these codes and raised a number of RQs.

607. My TSC also conducted a review (Ref. 159) of the CGN software quality assurance documentation, which it found to be in accordance with its expectations and with RGP such as guidance in IAEA SSG-2 (Ref. 18). The only matter remaining after completion of my TSC's review was a lack of clarity over whether the procedures covered code application by the end-users as well as code development. This lack of clarity was caused by some ambiguity in the make-up of the CGN organisation. I have since

gained clarity from the RP in response to RQ-UKHPR1000-1343 (Ref. 33) that the end-users are part of the 'Reactor Engineering and Safety Research Center' part of CGN, which is subject to the submitted procedures. I am therefore satisfied that the status of CGN's software quality assurance documentation was adequate for the purposes of GDA.

608. Interactions with the RP through joint technical workshops with Fault Studies inspectors have also given me confidence that procedures are in place for transferring and control of data between disciplines (use of COCO output data in fault analysis is an important example of this).

609. There has been no opportunity in GDA for me to conduct a visit to verify CGN's application of QA procedures for development or application of software, or transfer and control of data. These procedures are an important part of the overall method of analysis. However, I have been able to gain confidence in correct application of some of the codes for GDA through confirmatory analysis work. As noted in subsection 2.3, it will be for the licensee to decide what computer codes it utilises to support its safety case and operations, and what arrangements it uses to control their use and ensure their adequacy. I am confident these decisions and arrangements will be subject to appropriate ONR attention as part of routine regulatory interventions associated with licensing and permissioning.

4.13.1.1 COCO

610. COCO is a best-estimate 3D nuclear design code that uses two-energy-group fuel assembly parameters provided by PINE (4.13.1.2) to perform nodal neutron diffusion calculations. Its main purpose is to simulate a wide range of core operating states to produce data such as cycle length, discharge burnups, critical boron concentration curve, power distributions and a range of other neutronic and kinetic data important for safety analyses. It is used directly in some DBA to calculate specific power distributions and other neutronic data following a fault. In other DBA, it is used indirectly to verify that fault analysis assumptions about neutronic data are correct (see subsection 4.3).

Confirmatory Analysis Findings

611. For my assessment of the COCO and PINE codes, I have drawn extensively upon confirmatory analyses undertaken by my TSC. These used an independent 3D nuclear design code named QUABOX/CUBBOX, coupled to a 2D lattice code named NEWT, part of the SCALE code-suite. These codes have pedigree in confirmatory analyses undertaken for previous GDA and are widely used internationally in other applications. As 3D diffusion and 2D lattice physics codes, these are fundamentally of a similar type to COCO and PINE. My findings associated with COCO and PINE from the confirmatory analysis are reported here together because the outputs of the 3D code cannot be considered in isolation of the lattice code. The remainder of my assessment of PINE is reported in subsection 4.13.1.2.

612. The main outcomes and conclusions from my TSC's confirmatory analysis are reported in Ref. 22. Further details of the findings associated with COCO and PINE are reported in Ref. 160. My TSC has compared a range of output parameters for all cycles and full power core operating states covered by the RP in the fuel management report (Ref. 29). In light of the final results, my TSC reported confidence that all the conclusions drawn from the confirmatory fault analyses performed, which relied upon the physics modelling, were valid. My TSC made no recommendations to ONR pertaining to the physics codes themselves (COCO and PINE).

613. Following initial development of the confirmatory physics models, my TSC's comparison of its outputs with data from the RP showed good agreement for most

parameters, including radial power distributions and RCCA bank integral worths. However, several differences between my TSC's and the RP's results were identified. First, differences in k-infinity were found between the PINE and SCALE outputs, especially for the fuel assembly types with higher uranium enrichment and higher number of the number of rods containing gadolinia. Second, for the 3D cycle calculations, especially the equilibrium cycle, some parameters such as the critical boron concentration curve, the BOC axial power profile and some RCCA bank differential worth curves, were showing differences.

614. The causes of most differences were eventually understood by my TSC to be differences in the way that my TSC and the RP had modelled the dimensions of the fuel when hot (accounting for thermal expansion), and an error in the RCCA geometry data transferred to my TSC. Once these differences were addressed in my TSC's analysis, the majority of output parameters from both PINE and COCO showed good agreement. However, the axial power profile and RCCA bank differential worth curves at the beginning of the equilibrium cycle still show some differences; my TSC report that this is likely to be caused by the nature of the burnup distribution provided by the RP for this core operating state, which my TSC used as a starting point for its equilibrium cycle analysis. I am content that these remaining differences do not pose a significant concern in GDA because (1) the likely root cause has been identified, (2) the total RCCA worth still shows satisfactory agreement and (3) these parameters should all be verified by physics tests during UK HPR1000 commissioning.

615. My TSC undertook extensive investigations of the observed differences before identifying the reasons discussed above. These investigations have provided me with some additional confidence in the outputs of the PINE and COCO codes by providing further verification of the SCALE modelling against which PINE results were compared. The investigations included:

- refinement of the time discretisation used in SCALE, which improved the agreement with PINE;
- comparison between k-infinity predictions from SCALE and the Monte Carlo code KENO, which showed good agreement; and
- sensitivity analysis for the number of rings modelled in SCALE in the fuel pins containing gadolinia, which showed that a greater number of rings made little difference to the k-infinity results as a function of burnup.

616. Following discovery of the root causes discussed above, my TSC also reported that a confirmatory 2D Monte Carlo calculation was performed with the corrected input data to provide additional verification of a selection of the 2D lattice code outputs, and the Monte Carlo code results were in between the SCALE and the PINE results. The subsequent agreement in outputs between the 3D codes (QUABOX/CUBBOX and COCO) was largely satisfactory, as discussed in paragraph 615.

617. In summary, the output from my TSC's confirmatory analysis has given me improved confidence in the adequacy of the physics codes PINE and COCO for the UK HPR1000 application in GDA. I do not judge the remaining differences to be a significant concern in GDA. Through the analysis I have also gained some further insight in to the way in which PINE and COCO are applied in analysis of reactivity faults. My TSC conclude in Ref. 22 that the methods and assumptions used by the RP for an RCCA bank withdrawal fault and a rod drop fault both lead to conservative results. This topic is discussed further in the Fault Studies Assessment Report (Ref. 7).

Findings from Code Documentation Review

618. The COCO V&V report is Ref. 161 and my TSC's review is reported in Ref. 162. My TSC's review found that the V&V report provides extensive documentation on the implementation of the physical models. It also found that the objective of the V&V effort

is explained and corroborated by the presented results, and that statistical analysis is applied in order to quantify uncertainties. My TSC commented that COCO seems to possess the main characteristics of well-established codes familiar to it from other work. However, it also identified a number of specific items that were not fully addressed in the V&V report. I have followed these up through my own assessment where I judged necessary, as described in the following paragraphs.

619. My TSC identified that there was little information available about the experimental measurement precision and criteria for deciding whether the agreement between calculation and measurement is reasonable. I consider this to be a minor documentation issue. In my opinion the uncertainties associated with COCO outputs are mostly adequately articulated in the safety case and the data in the V&V report has been used in deriving these uncertainties where practicable. There was one exception found by my TSC's review, associated with the uncertainty to be applied to predictions of the Doppler coefficient. In my opinion this was the most significant finding of my TSC's work because the Doppler coefficient plays a significant role in many accident transient analyses. I therefore raised RQ-UKHPR1000-0795 (Ref. 33) to request more information on how uncertainties were defined for Doppler coefficients produced by COCO and how these were applied in the safety analyses. The RP's response to the RQ gave sufficient explanation of the method used and showed that the safety analyses actually assumed a higher uncertainty, bounding that which was predicted. This information has since been consolidated in to the COCO V&V report (Ref. 161). I am therefore satisfied that in this respect the code predictions allow for a conservative safety case analysis, meeting the expectations of SAPs AV.3 and FA.7.

620. My TSC also identified that more documentation should be provided on the way in which the radial reflector is modelled in COCO. I judged this important to meet the expectations set by SAP AV.1, however, I had already obtained a more detailed description of the reflector model and sensitivity analyses to show that modelling simplifications have small effect on the predicted core parameters, in the RP's core reflector calculation report (Ref. 163). I am therefore content that the safety case is adequate on this point, informed by both SAPs AV.1 and AV.6.

621. My TSC also identified that additional basic and separate effects tests should be provided for the validation of the thermal-hydraulic module in COCO and identified that the COCO predictions were not validated experimentally for the application in a rod drop fault, as per SAP AV.2. However, I have gained sufficient confidence in these particular aspects of the COCO predictions for the purposes of the UK HPR1000 safety case through the positive outcomes from my TSC's confirmatory analysis discussed above. I consider this to be a minor shortfall in the code documentation.

Assessment of the Physics Modelling of Exposed Fuel with the RCCAs Fully Inserted

622. As discussed previously in subsection 4.4, I identified that a short length of active fuel is exposed at the bottom of the UK HPR1000 core when the RCCAs are fully inserted. The RP has submitted analysis to show that this has a very small effect on SDM. However, in my opinion this design presents greater challenge to physics modelling methods, particularly because the axial discontinuity in the fuel and control rod geometry sits part-way through a single axial layer in the COCO model. In addition to following up matters identified by my TSC and informed by SAPs AV.1 and AV.6, I have therefore looked for evidence that the COCO model adequately represents the UK HPR1000 core design for the purposes of SDM calculations, and that sensitivities to the modelling techniques are understood.

623. After sampling the V&V report I raised two RQs on this topic. The RP has submitted several pieces of relevant evidence including sensitivity analyses to investigate the

effects of reducing or increasing the axial mesh size in COCO. It has also submitted the results of code-to-code comparisons with a Monte Carlo physics code (OpenMC) and data from experiments including predictions (stated to be ‘blind’) of RCCA bank integral worths at existing Chinese CPR1000 plants.

624. The sensitivity analyses provide evidence that the chosen mesh size (a minimum of 16 axial layers) is sufficient to adequately predict key parameters including critical boron concentration, 2D power distribution, axial offset and SDM. The sensitivity in SDM is easily bounded by uncertainties applied to SDM in fault analyses.

625. The code-to-code comparisons of RCCA bank worths between COCO and OpenMC reported in Ref. 161 show differences that are all within the uncertainty range applied to COCO predictions of RCCA bank worths in the safety case. The RP further confirmed that the fidelity of the COCO models used for this validation work and for the UK HPR1000 safety case analysis work is equivalent. I note that these comparisons were only undertaken at the beginning of the first cycle, so do not account for any burnup behaviour, but I judge that they still provide useful evidence to verify that the COCO models adequately represent the geometry of the fuel and RCCAs.

626. The validation evidence involving blind predictions of RCCA bank integral worths at existing Chinese CPR1000 plants shows that all differences are well within the uncertainty range applied to the COCO outputs in the UK HPR1000 safety case. Although the CPR1000 fuel and RCCA designs are not identical to those of UK HPR1000 in all respects, I am satisfied that the experimental data supports the validity of this particular aspect of COCO modelling.

627. Overall, informed by SAPs AV.1 and AV.6 I am satisfied that sufficient evidence has been provided for GDA to show that COCO provides adequate predictions of SDM for UK HPR1000 despite the challenges posed by the RCCAs not covering the full active length of the fuel when inserted. Both experimental data and diverse analytical methods have been used to provide V&V data, while sensitivity studies have been carried out to demonstrate that the conclusions are not overly sensitive to the number of axial slices in the COCO model.

Application of HPR1000 Physics Test Data

628. Guidance associated with SAP AV.1 states “Models should be validated for each application made in the safety analysis. The validation should be of the model as a whole or, where this is not practicable, on a module basis, against experiments that replicate as closely as possible the expected plant condition.” Further detailed guidance in Ref. 14 states that tests carried out in full-sized plants during commissioning or start-up procedures, as well as operational transients or accidents, can be a useful source of data and should, where practical, be included in the validation report.

629. Drawing on this guidance, I consider that validation evidence for nuclear physics codes and their models of a reactor core should include comparisons with measurements taken during commissioning or start-up procedures for a core design that is very similar or, if practical, the same as that being modelled.

630. None of the experimental data used in COCO validation to date has come from a HPR1000 plant with the same reactor core and fuel design as the UK HPR1000 will have. This is not practical in GDA because the HPR1000 is a new reactor design. The most relevant operating plant data used by the RP is from the CPR1000 fleet, but this has a very slightly different fuel and RCCA design to the UK HPR1000, as well as a smaller core with fewer fuel assemblies and RCCAs. I understand that multiple other HPR1000 plants are currently in build or in planning stages internationally and will be much closer to the UK HPR1000 in design terms than the CPR1000 fleet is.

631. For these reasons, I have raised an Assessment Finding to prompt the licensee to make use of new HPR1000-specific physics test data as it becomes available, to improve the validation of the nuclear physics code outputs (COCO and PINE, or any alternatives used by the licensee). In my opinion, informed by Ref. 14, this would provide a stronger validation case and may also allow the remaining differences observed between the RP results and my TSC's confirmatory analysis at the beginning of the equilibrium cycle to be resolved.

632. Resolution of this Assessment Finding should not require any additional measurements to be undertaken on HPR1000 plants outside the UK. It should only require additional nuclear analyses by the licensee to predict the results of those tests using UK HPR1000 methods, and the sharing of data between HPR1000 plants to allow comparisons to be made.

AF-UKHPR1000-0008 – The licensee shall include in the validation base for its chosen nuclear physics codes and models a range of comparisons with measured physics test data. This should include axial power distributions and Rod Cluster Control Assembly bank differential worth curves, from reactor core and fuel designs that are as close as practicable to those of UK HPR1000. If practical the measured data should come from HPR1000 plant(s). The comparisons should include measured data from cores containing both fresh and partially-burnt fuel.

Conclusions

633. Confirmatory analyses show generally good agreement between the outputs of the COCO and PINE codes and the outputs of the QUABOX/CUBBOX and SCALE (NEWT) codes used by my TSC. This applies across all core cycle designs and full power core operating states for a wide range of physics parameters. I do not judge the remaining differences to be a significant concern in GDA.

634. After assessing documentation associated with COCO I am satisfied that the expectations set by the AV series of SAPs are met for the purposes of GDA.

635. Overall I am satisfied with the adequacy of the V&V evidence provided for COCO for the purpose of GDA.

636. I have raised Assessment Finding AF-UKHPR1000-0008 to require the licensee to make use of new HPR1000-specific physics test data as it becomes available, to further strengthen the validation base for the nuclear physics codes.

4.13.1.2 PINE

637. PINE is a 2D lattice physics code used to undertake neutron transport and depletion calculations for fuel assemblies, using nuclear data from external libraries. Its main purpose is to undertake these calculations for individual UK HPR1000 fuel assembly types under a range of conditions and provide homogenized two-energy-group data for application in COCO. This data includes average macro cross-sections, neutron diffusion coefficients and assembly surface discontinuity factors. COCO does not operate without input data from PINE and therefore PINE can be considered to have the same set of applications as COCO.

Confirmatory Analysis Findings

638. For my assessment of PINE, I have drawn extensively upon confirmatory analyses undertaken by my TSC and reported in Ref. 160, as discussed in subsection 4.13.1.1. After investigations of differences in k-infinity predictions discovered early in the confirmatory analysis programme, my TSC's comparisons now show satisfactory

agreement between the outputs of the PINE code and the outputs of the SCALE code-suite used by my TSC.

Findings from Code Documentation Review

639. I observed from the PINE V&V report (Ref. 164) that the nuclear and cross-section data used with PINE in GDA is obtained from the IAEA Winfrith Improved Multigroup Scheme-D (WIMS-D) Library Update Programme (WLUP). The kinetics data is obtained from the Japanese Evaluated Nuclear Data Library (JENDL) 4.0. These are established international sources of data and should be applicable to the fuel design used by UK HPR1000, in accordance with the expectations of SAP AV.3. It is good practice to make use of more modern nuclear data libraries as they become available and I anticipate that the licensee may need to update the V&V report for PINE if a new data library is used. I consider this to be normal business.

640. My TSC's review (Ref. 165) found that PINE is similar to other lattice transport-depletion codes as used within their experience base for reactor physics applications. However, my TSC identified a number of specific items that were not fully addressed in the V&V report. I have followed these up through my own assessment where I judged necessary, as described in the following paragraphs.

641. My TSC identified that more information should be presented in the V&V report about the PINE validation range for moderator-to-fuel ratio. However, I am satisfied that the UK HPR1000 safety case is adequate in this respect because the moderator-to-fuel ratio in the reactor core is very similar to other modern PWR designs and is well within the range captured by the PINE V&V work.

642. My TSC identified that more information should be presented in the V&V report about experimental measurement precision and the criteria for deciding whether the agreement between calculation and measurement is satisfactory. This information was provided by the RP in response to RQ-UKHPR1000-0320 (Ref. 33) and I consider its omission from the safety case to be a minor shortfall.

643. My TSC identified that more information should be presented in the V&V report to allow the "richness" of the experimental database and its representativeness for supporting the validation range of the code to be assessed. By richness of the database, my TSC meant the dispersion of the available data throughout the range of claimed validation (for example, whether the data is evenly distributed or almost all concentrated in a small part of the claimed range).

644. I judged that the most important parameter ranges in this context were the fuel enrichment and burnup. For the purposes of UK HPR1000 GDA, I have taken confidence from the fact that the actual ranges of fuel enrichment and fuel burnup used for the UK HPR1000 are well within the validation ranges claimed for the PINE code, rather than towards the extremes of those ranges. I also observed that verification benchmark case results are available throughout the burnup range, not just at the extremes, and that some of the experimental results are provided for a number of different fuel enrichments. I am content to judge, supported by the results of the confirmatory analysis discussed previously, that it is unlikely there are any cliff edge effects in the PINE code's performance in the fuel enrichment and burnup ranges in which the UK HPR1000 core design lies. I therefore consider that the lack of clear information to allow the richness of the V&V data to be assessed is a minor shortfall in the safety case. In my opinion this shortfall does not undermine the validity of PINE for the specific application to the UK HPR1000 core designs.

645. My TSC also identified specifically that further experiments should be considered to validate the nuclide density and gadolinium burnable poison predictions as a function of fuel enrichment and burnup. No experimental data was provided in the V&V report

reviewed by my TSC to validate PINE predictions of isotopic inventories (as opposed to other parameters like reactivity or power distributions) at multiple enrichments or burnups. In my opinion this data is an important part of demonstrating that the code's calculations adequately represent the processes taking place (SAP AV.2) so I pursued further evidence through an RQ. In response, the RP has incorporated data from two additional experiments in Ref. 164, comprising post-irradiation measurements taken from spent fuel rods at PWRs in Italy and Japan. There are some differences shown between PINE predictions of isotopic inventories and the experimental results, although these are of similar magnitude to the differences seen with a 'reference code' and no concerning systematic trends appear to me to be present.

646. Using all of the evidence available to me from analytical verification, experimental validation and confirmatory analyses, I judge it likely that the PINE predictions of isotopic inventory are adequate for the code's application in the UK HPR1000 safety case. However, I believe it would be practical for the licensee to further strengthen the safety case in this area. Resolution of Assessment Finding AF-UKHPR1000-0008 raised previously should provide additional evidence that the nuclear physics codes (PINE and COCO, or alternatives chosen by the licensee) produce adequate predictions of neutronic parameters for the UK HPR1000 safety case, for cores with both fresh and partially-burnt fuel.

Conclusions

647. Confirmatory analyses by my TSC show satisfactory agreement between the outputs of the COCO and PINE codes and the outputs of the QUABOX/CUBBOX and SCALE codes used by my TSC. This applies across all core cycle designs and full power core operating states for a wide range of physics parameters, with the exception of axial power distribution and RCCA bank differential worths at the beginning of the equilibrium cycle.

648. After assessing documentation associated with PINE I am satisfied that the expectations set by the AV series of SAPs are largely met for the purposes of GDA.

649. Overall I am satisfied with the adequacy of the V&V evidence provided for PINE for the purpose of GDA.

650. I have raised Assessment Finding AF-UKHPR1000-0008 to prompt the licensee to make use of new HPR1000-specific physics test data as it becomes available, to further strengthen the validation base for the nuclear physics codes.

4.13.1.3 POPLAR

651. POPLAR is a 1-D core calculation code. Its main function is to merge 3D core data provided from the COCO code in to a 1D axial model and then solve the diffusion equation to produce predictions of axial flux and power distributions. It can be used for either steady state or transient conditions and forms a part of the analysis chain for DBA of some reactivity transients.

Confirmatory Analysis Findings

652. For my assessment of the POPLAR code, I have again drawn upon confirmatory analyses undertaken by my TSC. POPLAR is a simple 1D code used only for some fault analyses and is not used to produce any steady state physics data, so none of the confirmatory analysis reported by my TSC in its physics report (Ref. 160) is directly relevant. However, my TSC has reported specific confirmatory analysis for a RCCA drop transient in Ref. 166 and for an RCCA bank withdrawal from zero power in Ref. 167. The RP has applied POPLAR (together with COCO) in its fault analysis for both of

these faults. My TSC has used its QUABOX/CUBBOX code suite to produce 3D physics data for these transients, which should provide more realistic results.

653. Due to the difference in types of methods, I did not expect a very good match in results for these transients, but did expect the results produced by POPLAR, if applied correctly, to be conservative. My TSC's methods should give a better prediction of actual plant performance and therefore I expected my TSC's results to show more margin to acceptance criteria than the POPLAR results.

654. My TSC stated that their comparison of results showed that the RP's rod drop analysis has been performed on a conservative basis. My TSC started with analysis of the rod drop case that it understood to be most limiting, and then conducted several sensitivity analyses to assumptions including the number of falling RCCAs and the starting position of the R bank. My TSC's analysis results showed significant margin to the acceptance criteria for the main acceptance criteria in all cases. Due to the significant differences in modelling techniques between my TSC and the RP, my TSC were not able to explore and fully understand differences between the results. However, it was clear that the results produced for the rod drop transient using the RP's method with POPLAR were significantly more conservative than those produced using my TSC's method.

655. My TSC stated that the comparison of results showed that the RP's RCCA bank withdrawal analysis (from zero power) had also been performed on a conservative basis. Indeed, my TSC stated that large differences were observed for the power peak, which was both earlier and higher in the RP's analysis than the one predicted by my TSC. My TSC considered this was most likely due to a faster reactivity insertion in the RP's simulation. This faster reactivity insertion was present despite the fact that my TSC had assumed the maximum possible RCCA bank withdrawal rate in their analysis.

656. Overall, I judge from the results of the confirmatory analysis for these faults that the way the RP has applied the POPLAR code in its safety case leads to significantly more conservative results than those reached using my TSC's methods.

Findings from Code Documentation Review

657. The evidence presented in support of the validity of the COCO code (see subsection 4.13.1.1) is important for POPLAR in many respects because the 1D model data used in POPLAR by the RP is derived from the 3D COCO model outputs. Much of the POPLAR V&V report (Ref. 168) constitutes verification of the 1D POPLAR model against the equivalent 3D COCO model.

658. My TSC's review (Ref. 169) found that the validation report provided extensive documentation on the implementation of the physical models. It also found that the objective of the V&V effort was explained and corroborated by the presented results, and that statistical analysis was applied in order to quantify uncertainties. My TSC also commented that POPLAR seems to possess the main characteristics of well-established codes familiar to them within their experience base. However, it also identified a number of specific items that were not fully addressed in the V&V report. I have followed these up through my own assessment where I judged necessary, as described in the following paragraphs.

659. My TSC found that the predictions of POPLAR were not tested in Ref. 168 with respect to RCCA drop measurements. It also commented that the scope of application of POPLAR includes asymmetrical transients (including an RCCA drop). As the radial asymmetry is not accounted for, my TSC observed that there may be a limitation on the accuracy of predicted reactivity and power for these transients. The effect of this limitation was not addressed in the V&V report reviewed by my TSC.

660. My TSC also observed that critical boron concentration is the only parameter for which uncertainty is evaluated in POPLAR outputs.

661. I followed these points up with the RP through RQ-UKHPR1000-0794 (Ref. 33). The RP has subsequently explained in Ref. 168 that it deals with asymmetry in faults analysed by POPLAR by applying a conservative set of bounding inputs for POPLAR analyses that are derived from COCO analysis. Bounding radial power distribution and power distribution asymmetry data are calculated using COCO and then used to determine the limiting transient power peaking data with POPLAR. The RP also explained that uncertainties in the POPLAR modelling are dealt with by applying a conservative set of bounding inputs. Evidence has been submitted in Ref. 168 that using the combination of COCO and POPLAR leads to more conservative transient analysis results than using COCO alone, for a selection of RCCA drop, RCCA bank withdrawal from zero power, and single RCCA withdrawal fault cases. I am content that the way the codes have been applied in these fault transients has resulted in conservative predictions of power peaking factors and other neutronic parameters.

Conclusions

662. Confirmatory analyses by my TSC show that the way the RP has applied the POPLAR code in its safety case leads to results for rod drop and RCCA bank withdrawal (from zero power) transients that are significantly more conservative than those predicted by my TSC's methods.

663. After assessing documentation associated with POPLAR I am satisfied that the expectations set by the AV series of SAPs are largely met for the purposes of GDA. Where gaps were identified by my TSC, I have satisfied myself that POPLAR will nevertheless produce conservative data in UK HPR1000 DBA, in accordance with the expectation set by SAP FA.7. I have done this through a combination of the confirmatory analysis work and further interaction with the RP via RQs.

664. Overall, I am satisfied with the adequacy of the POPLAR code and associated V&V evidence for the specific applications for which it has been used in the UK HPR1000 safety case.

4.13.1.4 PALM

665. PALM is a core depletion calculation code. It uses nuclear data sourced from an external library to predict how the isotopic inventory of the fuel in the core will evolve during operation. This can be used to produce predictions of total decay heat, which are used in DBA. It can also be used to predict the inventory of specific isotopes, for consideration as part of accident source terms or consideration in determining spent fuel storage requirements.

666. In this report I have only considered the adequacy of PALM for the purposes of decay heat predictions to support DBA. The adequacy of PALM for the purposes of isotopic inventory and source term calculations is considered separately in the Radiation Protection and Criticality Assessment Report (Ref. 9).

667. I observed that the nuclear data source used by PALM is the internationally-recognised Joint Evaluated Fission and Fusion File (JEFF) 3.3 database. Uncertainty allowances are derived in the PALM V&V report (Ref. 170) for use with decay heat data calculated using PALM.

668. My TSC's review (Ref. 171) found that the detailed validation cases and sensitivity cases presented in the PALM V&V report showed satisfactory results, in the same order of magnitude as well-known codes such as SCALE and ORIGEN, which my TSC have experience of using.

669. The only matter raised by my TSC in Ref. 171 was that the data in the V&V report did not allow it to assess the limits of applicability of the bounding uncertainties given for decay heat predictions. In other words, my TSC could not be sure that the uncertainties provided in Ref. 170 were valid in all circumstances, such as at low powers or at different times in cycle.

670. For several reasons, I have formed a judgement that this is a minor shortfall and does not undermine the validity of PALM for its specific application in the UK HPR1000 safety case. Firstly, my TSC have confirmed the RP's argument that the uncertainties in decay heat production are mainly due to uncertainties in the underlying nuclear data rather than the code itself; the nuclear data source used by the RP is well-established internationally and my TSC stated that the uncertainties in the nuclear data considered by the RP are suitable. Secondly, the RP has run a number of 'typical' calculation cases with PALM to estimate a decay heat uncertainty range due to the underlying uncertainties in the nuclear data. These calculations included cases for both the first and equilibrium cycle core designs defined in GDA. The RP selected the largest predicted decay heat uncertainty from all of these cases for each time step, in order to produce a bounding set of decay heat uncertainties as a function of time after shutdown. This means that the set of uncertainties should be conservative for the fuel designs (for example, enrichments) used in these core designs. Thirdly, I have found through my own assessment of the decay heat report (Ref. 50, see sub-section 4.3.1.4) that the RP have used an uncertainty range of 1.645 σ (standard deviations) from this bounding uncertainty data in order to produce decay heat data for safety case application. The measured data presented by the RP in Ref. 170 as validation evidence for the decay heat predictions all falls within +/- 1 σ of the nominal PALM prediction and therefore supports the RP's position that the uncertainties are bounding.

671. Although the declared uncertainties are not explicitly proven in Ref. 170 to apply in all core operating states, I am satisfied on balance that their application should produce conservative decay heat predictions. Informed by SAP AV.3, I consider the lack of explicit data to prove the declared uncertainties are valid in all core operating states to be a minor shortfall in the safety case.

672. Overall I am satisfied with the adequacy of the PALM code for use in generating decay heat predictions in the UK HPR1000 safety case. My assessment of the decay heat predictions themselves is reported in subsection 4.3.1.4.

4.13.1.5 BIRCH

673. BIRCH is a fuel rod temperature analysis code. It is used to calculate the radial temperature distribution of a fuel rod cross-section. Other key parameters such as heat flux at the cladding surface, energy stored in the fuel pellet and cladding oxidation can also be calculated. It is used in DBA to predict the maximum fuel temperatures reached in some fault conditions and can be used to calculate RAFPE for RCCA ejections.

Confirmatory Analysis Findings

674. For my assessment of the BIRCH code, I have drawn partially upon confirmatory analyses undertaken by my TSC. This was not a prime objective of the confirmatory analysis programme. However, as part of my TSC's analysis of an RCCA bank withdrawal fault from zero power (Ref. 167), the rate of bank withdrawal was artificially increased beyond the declared maximum withdrawal rate in order that the power transient predicted by my TSC more closely matched that predicted by the RP. This allowed a comparison to be made by my TSC between the margin to DNBR and fuel temperature acceptance criteria in the two different cases, which was not clouded by the known difference (see paragraph 656) in power peaking predictions.

675. My TSC concluded that the maximum fuel temperature remained more limiting in the RP's analysis, thus suggesting that the results obtained with BIRCH were more conservative than those obtained with my TSC's methods. For fuel temperature predictions, my TSC made use of a simple fuel heat transfer model within the system transient code ATHLET.

676. This only represented a single transient analysis case against a single other method, which itself is not a dedicated fuel performance code. In isolation this result is not sufficient to provide high confidence in BIRCH outputs. However, I judge that it supports my assessment of the code documentation reported below by providing an independent source of verification.

Findings from Code Documentation Review

677. The BIRCH V&V report is Ref. 172 and my TSC's review is reported in Ref. 173. My TSC's review observed that BIRCH is similar to other fuel temperature analysis codes that it is familiar with and that the V&V report shows that the code can predict fuel temperatures conservatively. However, my TSC highlighted that the simulation of the fuel to clad gap's heat conductance and BIRCH's validation range were areas of potential weakness against the expectations of SAP AV.2.

678. In addition to my TSC's review, I sampled some aspects of the BIRCH V&V report (Ref. 172) myself to give me further confidence that the analysis code was adequate for its application. From my sample, I observed that the V&V report did not include validation evidence relevant for M5 clad or fuel rods containing Gd_2O_3 burnable poison, which constituted a potential shortfall against the expectation set by SAP AV.1.

679. I raised queries to follow up both my own and my TSC's findings through RQ-UKHPR1000-0778 (Ref. 33). In response, the RP explained that it uses an output from the COPERNIC code to input the gap conductance data to BIRCH and provided evidence to show that the relevant fault analyses in which BIRCH is applied are within the validation range of the code. In addition, the RP has updated the BIRCH V&V data in Ref. 172 to capture data for M5 clad and fuel containing Gd_2O_3 burnable poison. I considered these to be adequate responses to my queries.

680. I also observed inconsistencies between Ref. 172 and the fault analysis report for RCCA Ejection (Ref. 41) about the source of boundary condition data used for calculating RAFPE with BIRCH. In response to RQ-UKHPR1000-1690 (Ref. 33), the RP confirmed that these apparent inconsistencies were implied by an error in Ref. 172, which has subsequently been addressed.

Conclusions

681. Confirmatory analysis by my TSC indicates that BIRCH provides conservative fuel temperature results for RCCA bank withdrawal (from zero power) transients. This provides limited, but independent, verification that BIRCH outputs are conservative.

682. After assessing documentation associated with BIRCH I am satisfied that the expectations set by the AV series of SAPs are met for the purposes of the UK HPR1000 safety case in GDA.

683. Overall, I am satisfied with the adequacy of the V&V evidence provided for BIRCH for the purpose of GDA.

4.13.1.6 LINDEN

684. LINDEN is a sub-channel analysis code applied to the thermal hydraulic design and fault analysis for UK HPR1000. It is used to calculate thermal hydraulic parameters of the coolant and DNBR in the reactor core under normal operating and fault conditions.

Findings from Code V&V Documentation Review

685. It has not been possible for me to gain any confidence in the validity of LINDEN through my TSC's confirmatory analysis work because the ATHLET code used by my TSC to produce DNBR predictions is a system code rather than a sub-channel code. LINDEN is important to the UK HPR1000 safety case because it is used to produce DNBR predictions for a wide range of frequent and infrequent design basis faults. Supported by my TSC, I have therefore sampled the V&V report submitted by the RP for the LINDEN code (Ref. 174) in greater depth and breadth than for the other codes discussed above. I have also assessed the implementation in LINDEN of the two CHF correlations used for UK HPR1000 (previously discussed in subsection 4.6.1.1), because valid DNBR predictions depend on both the code and the correlation used.

686. My TSC reported (Ref. 100) that its documentation review found no indication that the LINDEN code is not suitable for application in the UK HPR1000 safety case. It reported that the LINDEN code uses models and calculation methods that are commonly used in existing sub-channel thermal-hydraulic codes and that the validation experiments are classical well-known experiments. It reported high confidence that the code will be fit for typical steady-state calculations in the UK HPR1000 safety case. There were no significant shortfalls found in the range of data covered by the experimental validation when compared to the code applicability range. However, my TSC did identify some specific matters for ONR to consider pursuing further due to identification of potential shortfalls or weaknesses in some of the evidence. I have followed these up through my own assessment where I judged necessary, as described in the following paragraphs.

687. My TSC identified that the RP should present detailed code-to-code comparisons of local sub-channel parameters, to provide further confidence in the code and a better understanding of its local behaviour. I observed that this recommendation aligned with the expectations of both IAEA SSG-2 (Ref. 18) and the SAPs (paragraph 682) where direct validation by experiment is not practicable (which is true of local sub-channel parameters in a reactor). I have subsequently pursued it with the RP during GDA.

688. CGN shared with me [redacted] The parameters compared included local pressure drop, local mass velocity, local equilibrium quality and local void fraction. CGN stated that these were selected because they are the parameters most critical to prediction of CHF. A variety of operating conditions were considered and I observed that in general the comparisons showed good agreement. Where differences existed, CGN were able to provide thorough and logical answers to my questions, from which I gained confidence in CGN's understanding of the code's behaviour. None of these differences undermined my confidence in the capability of the LINDEN code for the purposes of DNBR predictions.

689. For the purposes of GDA I am satisfied that this matters presents little risk to the UK HPR1000 design or fault analysis. However, these comparisons are not part of the UK HPR1000 safety case. Therefore, I consider it necessary to raise the following Assessment Finding to ensure that the licensee addresses this shortfall against RGP.

AF-UKHPR1000-0009 – The licensee shall, in the verification and validation evidence underlying the UK HPR1000 thermal hydraulic sub-channel code, present code-to-code comparisons for sub-channel local parameter predictions.

690. My TSC also identified that the RP should validate the implementation of the FC2000 CHF correlation in LINDEN, confirming the statistics, instead of applying a conservative margin to the UK HPR1000 DNBR design limit. This accords with the findings of my own assessment of the DNBR limits and CHF correlations used for UK HPR1000, which I have discussed in subsection 4.6.1.1. I have raised an Assessment Finding in subsection 4.6.1.1, AF-UKHPR1000-0004, to ensure that the licensee addresses this potential shortfall.

Findings from Additional Sampling

691. In addition to considering the findings of my TSC's review, I sampled the LINDEN V&V evidence myself. Though doing this I identified two potential matters that I followed up with the RP through RQ-UKHPR1000-1343 (Ref. 33). These were (1) an apparent under-prediction of pressure drop at high void fractions that was observed in two experiments and (2) a lack of clarity as to the root cause of some differences observed versus rod bundle test results in predictions of transverse mixing. On the second point, I was primarily concerned that the results may imply a systematic bias in predictions of transverse mixing as a function of quality or other parameters, which had not been previously identified.

692. The RP was able to provide additional clarification in the LINDEN V&V report (Ref. 174) that the observed under-prediction of pressure drop only occurred for void fractions either outside or very close to the limit of the application range for LINDEN in UK HPR1000. Further, the RP provided evidence that an under-prediction of local pressure drop of the magnitude observed in the experiments, only occurring high in a fuel assembly where void fraction is high, was insignificant in its effect on DNBR predictions. I judged this evidence adequate to demonstrate that the experimental comparisons do not undermine confidence in the LINDEN code for the purposes of the UK HPR1000 safety case.

693. The RP also submitted new analysis of results from the rod bundle tests intended to measure the amount of transverse mixing in Ref. 174 to show that the differences observed (quantified as an M/P ratio) were not biased above or below 1.0 and showed no systematic behaviour as a function of mass velocity, quality or pressure. The RP postulate that the differences observed versus experiment are due to the use of a constant thermal diffusion coefficient in the LINDEN code, rather than a coefficient that varies as a function of local parameters. I recognise this is an assumption common to other sub-channel thermal hydraulic codes.

694. Overall, I consider that this matter is unlikely to compromise the adequacy of the LINDEN sub-channel analysis for the purpose of DNBR predictions because the errors appeared randomly distributed rather than systematic in nature, and their effect on DNBR predictions should therefore be accounted for statistically in the derivation of the DNBR design limits discussed in subsection 4.6. I have taken some further confidence that these errors are not significant from [redacted] (see paragraph 689). Once the two Assessment Findings I have raised on validation of the selected CHF correlation limits in the sub-channel code used for fault analysis (AF-UKHPR1000-0004) and on submission of sub-channel code-to-code comparisons (AF-UKHPR1000-0009) have been resolved, I judge there will be adequate evidence to substantiate the licensee's DNBR predictions. This will resolve any residual uncertainty due to the differences discussed in paragraph 694.

Conclusions

695. After assessing documentation associated with LINDEN I am satisfied that the expectations set by the AV series of SAPs are met for the purposes of GDA.

696. I am satisfied with the adequacy of the V&V evidence provided for LINDEN for the purpose of GDA. However, I have raised an Assessment Finding, AF-UKHPR1000-0009, to ensure the licensee presents code-to-code comparisons to strengthen the validation of the chosen subchannel code's prediction of local sub-channel parameters. I also raised another relevant Assessment Finding, AF-UKHPR1000-0004 in subsection 4.6, to ensure the licensee validates the correlation limits for the chosen CHF correlation with the sub-channel code used for fault analysis.

4.13.2 Strengths

697. Following my assessment of the evidence underlying the UK HPR1000 Fuel and Core computer codes, I have identified the following strengths:

- Confirmatory analyses show generally good agreement between the outputs of the COCO and PINE codes and the outputs of the QUABOX/CUBBOX and SCALE (NEWT) codes used by my TSC.
- My TSC's reviews found that COCO, PINE, POPLAR, PALM, BIRCH and LINDEN were all similar in nature to their equivalents with which my TSC is familiar from its other work. I observed that the code V&V reports generally present an extensive range of validation data.
- After following up some initial findings, raised due to both my TSC's reviews and my own sampling of the V&V reports, the RP has now provided adequate evidence of the codes' validity in the vast majority of areas sampled.

4.13.3 Outcomes

698. Following my assessment of the evidence underlying the UK HPR1000 Fuel and Core computer codes, I have identified the following outcomes:

- I have raised an Assessment Finding requiring the licensee to include in the validation base for its chosen nuclear physics codes and models a range of comparisons with measured physics test data. This should include axial power distributions and RCCA bank differential worth curves, from reactor core and fuel designs that are as close as practicable to those of UK HPR1000. If practical the measured data should come from HPR1000 plant(s). The comparisons should include measured data from cores containing both fresh and partially-burnt fuel.
- I have raised an Assessment Finding requiring the licensee to present satisfactory code-to-code comparisons for sub-channel local parameter predictions as part of the V&V evidence underlying the UK HPR1000 thermal hydraulic sub-channel code.

699. I identified a small number of other minor shortfalls during my assessment.

4.13.4 Conclusion

700. Based on the outcome of my assessment, I have concluded that the COCO, PINE, POPLAR, PALM, BIRCH and LINDEN codes and their associated documentation are adequate for the purposes of their applications in the Fuel and Core safety case for UK HPR1000, subject to closure of two Assessment Findings by the licensee.

701. I have reached this conclusion using a combination of my TSC's confirmatory analysis results, my TSC's documentation reviews and my own sampling, considering advice primarily from the AV series of SAPs and NS-TAST-GD-42 (Ref. 14).

4.14 Demonstration that Relevant Risks Have Been Reduced to ALARP

4.14.1 Assessment

702. A nuclear licensee or dutyholder in the UK has a legal requirement to reduce risks So Far As Is Reasonably Practicable (SFAIRP). NS-TAST-GD-075 (Ref. 12) provides technical guidance to ONR inspectors on what they should expect of a dutyholder in meeting this. The term ALARP is usually used when referring to the level to which risks must be reduced in order to meet the legal requirement, and is considered equivalent to SFAIRP.

703. Annex 2 to Ref. 12 gives specific advice on ALARP for new reactors. It states that although nominally at the design stage, the proposed designs for GDA are essentially complete in terms of the overall concept and major systems, and have reached that stage after many years of development and optimisation in non-UK regulatory environments. This is indeed the case for the generic UK HPR1000 design, in which the core design is an evolution of the CPR1000 core design in the Chinese nuclear fleet and the AFA 3GAA fuel design is a standard French product used in numerous reactors around the world. Ref. 12 Annex 2 recommends that four main areas should be addressed for the overall ALARP demonstration for the design and I have used this guidance as the basis for my assessment. I also observed that the RP's ALARP methodology (Ref. 175) contains a set of steps that broadly align with my expectations from Ref. 12.

704. In the context of fuel and core design, I consider there are two distinct areas in which ALARP should be demonstrated.

- Firstly, the fuel and other in-core components themselves present a level of risk because they may fail due to through-life degradation mechanisms, with a safety consequence. An ALARP demonstration should therefore be provided for these components. The fuel assemblies themselves require a more rigorous ALARP assessment than other in-core components because the consequences of them failing, with a direct loss of the first barrier to fission product release, are higher. As a result, I have sampled the RP's fuel assembly ALARP demonstration in GDA.
- Secondly, the holistic core design has a major impact on the plant response to transients and the worth of protection systems. A core design which appears to meet RGP when viewed in isolation may therefore not always reduce risks ALARP if there are further improvements that could practicably be made to reduce the consequence of faults. This can only be sensibly judged for a specific plant design using DBA, which provides the "risk assessment" element of the ALARP demonstration recommended in Ref. 12. As a result, I have sampled the RP's core design ALARP demonstration in GDA.

4.14.1.1 Fuel Design

705. The AFA 3GAA fuel assembly OpEx report (Ref. 31) contains several examples of learning from experience and provides me with high confidence in the pedigree of the fuel system design. It has also specified the codes and standards used to develop the fuel design in Ref. 176. The range of codes applied includes AFCEN RCC-C, AFCEN RCC-M, ASME III and RGP from the IAEA. In conjunction with the demonstration of application of these codes and standards in the detailed fuel design substantiation, I consider Ref. 176 to be an adequate demonstration of the use of RGP to develop the design. I have reported my assessment of the fuel design substantiation work in

subsection 4.4 of this report, concluding that the UK HPR1000 fuel system design and substantiation are adequate for GDA but raising an Assessment Finding, AF-UKHPR1000-0001, associated with structural integrity in a combined seismic event and LOCA.

706. The RP has submitted a specific ALARP demonstration report for the fuel design (Ref. 60). Ref. 60 shows that a number of measures have been taken to reduce risk by improvement of the AFA 3GAA fuel system design prior to its adoption in the UK HPR1000 plant design. It describes the evolution of AFA 3G family of fuel assemblies, including the key changes made between AFA, AFA 2G and AFA 3G families and the reasons for those changes.

707. Ref. 60 identifies two potential new design improvements as a result of recent operational feedback, a new thermal treatment for grid springs to reduce the risk of spring cracking, and a guide tube material change to reduce fuel assembly bow amplitudes. After raising RQ-UKHPR1000-0559 and RQ-UKHPR1000-0614 (Ref. 33), I have established that the new grid spring thermal treatment will be implemented for UK HPR1000 and is being applied to similar French reactors in advance, but that the change in guide tube material will not be implemented for UK HPR1000. The new guide tube material is being used instead of M5 in some fuel assemblies in reactors with a 14-foot active length in order to reduce amplitudes of fuel assembly bow. However, the same change is not recommended by Framatome for reactors like the UK HPR1000 with 12-foot active fuel length. I have considered the impacts of fuel assembly bow on the UK HPR1000 safety case in subsections 4.4.1.5 and 4.6.1.4 of this report and concluded that the safety case for fuel assembly bow is adequate. There would be no significant safety case benefit to be gained by this design change for UK HPR1000 and I am therefore satisfied with the position put forward that it not be adopted during GDA.

708. Overall, I am satisfied that the AFA 3GAA fuel design has been developed in accordance with RGP, that the extensive OpEx has resulted in an evolution of the design to improve safety and that some consideration has been given to further improving the design. I judge that adequate risk assessment work for the fuel exists in the form of extensive DBA (assessed fully in the Fault Studies Assessment Report, Ref. 7) but this DBA has not been used to identify any specific engineering improvements to the fuel design for UK HPR1000. Due to the possible need for small changes to the fuel design when the licensee addresses Assessment Findings in this report, it is not yet possible to reach a clear conclusion that there are no further reasonably practicable improvements that could be implemented. On balance however, I judge that such changes are unlikely to have a wider impact on the UK HPR1000 plant outside of the fuel and core design. I also recognise that it is common for fuel designs to evolve and improve through the life of a power station.

709. I am therefore satisfied that the RP’s ALARP demonstration for the fuel system adequately meets the expectations of Ref. 12 for the purposes of GDA.

4.14.1.2 Core Design

710. Core design (particularly nuclear design) is a complex multi-dimensional optimisation exercise in which numerous compromises must be made between different safety-related parameters. I expect RGP to be used to inform the core design. However, a comparison of the core design against RGP such as that contained within IAEA guidance (primarily SSR-2/1 and SSG-52, Ref. 17 and Ref. 19), is not sufficient in itself to demonstrate that a core design reduces risks to ALARP. This demonstration can only be made by undertaking DBA for the specific plant and core design together, in an integrated fashion. As described in Ref. 12, this risk assessment work should be used to identify potential engineering and/or operational improvements to the design. Whilst

such improvements may lie outside of the core in the wider plant design, I do expect improvements to the core design to be considered. Therefore, I expect the ALARP demonstration for the core to be informed by the results of DBA, as well as by a comparison with RGP.

711. In the context of DBA, NS-TAST-GD-075 (Ref. 11) provides guidance that ONR expect that fuel failure should not be predicted to occur in any frequent design basis faults (IEF > 10^{-3} per year), and the risk of fuel failure should be reduced ALARP in infrequent design basis faults. The risk of fuel failures should also be reduced ALARP in DEC-A, however, the level of conservatism in the analysis of these faults may be reduced when compared to that for design basis faults.

712. My expectation in this assessment is therefore that as part of the core design ALARP demonstration, for any faults in which fuel failures are predicted by DBA, potential engineering and/or operational improvements to the nuclear design of the core are identified and evaluated to see if the number of fuel failures can be reduced. Such potential improvements should be considered through an ALARP process.

Compliance with Relevant Good Practice

713. The RP has submitted the PCSR Chapter 5 ALARP demonstration report (Ref. 30) to provide a holistic demonstration that risks associated with the core design have been reduced ALARP. Ref. 30 provides brief statements of compliance against RGP for core design in a small selection of the ONR SAPs and Ref. 11. In isolation this would not be adequate, but reference is also provided to a more thorough compliance analysis of codes and standards in fuel and core design (Ref. 177), which provides a comparison against guidance contained within IAEA Safety Guide No. NS-G-1.12 (Ref. 178). I have reviewed this compliance analysis, which is largely a qualitative ‘signposting’ document pointing to other areas of the safety case for compliance with specific pieces of IAEA guidance. I am satisfied that it is adequate for the purpose of showing the relevant guidance within Ref. 178 has all been considered. Furthermore, I have assessed almost all of the topics covered by Ref. 177 in detail in earlier subsections of this report and, through that assessment, I am satisfied the design is sufficiently compliant with guidance in Ref. 178 except in those areas where I have previously reported findings. However, I observe that whilst Ref. 178 was until recently the most relevant available international guidance for new core design, it has been superseded by SSG-52 (Ref. 19) during the course of GDA Step 4. I would expect the licensee as part of its normal business to complete a comprehensive update of the compliance analysis against the most recent revision of IAEA guidance.

Evolution of the Design

714. Ref. 30 states that the UK HPR1000 core design is an evolution of that used in the Chinese CPR1000 operating plants. In my opinion, there is a benefit in the UK HPR1000 being derived from the CPR1000 core and being almost identical to that of the HPR1000 reference plant at Fangchenggang Unit 3 in China, and potentially other future HPR1000 plants. Firstly this is because the CPR1000 fleet provides the RP’s greatest OpEx base and has been used to provide data for validation of some of its analysis methods and computer codes. Secondly it means that physics testing data and other OpEx from other HPR1000 plants could potentially be used to provide additional confidence in the modelling predictions for the UK HPR1000 core, which could potentially also feed safety improvements back in to the design in future. Such benefits of standardisation are recognised by the ONR TAG on ALARP (Ref. 12). However, Ref. 12 advises that an ALARP demonstration for such evolutionary designs should also demonstrate how the evolution has maintained or improved the design from a safety perspective and evaluate further options for improvement.

715. Ref. 30 provides a description of the changes from the CPR1000 core and their benefits. These changes primarily constitute the addition of 20 fuel assemblies at the edge of the core (which slightly reduces the height/diameter ratio, reducing neutron leakage) and a slight reduction in the core average power density. The RP argues that the reduced power density will result in a higher thermal margin and hence a safety benefit, whilst the reduced neutron leakage will result in slightly higher average discharge fuel burnup and therefore fewer spent fuel assemblies per unit of energy production. In my opinion the changes are relatively modest, but will be effective as described. I have also observed that the slightly higher maximum burnup reached in the UK HPR1000 core is still well within the range of international experience with the UK HPR1000 fuel design, as noted in earlier subsections of this report. Other changes from the CPR1000 core design such as the number of RCCAs and locations of ex-core detectors follow on from the increased number of fuel assemblies. I am satisfied that the RP has given proper consideration to safety requirements associated with these components in the evolution of the CPR1000 design. The only aspect of the core design that constitutes a significant change from CPR1000 is the use of in-core SPNDs. The detailed design of these detectors will be finalised post GDA, but Ref. 30 provides an adequate explanation for the selected number of SPNDs and their locations. As discussed in subsection 4.10, I am also now satisfied that they have been assigned an appropriate safety classification. I judge that these detectors should provide for a practical safety improvement in UK HPR1000 over the CPR1000 fleet by providing the operators with earlier visibility of any unexpected distortions in the core power distribution.

716. Ref. 30 also provides a comparison of some key core thermal hydraulic parameters with those for EPR and AP1000. This shows that none of the parameters chosen are significantly out of step with those chosen for other PWR designs that have undergone GDA. However, in my opinion this fact does not contribute a great deal to the ALARP justification because the performance of a core design must always be assessed in conjunction with both the particular fuel used and with the wider reactor plant.

717. As a result of the work discussed in the paragraphs above, I am satisfied that the RP has demonstrated adequately for GDA that the UK HPR1000 core design is aligned with RGP and has evolved from its predecessors in a way that will improve safety. The RP's use of DBA as a risk assessment tool to identify further improvements to the core design is discussed in the following paragraphs.

Risk Assessment and Identification of Further Improvement Options

718. Following a series of interactions during GDA in which I provided advice to the RP in conjunction with Fault Studies inspectors, the RP has used its DBA as a risk assessment tool, to determine where core design or operational improvements could potentially be made to reduce the consequences of faults. In particular, insights and resulting actions reported by the RP include:

- a design change has been made during GDA Step 4 to the overpower ΔT reactor trip function when axial offset is negative. This change removes the potential for fuel failure due to PCI occurring in a frequent fault, as discussed in subsection 4.7 of this report; and
- a series of possible core design improvements have been postulated in a dedicated ALARP assessment for DNB analysis (Ref. 37, a sub-reference to Ref. 30) in order to reduce the consequences of two infrequent faults that are predicted to cause fuel failures due to CHF being reached in some parts of the core. The two faults are an RCP locked rotor and an RCCA Ejection. However, Ref. 37 ultimately argues that the core design and operational improvements postulated are not reasonably practicable, instead claiming some benefit in

reduced consequences due to improvements in the fault analysis. My assessment of these arguments is reported in the following paragraphs.

719. As well as supporting Ref. 30 for the core design, Ref. 37 forms part of a suite of reports that provide a holistic demonstration that the consequences of UK HPR1000 faults have been reduced ALARP. The whole suite of documents, including comparisons against the numerical targets in the SAPs, has been assessed in the Fault Studies Assessment Report (Ref. 7). The objective of Ref. 37 specifically is to show that the number of fuel rods where heat flux reaches CHF and DNB occurs (assumed to cause a loss of clad integrity) has been reduced ALARP in faults.

Assessment of ALARP Arguments for RCP Locked Rotor and RCCA Ejection

720. Considerations of radiological consequences and fault frequencies form part of the wider suite of ALARP documents assessed by Fault Studies inspectors. However, for the purpose of framing my assessment of Ref. 37, I observe that both a locked rotor and RCCA ejection are infrequent faults with frequencies < 10^{-4} per year and both faults have radiological consequences that lie between the Target 4 Basic Safety Objective (BSO) and Basic Safety Level (BSL) from the SAPs.

721. The RP has postulated improvements to reduce the predicted consequences of these faults in four categories: (1) analysis improvements, (2) mitigation measures (by which it means improvements to the design of protection systems), (3) nuclear design and operation, (4) thermal hydraulic design.

722. For both faults, the RP has been able to show that by changing to a ‘cycle-by-cycle’ analysis for which the neutronic and kinetic data inputs bound each of the current individual cycle designs rather than an artificially conservative combination of all cycles, the predicted number of fuel rods undergoing DNB can be significantly reduced, but not to zero. This will require the licensee to check each new re-load design during plant lifetime against the assumptions made in the analysis of these two faults. I consider this to be normal business for the licensee. I am satisfied that these changes in analysis assumptions will still result in conservative predictions of the fault consequences, in accordance with my expectations derived from SAP FA.7.

723. For both faults, the RP explains that no protection system improvements can reduce the fault consequences because of the very fast nature of the transients, in which fuel damage occurs before any protection action can take effect. The only thermal hydraulic design improvements identified that could reduce the fault consequences are a reduction to the reactor rated power, an increase in normal operating pressure, an increase in coolant flow-rate or a reduction in average coolant temperature. However, the RP argues that changes to these parameters are not reasonably practicable because of significant economic downsides and impacts on other aspects of plant safety. Although the RP has not fully quantified the downsides of these changes in Ref. 37, for faults of this low frequency with consequences below the BSL, I judge that the downsides of these design changes would likely be grossly disproportionate to the benefit. It is therefore my opinion that these changes are not reasonably practicable for the purposes of reducing the consequences of these faults. However for both faults, I judge improvements may be reasonably practicable in the core nuclear design and operation.

Potential Improvements to Nuclear Design and Operation to Reduce the Consequence of RCP Locked Rotor and RCCA Ejection Faults

724. For the locked rotor fault consequences, the RP reports sensitivity analysis in Ref. 37 to show that the most significant nuclear parameters are the axial power distribution, radial power distribution and MTC. The RP has identified nuclear design or operational improvements to improve each of these parameters, conducted sensitivity analyses to

show the potential benefit to the fault consequences and conducted other analyses to show the downsides including to achievable fuel burnup, cycle length and other safety parameters. I am satisfied this work adequately shows that design changes to improve the radial power distribution and MTC are not reasonably practicable.

725. However, I judge that further refinement of the maximum allowable positive axial offset may be reasonably practicable to make the limiting axial power distribution less onerous. The RP has shown that this would further reduce the consequences of a locked rotor fault. The RP has argued that the change is not reasonably practicable and has provided evidence, including operational data from a CPR1000 plant, to show that a large reduction in the positive axial offset limit would cause difficulties in keeping axial offset within the limits during power changes, particularly when making changes in boron concentration. However, in my opinion this work is not adequate to show that the current limit is optimised, nor therefore that a reduction of some magnitude is not reasonably practicable. I have therefore raised an Assessment Finding to prompt the licensee to address this shortfall, AF-UKHPR1000-0010. As a well as a review of the core axial offset limits, the Assessment Finding includes a review of the R bank insertion limits, which is justified in the following paragraphs.

AF-UKHPR1000-0010 – The licensee shall justify the UK HPR1000 temperature regulation Rod Cluster Control Assembly bank insertion limits and core axial offset limits together to demonstrate that the consequences of faults for fuel integrity have been reduced to as low as reasonably practicable.

726. For the RCCA ejection fault consequences, the RP reports sensitivity analyses in Ref. 37 to show that the most significant nuclear parameters are the ejected RCCA worth, the Doppler temperature coefficient and the delayed neutron fraction. The latter two parameters cannot easily be changed by design and so the RP has rightly focused efforts on changes that could reduce the ejected RCCA bank worth. For faults occurring from below full power, this could be done by changing the design and/or operating philosophy for the power compensation (N and G) banks of RCCAs, such that the maximum inserted individual RCCA worth for a given total inserted RCCA bank worth (associated with a given power level) is reduced. For faults occurring from full power, it could be done by changing the insertion limit of the R bank of RCCAs.

727. Following re-submission of Ref. 37 in the later stages of Step 4, the RP has identified several possible design changes to the G and/or N banks that could be used to change the maximum ejected rod worth at a particular power level. These include modifying the worth of individual banks by switching between “grey” and “black” designs, modifying the overall number of power compensation banks by utilising some RCCAs from shutdown banks, or modifying the control logic for insertion of the G and N banks such that multiple banks insert together. In each case the RP has quantified the benefits the change would have for the limiting consequences of an RCCA ejection and also explored the downsides. The RP argues that none of these changes should be implemented because of downsides for safety margin in other faults or/and for the ability to adequately control the core power distribution during normal operational transients, and has provided analytical evidence to support these arguments.

728. I judge that the RP’s evidence is adequate to show that improvements to the G and/or N banks to reduce ejected RCCA worth at reduced power are not reasonably practicable. However, I judge that further refinement of the R bank insertion limit may be reasonably practicable to reduce the ejected RCCA worth in a fault initiating from full power, in which mode the plant will likely operate for most of its life. The RP has shown that tightening the R bank insertion limit could significantly reduce the consequences of an RCCA ejection from full power, but argues the change is not reasonably practicable. The RP has provided evidence, including operational data from CPR1000 plants, to show that a large reduction in the limit would cause difficulties in

keeping axial offset within the limits during power changes, particularly when making changes in boron concentration. This is a similar argument to that made about the limits of the operating domain for the locked rotor fault. However, again in my opinion the work is not adequate to show that the current insertion limit is optimised, nor therefore that a somewhat tightened limit is not reasonably practicable. This shortfall should also be addressed by the licensee in resolving the Assessment Finding described previously, AF-UKHPR1000-0010.

729. I recognise that tightening the UK HPR1000 R bank insertion limit would make it more difficult to control axial offset below the maximum positive limit in some circumstances. The potential changes discussed above to reduce the consequences of these two faults are therefore not independent of each other.

Potential Benefits for Other Faults

730. Apart from the RCP locked rotor and RCCA ejection accidents, the only other design basis fault predicted by the RP to cause CHF to occur on some fuel rods is the IB-LOCA. The RP's radiological consequences analysis assumes that CHF is reached on all fuel rods in the LB-LOCA. It also assumes that where CHF is reached, local failure of the fuel rod cladding occurs. Another report within the suite of Fault Studies ALARP submissions, the ALARP Assessment for DBC Radiological Consequences (Ref. 85), provides the ALARP assessment for a range of faults including these LOCAs. However, core design changes are not considered to reduce the predicted consequences of LOCAs.

731. I have not pursued the consideration of core design changes specifically to reduce the consequences of LOCA faults in GDA because, when compared to all other PWR core designs that have been assessed in the UK and for which adequate LOCA safety cases have been made, the UK HPR1000 core has similar or lower power density and relatively modest power peaking factors (sub-section 4.2.1.2). These PWRs all have different RCS designs to UK HPR1000 and the transients are specific to each reactor plant. However, were all other things equal I judge the UK HPR1000 core design would be relatively robust to these LOCA transients. As a result, I do not currently envisage that their consequences could be significantly reduced by making reasonably practicable changes to the UK HPR1000 fuel or/and core designs. However, modest benefits may be realisable by the licensee through the review of operating limits captured in AF-UKHPR1000-0010.

732. I also anticipate that if the licensee makes changes to operating limits to address AF-UKHPR1000-0010, then it will likely improve safety margin for a number of other faults. These benefits may be less significant because all other design basis faults currently show margin to acceptance criteria and to the DNBR limit. However, the full range of potential benefits should be considered when the Assessment Finding is addressed.

733. I am satisfied that Ref. 37 provides adequate insights from risk assessment to support the holistic PCSR Chapter 5 ALARP demonstration (Ref. 30), subject to resolution of the above Assessment Finding. I also judge that resolution of the Assessment Finding does not pose significant risk to the actual design of the UK HPR1000 core or fuel proposed in GDA and is rather an operational issue.

734. Overall, I am satisfied that the RP's ALARP demonstration for the reactor core adequately meets the expectations of Ref. 12 for the purposes of GDA. I also observed that it is broadly consistent with the RP's own ALARP methodology outlined in Ref. 175.

4.14.2 Strengths

735. Following my assessment of the demonstration that the UK HPR1000 fuel and core designs reduce risks ALARP I have identified the following strengths:

- The reactor core ALARP demonstration sets out to address all of the key expectations for new reactor designs defined in NS-TAST-GD-005 (Ref. 12).
- The RP has provided a demonstration of how the core design has evolved from CPR1000 in a way that should improve safety and has made further improvements following risk assessment work in GDA.
- The RP has provided a demonstration that the fuel design benefits from extensive OpEx and has evolved from earlier AFA 3G models in a way that will improve safety.

4.14.3 Outcomes

736. Following my assessment of the demonstration that the UK HPR1000 fuel and core designs reduce risks ALARP I have identified the following outcomes:

- I have raised an Assessment Finding to ensure that the licensee justifies the R bank insertion limits and core axial offset limits together to demonstrate that the consequences of faults for fuel integrity have been reduced to ALARP.
- Several Assessment Findings I have raised previously in this report are also relevant to the demonstration that risks are reduced to ALARP.

4.14.4 Conclusion

737. Based on the outcome of my assessment, I have concluded that the Fuel and Core ALARP case submitted by the RP addresses the key expectations for new reactor designs within ONR ALARP guidance and is adequate for GDA.

4.15 Consolidated Safety Case

4.15.1 Assessment

738. My assessment of the UK HPR1000 fuel and core designs for GDA has been based on:

- the set of safety case submissions provided by the RP and summarised in Section 3;
- responses provided by the RP to ROs and RQs that I raised during GDA; and
- information provided to me during my technical interactions with the RP during GDA.

739. At the end of GDA, the RP is expected to capture relevant information from RQs, ROs and other interactions in final versions of the safety case documents. These final safety case submissions are captured in the Master Document Submission List (MDSL) (Ref. 179) and constitute the basis for future development of the safety case by a licensee. It is these documents, including version 2 of the PSCR (noting that Ref. 3 is version 1), against which a DAC or interim DAC will be awarded, if that is the outcome from GDA.

740. I have therefore undertaken a further sample to check that information I was previously provided that I considered relevant to my assessment has subsequently been consolidated sufficiently well in to the safety case submissions captured in the MDSL.

741. Through this sample I have determined that in the Fuel and Core area, sufficient fuel and core information provided by the RP in response to RO-UKHPR1000-0015 (fuel

deposits) and RO-UKHPR1000-0045 (thermal hydraulic performance at the fuel assembly edge) has been consolidated in the safety case.

742. The largest amount of information provided to me by the RP outside of the originally planned safety case submissions has been in response to RQs, of which I have raised around 70 during Step 4 alone. My sample of recently revised submissions, undertaken at the point of completing my assessment, has determined that RQ responses provided by the RP during GDA have now been adequately consolidated within the safety case submissions in the MDSL. I am therefore satisfied that all the information shared with me by the RP through RQ responses that is relevant to my assessment has been consolidated within the UK HPR1000 safety case.

743. All information provided to me by the RP through interactions and meetings that is relevant to my assessment has subsequently been provided in consolidated submissions, with the exception of some specific pieces of information that I have identified in this report and about which I have raised Assessment Findings where appropriate (AF-UKHPR1000-0004, AF-UKHPR1000-0009 and AF-UKHPR1000-0127).

744. I have also sampled revision 2 of PCSR Chapter 5 (Ref. 180), submitted after the end of my formal assessment period, to check it contains the information expected and is consistent with my assessment.

745. My review of Ref. 180 did not focus on technical detail, which I have covered in previous subsections of this report. Rather I have reviewed it for completeness against SAP SC.4, for ease of readability and to check whether interfaces and references are captured with sufficient clarity that the safety case is coherent.

746. SAP SC.4 sets the expectation that a safety case should:

- explicitly set out the argument for why risks are ALARP;
- link the information necessary to show that risks are ALARP;
- support claims and arguments with appropriate evidence, and with experiment and/or analysis that validates performance assumptions;
- accurately and realistically reflect the proposed activity, facility and its SSCs;
- identify all the limits and conditions necessary in the interests of safety (operating rules); and
- identify any other requirements necessary to meet or maintain the safety case, such as surveillance, maintenance and inspection.

747. In my opinion, Ref. 180 provides adequate linkage between the top-level claims, sub-claims, supporting arguments and evidence in the Fuel and Core safety case (as summarised in Section 3). It summarises the ALARP arguments that I have assessed in subsection 4.14 of this report. The technical content is of sufficient detail for this document, with references provided for further detail. I am satisfied that the document covers all aspects of the safety case expected by SAP SC.4 with the exception that operating rules are covered separately in PCSR Chapter 31 (Ref. 157) and that information about commissioning and EMIT is relatively limited in GDA, as discussed previously in this report.

748. Overall, I judge that revision 2 of PCSR Chapter 5 (Ref. 180) is adequate to provide an overview of the consolidated Fuel and Core safety case with references out to further information.

4.15.2 Strengths

749. Following my assessment of the UK HPR1000 consolidated safety case I have identified the following strengths:

- Based on my sample of submissions late in GDA, sufficient information previously shared with me by the RP that was important to my assessment (with known exceptions for reasons referred to previously in this report) has now been consolidated within the UK HPR1000 safety case.
- PCSR Chapter 5 revision 2 (Ref. 180) provides an adequate overview of the consolidated safety case with references out to supporting information and meets the expectations set by SAP SC.4 for the purpose of GDA.

4.15.3 Outcomes

750. Following my assessment of the UK HPR1000 consolidated safety case I have not identified any additional minor shortfalls or Assessment Findings.

4.15.4 Conclusion

751. Based on the outcome of my assessment, I have concluded that information provided to me by the RP that is relevant to my assessment has now been sufficiently well consolidated in submissions in the MDSL (Ref. 179). I have concluded that revision 2 of PCSR Chapter 5 (Ref. 180) provides an adequate overview of the consolidated safety case with references out to supporting information and meets the expectations set by SAP SC.4 for the purpose of GDA.

4.16 Comparison with Standards, Guidance and Relevant Good Practice

752. As explained in subsection 2.4, the key SAPs I have used in this assessment are EKP.1, EKP.2, EKP.3, EKP.4, EAD.1, EAD.2, ERC.1, ERC.2, ERC.3, ERC.4, FA.7, AV.1, AV.2 and AV.3. I have referred to other SAPs on an occasional basis throughout this report and the full list is presented in Annex 1.

753. The most commonly applicable ONR TAG for assessment of reactor core design is NS-TAST-GD-075 (Ref. 11). NS-TAST-GD-005 (Ref. 12) and NS-TAST-GD-042 (Ref. 14) have also been particularly important to parts of my assessment.

754. The most commonly applicable IAEA guidance for assessment of reactor core design is SSG-52 (Ref. 19). SSR-2/1 (Ref. 17) also contains a number of relevant requirements.

755. Other than where I have identified minor shortfalls or Assessment Findings in this report, I am satisfied that the expectations I derived from these sources of RGP have been met by the UK HPR1000 fuel and core design in the areas I have sampled.

756. As I observed in subsection 4.14, the RP has also completed a specific compliance assessment against the predecessor to SSG-52, IAEA Safety Guide No. NS-G-1.12 (Ref. 178).

5 CONCLUSIONS AND RECOMMENDATIONS

5.1 Conclusions

757. This report presents the findings of my Fuel and Core assessment of the generic UK HPR1000 design as part of the GDA process.

758. Based on my assessment, undertaken on a sampling basis, I have concluded that for the purposes of GDA:

- the nuclear design of the core, the thermal hydraulic design of the core and the thermo-mechanical design of the fuel system are adequate;
- improvements have been made to the core design and safety case such that there is adequate protection against fuel failures from PCI in frequent faults and the risks associated with fuel deposits are ALARP;
- sufficient controls are in place to mitigate the risks due to a core mis-load and the SPND sub-system has been upgraded to an appropriate safety classification;
- the neutronic and kinetic data provided for DBA is conservative;
- the technical acceptance criteria provided for fuel in DBA are generally supported by the experimental evidence, will help to ensure that radiological consequences from faults are predicted conservatively and allow a demonstration that the fault consequences are ALARP;
- the fuel and core computer codes and their associated documentation are generally adequate for the purposes of their applications in the fuel and core safety case;
- further work is needed to fully substantiate the RP's claim that a coolable geometry will be maintained in the unlikely event of a LB-LOCA;
- the strategy for management of failed fuel in operation does not meet some expectations of ONR guidance;
- operating limits and conditions, EMIT and commissioning activities required or assumed by the fuel and core safety case are captured adequately in a technical sense but further work is required post-GDA to ensure all such information is sufficiently clear and traceable; and
- an explicit demonstration that the reactor fuel and core designs reduce risks ALARP has been provided, which addresses the key expectations for new reactor designs within ONR ALARP guidance and is adequate for GDA.

759. In areas where my assessment uncovered shortfalls, I have followed the decision-making guidance in ONR-GEN-IN-021 – Identification and Management of GDA Issues, Assessment Findings and minor shortfalls for the GDA of UK HPR1000 (Ref. 5). I have raised a total of 11 Assessment Findings to be addressed by the licensee.

760. The most significant category of shortfall in Ref. 5 is a GDA Issue. I have not identified any GDA Issues because I did not judge that any of the shortfalls uncovered were significant enough to meet the definition of a GDA Issue provided by Ref. 5.

761. Except where I have identified minor shortfalls or Assessment Findings, I am satisfied that the expectations I derived from key SAPs, TAGs and other sources of RGP have been met by the UK HPR1000 fuel and core designs in the areas I have sampled.

762. Overall, based on my sample assessment of the safety case for the generic UK HPR1000 design undertaken in accordance with ONR's procedures, I am satisfied that the case presented within the PCSR and supporting documentation is adequate. On this basis, I am content that a DAC should be granted for the generic UK HPR1000 design from a Fuel and Core perspective.

5.2 Recommendations

763. Based upon my assessment detailed in this report, I recommend that:

- **Recommendation 1**: From a Fuel and Core perspective, ONR should grant a DAC for the generic UK HPR1000 design.

- **Recommendation 2**: The 11 Assessment Findings identified in this report should be resolved by the licensee for a site-specific application of the generic UK HPR1000 design.

6 REFERENCES

1. *New nuclear reactors: Generic Design Assessment: Guidance to Requesting Parties for the UK HPR1000*. ONR-GDA-GD-001. Revision 4. ONR. October 2019. www.onr.org.uk/new-reactors/ngn03.pdf

2. *Safety Assessment Principles for Nuclear Facilities*. 2014 Edition, Revision 1. ONR. January 2020. http://www.onr.org.uk/saps/saps2014.pdf

3. *Pre Construction Safety Report – Chapter 5 – Reactor Core*. HPR/GDA/PCSR/0005. Revision 1. CGN. January 2020. CM9 Ref. 2020/13630.

4. *Guidance on Mechanics of Assessment*. NS-TAST-GD-096. Revision 0. ONR. April 2020. http://www.onr.org.uk/operational/tech_asst_guides/index.htm

5. *Identification and Management of GDA Issues, AFs and MS for the GDA of UK HPR1000 - Document Number 1830*. ONR-GEN-IN-021. Revision 0. ONR. January 2021. CM9 Ref. 2021/3583.

6. *GDA Step 4 Assessment Plan of Fuel & Core topic for the UK HPR1000 Reactor*. ONR-GDA-UKHPR1000-AP-19-013. Revision 0. ONR. January 2020. CM9 Ref. 2019/357847.

7. *Step 4 Assessment of Fault Studies for the UK HPR1000 Reactor*. ONR-NR-AR-21-014. Revision 0. January 2022. ONR. CM9 Ref. 2021/44803.

8. *Step 4 Assessment of Severe Accident Analysis for the UK HPR1000 Reactor*. ONR-NR-AR-21-008. Revision 0. January 2022. ONR. CM9 Ref. 2021/44803.

9. *Step 4 Assessment of Radiation Protection & Criticality for the UK HPR1000 Reactor*. ONR-NR-AR-21-022. Revision 0. January 2022. ONR. CM9 Ref. 2021/52054.

10. *New Nuclear Power Plants: Generic Design Assessment Technical Guidance*. ONR-GDA-GD-007. Revision 0. May 2019. ONR. http://www.onr.org.uk/new-reactors/reports/onr-gda-007.pdf

11. *Safety of Nuclear Fuel in Power Reactors*. NS-TAST-GD-075. Revision 3. ONR. October 2020.

12. *Guidance on the Demonstration of ALARP*. NS-TAST-GD-005. Revision 11. ONR. November 2020.

13. *Design Basis Analysis*. NS-TAST-GD-006. Revision 5. ONR. October 2020.

14. *Validation of Computer Codes and Calculation Methods*. NS-TAST-GD-042. Revision 4. ONR. March 2019.

15. *Safety Aspects Specific to Storage of Spent Nuclear Fuel*. NS-TAST-GD-081. Revision 3. ONR. June 2019.

16. *Categorisation of Safety Functions and Classification of Structures, Systems and Components*. NS-TAST-GD-094. Revision 1. ONR. July 2019.

17. *Safety of Nuclear Power Plants: Design*. Specific Safety Requirements No. SSR-2/1. IAEA. Vienna. 2016. www.iaea.org.

18. *Deterministic Safety Analysis for Nuclear Power*. Specific Safety Guide No. SSG-2. IAEA. Vienna. 2019.

19. *Design of the Reactor Core for Nuclear Power Plants*. Specific Safety Guide No. SSG-52. IAEA. Vienna. 2019.

20. *Safety of new NPP designs*. WENRA Reactor Harmonization Working Group. March 2013. www.wenra.eu.

21. *Nuclear Fuel Safety Criteria Technical Review*. NEA No. 7072. OECD/NEA. Second Edition. 2012. https://www.oecd-nea.org

22. *UK HPR1000 – ONR396 Fault Studies Confirmatory Analysis – D2.2 Phase 2 Technical Report Summary of Confirmatory Analysis results*. Revision 1. GRS. April 2021. CM9 Ref. 2021/33581.

23. *UK HPR1000 Design Reference Report*. NE15BW-X-GL-0000-000047. Revision I. CGN. September 2021. CM9 Ref. 2021/68330.

24. *Production Strategy for Fuel and Core Design*. GHX00100021KPGB03GN. Revision H. CGN. April 2021. CM9 Ref. 2021/32273.

25. *UK HPR1000 - GDA Step 3 Summary Report*. ONR-NR-AR-19-001. Revision 0. ONR. February 2020. CM9 Ref. 2020/11336.

26. *Nuclear Design Basis*. GHX00600001DRDG03GN. Revision B. CGN. November 2019. CM9 Ref. 2019/354654.

27. *Nuclear Design Report for First Cycle*. GHX00600001DRDG03TR. Revision B. CGN. January 2020. CM9 Ref. 2020/21422.

28. *Nuclear Design Report for Equilibrium Cycle*. GHX00600002DRDG03TR. Revision B. CGN. January 2020. CM9 Ref. 2020/21419.

29. *Fuel Management Report*. GHX00600009DRDG03GN. Revision B. CGN. November 2019. CM9 Ref. 2019/332524.

30. *ALARP Demonstration Report of PCSR Chapter 05*. GHX00100048KPGB03GN. Rev I. CGN. May 2021. CM9 Ref. 2021/43521.

31. *Operating Experience with AFA 3GAA Fuel Assemblies*. FS1-0043880. Revision 3.0. Framatome. May 2020. CM9 Ref. 2020/142655.

32. *Verification and Validation of the Fuel System Design Software – COPERNIC*. FS1-0044846. Revision 3.0. Framatome. November 2020. CM9 Ref. 2020/311143.

33. *UK HPR1000 – Regulatory Query (RQ) Tracking Sheet*. November 2021. ONR. CM9 Ref. 2017/407871.

34. *Void Fraction for Fuel Assembly Equipped with M5 Cladding Tube*. FS1-0044886. Revision 3.0. Framatome. May 2020. CM9 Ref. 2020/142652.

35. *Thermal and Boiling Parameters of UK HPR1000 Core Designs*. GHX00100060DRAF03GN. Revision A. CGN. September 2020. CM9 Ref. 2020/259661.

36. *Neutronic Data for LOCA Analysis*. GHX00600015DRDG03GN. Revision A. CGN. June 2020. CM9 Ref. 2020/195833.

37. *Supporting report on ALARP Assessment for DNB analysis*. GHX00120001DRAF00GN. Revision D. CGN. March 2021. CM9 Ref. 2021/28197.

38. *Reactor Coolant Pump Seizure (Locked Rotor) or Reactor Coolant Pump Shaft Break*. GHX00600049DRAF02GN. Revision C. CGN. November 2019. CM9 Ref. 2019/352790.

39. *Synthesis Report on Positive Moderator Temperature Coefficient*. GHX00600369DRAF02GN. Revision A. CGN. September 2020. CM9 Ref. 2020/259776.

40. *Post-GDA Commitment List*. GHX00100084KPGB03GN. Revision C. CGN. August 2021. CM9 Ref. 2021/65125.

41. *Spectrum of RCCA Ejection Accident*. GHX00600092DRAF02GN. Revision C. CGN. November 2019. CM9 Ref. 2019/352936.

42. *Analysis of the Potential Gaps due to the UK Grid Code Requirement*. GHX00600003DRAF03GN. Revision C. May 2021. CM9 Ref. 2021/43593.

43. *Steam System Piping Large Break*. GHX00600081DRAF02GN. Revision C. CGN. November 2019. CM9 Ref. 2019/352947.

44. *AFA 3GAA Fuel Assembly Description for HPR1000 Reactor*. FS1-0043782. Revision 3.0. Framatome. August 2019. CM9 Ref. 2019/224212.

45. *UK HPR1000 HARMONI RCCA - Description, Functional Requirements and Material properties*. FS1-0044917. Revision 2.0. Framatome. February 2021. CM9 Ref. 2021/14344.

46. *Fuel Loading Sequence Analysis of Cycle 1*. GHX00600013DRDG03GN. Revision A. CGN. March 2020. CM9 Ref. 2020/92078.

47. *Optioneering Report for Secondary Neutron Source*. GHX00100001DRRL03TR. Revision A. CGN. August 2020. CM9 Ref. 2020/258579.

48. *General Nuclear Data and Key Neutronic Data Report*. GHX00600014DRDG03GN. Revision B. CGN. November 2020. CM9 Ref. 2020/314520.

49. *Power Envelop for Frequent Fault*. GHX00600021DRDG03GN. Revision B. CGN. November 2020. CM9 Ref. 2020/314517.

50. *Decay Heat Report*. GHX00100047DRAF03GN. Revision B. CGN. November 2019. CM9 Ref. 2019/355865.

51. *Uncontrolled RCCA Bank Withdrawal at Power*. GHX00600094DRAF02GN. Revision C. CGN. November 2019. CM9 Ref. 2019/352916.

52. *Excessive Increase in Secondary Steam Flow*. GHX00600082DRAF02GN. Revision D. CGN. November 2019. CM9 Ref. 2019/352956.

53. *Uncontrolled RCCA Bank Withdrawal at a Subcritical or Low Power Startup Condition (State A).* GHX00600095DRAF02GN. Revision C. CGN. November 2019. CM9 Ref. 2019/352921.

54. *Assessment of Fuel Crud for UK HPR1000*. GHX00100061DRAF03GN. Revision B. CGN. March 2021. CM9 Ref. 2021/22414.

55. *LOCUST - A Thermal-hydraulic System Analysis Code: Verification and Validation Report*. GHX00600143DRAF02TR. Revision D. CGN. March 2021. CM9 Ref. 2021/28217.

56. *Decay Heat Power in Light Water Reactors* (withdrawn). ANS 5.1-1994. American Nuclear Society. January 1994.

57. *Step 4 Assessment of Electrical Engineering for the UK HPR1000 Reactor*. ONR-NR-AR-21-011. Revision 0. January 2022. ONR. CM9 Ref. 2021/51507.

58. *Functional Requirements of the RGL [RPICS].* GHX00600010DRDG03GN. Revision B. CGN. January 2021. CM9 Ref. 2021/7683.

59. *RCCA Misalignment up to Rod Drop Without Limitation*. GHX00600091DRAF02GN. Revision C. CGN. November 2019. CM9 Ref. 2019/352893.

60. *UK HPR1000 - ALARP Demonstration of Fuel System*. FS1-0044065. Revision 4.0. Framatome. July 2021. CM9 Ref. 2021/57820.

61. *SCCA - Description, Functional Requirements and Material Properties*. FS1-0044918. Revision 4.0. Framatome. March 2021. CM9 Ref. 2021/25887.

62. *Review of Fuel Failures in Water-Cooled Reactors (2006-2015).* IAEA Nuclear Energy Series NF-T-2.5. IAEA. 2019.

63. *The Design Condition List and Acceptance Criteria*. GHX00100029DOZJ04GN. Revision J. CGN. January 2021. CM9 Ref. 2021/2094.

64. *Step 4 Assessment of Mechanical Engineering for the UK HPR1000 Reactor*. ONR-NR-AR-21-004. Revision 0. January 2022. ONR. CM9 Ref. 2021/53696.

65. *Demonstration report for the fuel failure mechanism in fuel route*. GHX42500020SFSL44GN. Revision 003. Framatome. February 2021. CM9 Ref. 2021/14348.

66. *Long Term Storage of Spent Fuel - Design Criteria*. FS1-0052351. Framatome. Revision 004. July 2021. CM9 Ref. 2021/54452.

67. *Supporting Report on No Fuel Failure for Frequent Faults*. GHX00600274DRAF02GN. Revision E. CGN. July 2021. CM9 Ref. 2021/53914.

68. *Topical Report on Grid to Rod Fretting Performance*. GHX42500033SFSL44GN. Revision 003. Framatome. January 2021. CM9 Ref. 2021/8885.

69. *Fuel Assembly Mechanical Design*. FS1-0049660. Revision 2.0. Framatome. September 2020. CM9 Ref. 2020/259703.

70. *UK HPR1000 - Fuel Assembly Anti-Debris Filter Hydraulic Test*. FS1-0043994. Revision 2.0. Framatome. August 2019. CM9 Ref. 2019/224251.

71. *UK HPR1000 - Fuel Rod Design*. FS1-0051388. Revision 3.0. Framatome. December 2020. CM9 Ref. 2020/323643.

72. *Thermal Hydraulic Performance at Fuel Assembly Edge*. RO-UKHPR1000-0045. Revision 0. ONR. May 2020. CM9 Ref. 2020/114823.

73. *UK HPR1000 - AFA 3GAA fuel assembly for HPR1000 reactor - Assessment of water gap core distribution*. FS1-0051497. Revision 3.0. Framatome. August 2021. CM9 Ref. 2021/60069.

74. *The Effect of Fuel Assembly Bow on DNBR*. GHX00600360DRAF02GN. Revision C. CGN. June 2021. CM9 Ref. 2021/44479.

75. *Fuel Assembly Mechanical Design Methodology*. GHX42500002SFSL44GN. Revision 2.0. Framatome. August 2019. CM9 Ref. 2019/252668.

76. *CASAC 5.3 - Qualification Summary Report*. FS1-0034319. Revision 1.0. Framatome. November 2017. CM9 Ref. 2021/33363.

77. *Asymmetric Blowdown Loads on PWR Primary Systems – Resolution of Generic Task Action Plan A-2.* NUREG-0609. US NRC. 1981. http://wwww.nrc.gov

78. *Step 4 Assessment of External Hazards for the UK HPR1000 Reactor*. ONR-NR-AR-21-006. Revision 0. January 2022. ONR. CM9 Ref. 2021/46598.

79. *UK HPR1000 - RCCA Mechanical Design Report*. FS1-0050192. Revision 2.0. November 2020. CM9 Ref. 2020/315784.

80. *Operating Experience with Harmoni RCCA*. FS1-0044149. Revision 1.0. Framatome. August 2019. CM9 Ref. 2019/233001.

81. *Step 4 Assessment of Structural Integrity for the UK HPR1000 Reactor*. ONR-NR-AR-21-016. Revision 0. January 2022. ONR. CM9 Ref. 2021/52300.

82. *UK HPR1000 - PCI Technological Limit for Fuel Rods with M5$_{Framatome}$ Cladding*. FS1-0044493. Revision 1.0. Framatome. July 2019. CM9 Ref. 2019/209440.

83. *Fuel Rod Design Methodology*. FS1-0044752. Revision 003. Framatome. February 2021. CM9 Ref. 2021/14347.

84. *Start of the Art Report on Nuclear Fuel Behaviour in Loss-of-coolant-accident (LOCA) conditions*. NEA No. 6846. OECD/NEA. 2009. https://www.oecd-nea.org

85. *ALARP Assessment for DBC Radiological Consequence*. GHX00600375DRAF02GN. Revision B. CGN. April 2021. CM9 Ref. 2021/31456.

86. *Peak Cladding Temperature Criteria for Fuel*. FS1-0048927. Revision 1.0. Framatome. June 2020. CM9 Ref. 2020/164028.

87. *Fuel Rod PCMI failure criterion for RCCA ejection*. FS1-0045252. Revision 003. Framatome. February 2021.

88. *Pressurized-Water Reactor Control Rod Ejection and Boiling-Water Reactor Control Rod Drop Accidents*. RG 1.236. US NRC. 2020. http://wwww.nrc.gov

89. *CABRI CIP programme*. https://www.irsn.fr/EN/Research/Research-organisation/Research-programmes/CABRI-International-program/Pages/CABRI-CIP-program.aspx

90. *Spent Fuel Interim Storage Facility Design*. GHX00100081DNFF03GN. Revision H. CGN. May 2021. CM9 Ref. 2021/43518.

91. *Preliminary Safety Evaluation of Spent Fuel Interim Storage*. GHX00100046DNFP03GN. Revision H. CGN. May 2021. CM9 Ref. 2021/43511.

92. *UK HPR1000 GDA Scope for Spent Fuel Interim Storage (SFIS).* REG-GNS-0031N. ONR & Environment Agency. October 2018. CM9 Ref. 2018/329187.

93. *Step 4 Assessment of Spent Fuel Interim Storage for the UK HPR1000 Reactor*. ONR-NR-AR-21-017. Revision 0. January 2022. ONR. CM9 Ref. 2021/51327.

94. *Fuel Handling and Storage System Design Manual Chapter 4, System and Component Design*. GHX17PMC004DPFJ45GN. Revision D. CGN. February 2021. CM9 Ref. 2021/16423.

95. *Thermal Hydraulic Design*. GHX00100004DRRG03GN. Revision F. CGN. March 2021. CM9 Ref. 2021/28190.

96. *DNBR Design Limit*. GHX00100001DRRG03GN. Revision H. CGN. August 2020. CM9 Ref. 2020/232820.

97. *UK HPR1000 - FC2000 CHF Correlation*. FS1-0045927. Revision 3.0. Framatome. July 2020. CM9. Ref. 2020/207176.

98. *W3 CHF correlation*. FS1-0046052. Revision 2.0. Framatome. December 2019. CM9 Ref. 2019/378761.

99. *LINDEN - A Subchannel Analysis Code: Verification and Validation Report*. GHX00600142DRAF02TR. Revision E. CGN. January 2021. CM9 Ref. 2021/8988.

100. *LINDEN Detailed Code Review for the UK HPR1000 Safety Case on behalf of ONR*. ONRTSF/4NT/0711846/000/01. Tractebel. November 2020. CM9 Ref. 2020/309367.

101. *Turbulence Diffusion Coefficients*. FS1-0045294. Revision 2.0. Framatome. December 2019. CM9 Ref. 2019/378765.

102. *Justification with W3 CHF Correlation of one SLB Transient State Point*. FS1-0046280. Revision 2.0. Framatome. February 2020. CM9 Ref. 2020/55167.

103. *UK HPR1000 - Melting Point of UO2 and (U,Gd) O2 Fuels*. FS1-0043959. Revision 1.0. Framatome. September 2019. CM9 Ref. 2019/282288.

104. *Core Hydrodynamic Instability Analysis*. GHX00600294DRAF02GN. Revision B. CGN. November 2019. CM9 Ref. 2019/357677.

105. *Decrease in Boron Concentration in Reactor Coolant due to malfunction of RCV [CVCS], REA [RBWMS] and TEP [CSTS].* GHX00600002DRDG02GN. Revision F. CGN. April 2021. CM9 Ref. 2021/34065.

106. *Interface Data for Reactor Design*. FS1-0043303. Revision 3.0. Framatome. April 2020. CM9 Ref. 2020/105087.

107. *The Assessment Report of Burn-up Limit for DNBR Calculation*. GHX00600322DRAF02GN. Revision B. CGN. April 2020. CM9 Ref. 2020/105076.

108. *UK HPR1000 - AFA 3GAA fuel assembly for HPR1000 reactor - CHF in peripheral region*. FS1-0054052. Revision 2.0. Framatome. August 2021. CM9 Ref. 2021/64971.

109. *UK HPR1000 - AFA 3GAA fuel assembly for HPR1000 reactor - Local Pressure Loss Coefficients*. FS1-0056154. Revision 2.0. Framatome. August 2021. CM9 Ref. 2021/64970.

110. *Assessment of the response to RO-UKHPR1000-0045 – Thermal hydraulic performance at fuel assembly edge*. ONR-NR-AN-21-043. Revision 0. ONR. July 2021. CM9 Ref. 2021/34180.

111. *Overtemperature T and Overpower T Protection Setpoints Design*. GHX00100002DRRG03GN. Revision F. CGN. March 2021. CM9 Ref. 2021/26821.

112. *Power Capability Analysis Report*. GHX00600012DRDG02GN. Revision B. CGN. March 2020. CM9 Ref. 2020/73780.

113. *Expected Number of Fuel Rods Experiencing DNB with Statistical Method*. GHX00600323DRAF02GN. Revision B. June 2020. CM9 Ref. 2020/181916.

114. *Pressure Drop and Bypass Flow*. GHX00100044DRAF03GN. Revision B. CGN. November 2019. CM9 Ref. 2019/357553.

115. *Synthesis of Pressure Loss Measurements in Support of Design Analysis*. FS1-0044472. Revision 3.0. Framatome. December 2019. CM9 Ref. 2019/375714.

116. *The Delivery of UK HPR1000 GDA Design Modification-Cat1 "Modification of OPT Setpoint for Resolution on PCI Negative Margin Case (M88-GHTCN000198-A).* HPR-GDA-LETT-0110. GNSL. April 2021. CM9 Ref. 2021/31029.

117. *PCI Transient Analysis*. GHX00600383DRAF02GN. Revision B. CGN. March 2021. CM9 Ref. 2021/27281.

118. *UK HPR1000 - PCI Thermal-Mechanical Analysis*. FS1-0050581. Revision 4.0. Framatome. January 2021. CM9 Ref. 2021/3513.

119. *UK HPR1000 - PCI Thermal-Mechanical Analysis in Frequent Fault*. FS1-0051971. Revision 2.0. Framatome. March 2021. CM9 Ref. 2021/23951.

120. *Non-baseload Operation in Nuclear Power Plants: Load Following and Frequency Control Modes of Flexible Operation*. IAEA Nuclear Energy Series NPT-T-3.23. 2018. IAEA.

121. *Resolution Report of PCI Risk for INVDA*. GHX00600388DRAF02GN. Revision B. CGN. April 2021. CM9 Ref. 2021/36146.

122. *Verification of PCI bias for +-10%FP Primary Frequency Control*. FS1-0056071. Revision 1.0. Framatome. April 2021. CM9 Ref. 2021/35871.

123. *Review of the established good practices in development of clad ballooning and embrittlement models for higher burnup PWR fuel*. File Ref. 35.2.2. Version 4. Checkendon Hill. 2016. https://www.onr.org.uk/documents/2016/onr-rrr-014.pdf

124. *Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors*. 10 CFR 50.46. US NRC. As amended 2007. http://wwww.nrc.gov

125. *Decoupling LOCA Criteria for Fuel*. FS1-0046539. Revision 2.0. Framatome. March 2020. CM9 Ref. 2020/79659.

126. *LB-LOCA safety case clarification and core assessment report*. GHX00600007DRRL02GN. Revision C. CGN. June 2021. CM9 Ref. 2021/46979.

127. *Internal Events Level 1 PSA*. GHX00650001DOZJ02GN. Revision C. CGN. July 2021. CM9 Ref. 2021/58822.

128. *Large Break - Loss of Coolant Accident (up to double-ended break).* GHX00600004DRAF02GN. Revision C. CGN. November 2019. CM9 Ref. 2019/352765.

129. *Step 4 Assessment of Probabilistic Safety Analysis for the UK HPR1000 Reactor*. ONR-NR-AR-21-020. Revision 0. January 2022. ONR. CM9 Ref. 2021/49362.

130. *Intermediate Break and up to Surge Line Break - Loss of Coolant Accident*. GHX00100042DRAF03GN. Revision D. CGN. July 2021. CM9 Ref. 2021/58477.

131. *UK HPR1000 - RCA7 - Fuel behavior model in LOCA conditions - cladding deformation*. Revision 1.0. Framatome. July 2020. CM9 Ref. 2021/33627.

132. *A State-Of-The-Art Review of Past Programs Devoted to Fuel Behavior Under LOCA Conditions*. IRSN. 2005. http://www.irsn.fr/

133. *Report on Fuel Fragmentation, Relocation, Dispersal (FFRD).* NEA/CSNI/R(2016)16. OECD/NEA. 2016.

134. *Fuel Fragmentation, Relocation, and Dispersal During the Loss-of-Coolant Accident*. NUREG 2121. US NRC. 2012. http://www.nrc.gov/

135. *Demonstration that Risks Associated with Fuel Deposits are Reduced So Far As Is Reasonably Practicable (SFAIRP).* RO-UKHPR1000-0015. Revision 0. ONR. September 2019. CM9 Ref. 2019/224037.

136. *Step 4 Assessment of Chemistry for the UK HPR1000 Reactor*. ONR-NR-AR-21-002. Revision 0. January 2022. ONR. CM9 Ref. 2021/41488.

137. *Assessment of the response to RO-UKHPR1000-0015*. ONR-NR-AN-21-002. Revision 0. ONR. May 2021. CM9 Ref. 2021/17067.

138. *Status on CRUD Monitoring and Acceptability*. FS1-0045777. Revision 1.0. Framatome. November 2019. CM9 Ref. 2019/320743.

139. *UK HPR1000 GDA on CRUD in-core Deposits on Fuel Assemblies and their Risk Mitigation Worldwide Operational Experience - Supporting Information*. D-FD-20-01041. Revision 1.0. Framatome. May 2020. CM9 Ref. 2020/161713.

140. *Inadvertent Core Loading of Fuel Assemblies*. GHX00600006DRDG02GN. Revision C. CGN. November 2020. CM9 Ref. 2020/314660.

141. *Assessment of core misloading preventing strategy*. GHX00600032DRDG03GN. Revision A. CGN. January 2021. CM9 Ref. 2021/3515.

142. *Functional Requirements of the RPN [NIS].* GHX00600012DRDG03GN. Revision A. CGN. October 2019. CM9 Ref. 2019/285581.

143. *Functional Requirements of the RIC [IIS].* GHX00600011DRDG03GN. Revision B. CGN. November 2020. CM9 Ref. 2020/309227.

144. *RIC - In-core Instrumentation System Design Manual Chapter 4 System and Component Design*. GHX17RIC004DIYK45GN. Revision B. CGN. August 2020. CM9 Ref. 2020/233016.

145. *Methodology of Safety Categorisation and Classification*. GHX00100062DOZJ03GN. Revision B. June 2018. CM9 Ref. 2018/199731.

146. *Modification of Self Powered Neutron Detector (SPND) Sub-system Category*. HPR-GDA-LETT-0087. CGN. November 2020. CM9 Ref. 2020/312196.

147. *Self-Powered Neutron Detector Categorisation*. GHX00600029DRDG03GN. Revision B. CGN. March 2021. CM9 Ref. 2021/27280.

148. *Failure Mode and Failure Consequence of SPND Sub-system*. GHX00600030DRDG03GN. Revision A. CGN. December 2020. CM9 Ref. 2020/319234.

149. *Step 4 Assessment of Control & Instrumentation for the UK HPR1000 Reactor*. ONR-NR-AR-21-005. Revision 0. January 2022. ONR. CM9 Ref. 2021/46296.

150. *Generic water chemistry specification (LCO).* GHX00100101DCHS03GN. Revision E. CGN. May 2021. CM9 Ref. 2021/43591.

151. *Radiochemistry Parameters Value*. GHX00100057DNFP03GN. Revision E. CGN. July 2021. CM9 Ref. 2021/55619.

152. *Optioneering on the Reduction of SGTR Radiological Consequence*. GHX00600370DRAF02GN. Revision C. CGN. April 2021. CM9 Ref. 2021/31455.

153. *PCSR - Chapter 30 – Commissioning*. HPR/GDA/PCSR/0030. Revision 001. CGN. January 2020. CM9 Ref. 2020/13992.

154. *Development of a Suitable and Sufficient Safety Case*. RO-UKHPR1000-0004. Revision 0. ONR. September 2018. CM9 Ref. 2018/255957.

155. *The Purpose, Scope and Content of Safety Cases*. NS-TAST-GD-051. Revision 7. December 2019. ONR.

156. *Step 4 Assessment of Cross-cutting Topics for the UK HPR1000 Reactor*. ONR-NR-AR-21-007. Revision 0. January 2022. ONR. CM9 Ref. 2021/47905.

157. *PCSR - Chapter 31 - Operational Management*. HPR/GDA/PCSR/003. Revision 001. CGN. January 2020. CM9 Ref. 2020/13995.

158. *Generic Limits and Conditions for Normal Operation*. GHX37OTS001DOYX45GN. Revision B. CGN. May 2021. CM9 Ref. 2021/37697.

159. *Code Review for the UK HPR1000 Safety Case on behalf of ONR - Software Quality Assurance*. ONRTSF/4NT/0685039/000/00. Tractebel. October 2019. CM9 Ref. 2019/307311.

160. *UK HPR1000 – ONR396 Fault Studies Confirmatory Analysis – D2.2.10 Phase 2 Technical Report Investigation of core physics discrepancies*. Revision 1. GRS. April 2021. CM9 Ref. 2021/33579.

161. *COCO - A 3-D Nuclear Design Code: Verification and Validation Report*. GHX00600001DRDG02TR. Revision D. CGN. November 2020. CM9 Ref. 2020/314849.

162. *COCO Code Review for the UK HPR1000 Safety Case on behalf of ONR*. ONRTSF4NT068496800001. Tractebel. March 2020. CM9 Ref. 2020/96987.

163. *Core Reflector Calculation Report*. GHX00600022DRDG03GN. Revision A. CGN. June 2020. CM9 Ref. 2020/195830.

164. *PINE - A Lattice Physics Code: Verification and Validation Report*. GHX00600004DRDG02TR. Revision D. CGN. November 2020. CM9 Ref. 2020/314867.

165. *PINE Code Review for the UK HPR1000 Safety Case on behalf of ONR*. ONRTSF4NT068502400001. Tractebel. March 2020. CM9 Ref. 2020/96991.

166. *UK HPR1000 – ONR396 Fault Studies Confirmatory Analysis – D2.2.09 Phase 2 Technical Report RCCA Misalignment up to Rod Drop without Limitation*. Revision 1. GRS. March 2021. CM9 Ref. 2021/30135.

167. *UK HPR1000 – ONR396 Fault Studies Confirmatory Analysis – D2.2.08 Phase 2 Technical Report Uncontrolled RCCA Bank Withdrawal at Low Power Startup Conditions*. Revision 1. GRS. March 2021. CM9 Ref. 2021/30133.

168. *POPLAR - A 1-D Core Calculation Code: Verification and Validation Report*. GHX00600003DRDG02TR. Revision D. CGN. November 2020. CM9 Ref. 2020/314863.

169. *POPLAR Code Review for the UK HPR1000 Safety Case on behalf of ONR*. ONRTSF4NT068502600001. Tractebel. March 2020. CM9 Ref. 2020/97003.

170. *PALM - A Depletion Calculation Code: Verification and Validation Report*. GHX00600002DRDG02TR. Revision D. CGN. November 2020. CM9 Ref. 2020/314859.

171. *PALM Code Review for the UK HPR1000 Safety Case on behalf of ONR*. ONRTSF4NT068502500001. Tractebel. March 2020. CM9 Ref. 2020/96996.

172. *BIRCH - A Fuel Rod Temperature Analysis Code Verification and Validation Report*. GHX00600138DRAF02TR. Revision E. CGN. May 2021. CM9 Ref. 2021/37754.

173. *BIRCH Code Review for the UK HPR1000 Safety Case on behalf of ONR*. ONRTSF4NT068502800001. Tractebel. March 2020. CM9 Ref. 2020/97018.

174. *LINDEN - A Subchannel Analysis Code: Verification and Validation Report*. GHX00600142DRAF02TR. Revision E. CGN. January 2021. CM9 Ref. 2021/8988.

175. *UK HPR1000 ALARP Methodology*. GHX00100051DOZJ03GN. Revision D. April 2020. CM9 Ref. 2020/139106.

176. *UK HPR1000 Suitability Analysis of Codes and Standards in Fuel Design*. FS1-0044927. Rev 3.0. Framatome. July 2021. CM9 Ref. 2021/57081.

177. *Compliance Analysis of Codes and Standards in Fuel and Core Design*. GHX00800005DRRL02GN. Revision B. CGN. January 2021. CM9 Ref. 2021/8884.

178. *Design of the Reactor Core for Nuclear Power Plants*. IAEA Safety Standards Series No. NS-G-1.12. IAEA. 2005.

179. *UK HPR1000 - Master Document Submission List*. HPR/GDA/REPO/0197. Rev 001. GNSL. November 2021. CM9 Ref. 2021/84821.

180. *Pre-Construction Safety Report, Chapter 5, Reactor Core*. HPR-GDA-PCSR-0005. Revision 002. GNSL. September 2021. CM9 Ref. 2021/72679.

Annex 1

Relevant Safety Assessment Principles Considered During the Assessment

SAP No	SAP Title	Description
MS.2	Leadership and management for safety. Capable organisation.	The organisation should have the capability to secure and maintain the safety of its undertakings.
MS.3	Leadership and management for safety. Decision making.	Decisions made at all levels in the organisation affecting safety should be informed, rational, objective, transparent and prudent.
SC.4	The regulatory assessment of safety cases. Safety case characteristics.	A safety case should be accurate, objective and demonstrably complete for its intended purpose.
EKP.1	Engineering principles: key principles. Inherent safety.	The underpinning safety aim for any nuclear facility should be an inherently safe design, consistent with the operational purposes of the facility.
EKP.2	Engineering principles: key principles. Fault tolerance.	The sensitivity of the facility to potential faults should be minimised.
EKP.3	Engineering principles: key principles. Defence in depth.	Nuclear facilities should be designed and operated so that defence in depth against potentially significant faults or failures is achieved by the provision of multiple independent barriers to fault progression.
EKP.4	Engineering principles: key principles. Safety function.	The safety function(s) to be delivered within the facility should be identified by a structured analysis.
ECS.1	Engineering principles: safety classification and standards. Safety categorisation.	The safety functions to be delivered within the facility, both during normal operation and in the event of a fault or accident, should be identified and then categorised based on their significance with regard to safety.
EDR.2	Engineering principles: design for reliability. Redundancy, diversity and segregation.	Redundancy, diversity and segregation should be incorporated as appropriate within the designs of structures, systems and components.
ERL.1	Engineering principles: reliability claims. Form of claims.	The reliability claimed for any structure, system or component should take into account its novelty, experience relevant to its proposed environment, and uncertainties in operating and fault conditions, physical data and design methods.

SAP No	SAP Title	Description
EAD.1	Engineering principles: ageing and degradation. Safe working life.	The safe working life of structures, systems and components that are important to safety should be evaluated and defined at the design stage.
EAD.2	Engineering principles: ageing and degradation. Lifetime margins.	Adequate margins should exist throughout the life of a facility to allow for the effects of materials ageing and degradation processes on structures, systems and components.
ENM.6	Engineering principles: control of nuclear matter. Storage in a condition of passive safety.	When nuclear matter is to be stored on site for a significant period of time it should be stored in a condition of passive safety whenever practicable and in accordance with good engineering practice.
ERC.1	Engineering principles: reactor core. Design and operation of reactors.	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.
ERC.2	Engineering principles: reactor core. Shutdown systems.	At least two diverse systems should be provided for shutting down a civil reactor.
ERC.3	Engineering principles: reactor core. Stability in normal operation.	The core should be stable in normal operation and should not undergo sudden changes of condition when operating parameters go outside their permitted range.
ERC.4	Engineering principles: reactor core. Monitoring of parameters important to safety.	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.
ECR.1	Engineering principle: criticality safety. Safety measures.	Wherever a significant amount of fissile material may be present, there should be safety measures to protect against unplanned criticality.
FA.4	Fault analysis: design basis analysis. Fault tolerance.	DBA should be carried out to provide a robust demonstration of the fault tolerance of the engineering design and the effectiveness of the safety measures.
FA.7	Fault analysis: design basis analysis. Consequences.	Analysis of design basis fault sequences should use appropriate tools and techniques, and be performed on a conservative basis to demonstrate that consequences are ALARP.

SAP No	SAP Title	Description
FA.15	Fault analysis: severe accident analysis. Scope of severe accident analysis.	Fault states, scenarios and sequences beyond the design basis that have the potential to lead to a severe accident should be analysed.
AV.1	Fault analysis: assurance of validity of data and models. Theoretical models.	Theoretical models should adequately represent the facility and site.
AV.2	Fault analysis: assurance of validity of data and models. Calculation methods.	Calculation methods used for the analyses should adequately represent the physical and chemical processes taking place.
AV.3	Fault analysis: assurance of validity of data and models. Use of data.	The data used in the analysis of aspects of plant performance with safety significance should be shown to be valid for the circumstances by reference to established physical data, experiment or other appropriate means.
AV.5	Fault analysis: assurance of validity of data and models. Documentation.	Documentation should be provided to facilitate review of the adequacy of the analytical models and data.
AV.6	Fault analysis: assurance of validity of data and models. Sensitivity studies.	Studies should be carried out to determine the sensitivity of the analysis (and the conclusions drawn from it) to the assumptions made, the data used and the methods of calculation.

Annex 2

Assessment Findings

Note: These Assessment Findings must be read in the context of the sections of the report listed in this table, where further detail is provided regarding the matters that led to the findings being raised.

Number	Assessment Finding	Report Section
AF-UKHPR1000-0001	The licensee shall demonstrate that the combined design basis loss of coolant accident and site-specific seismic event analysed according to the fuel assembly mechanical design basis do not challenge the structural integrity of the fuel assemblies. Justification should be provided for the methods and assumptions used in the analysis.	4.4
AF-UKHPR1000-0002	The licensee shall justify technical acceptance criteria to prevent fuel fragmentation and minimise pellet-clad mechanical interaction induced fuel failures at all fuel burnups in a Rod Cluster Control Assembly ejection fault. The selection of underpinning experimental data should be justified.	4.5
AF-UKHPR1000-0003	The licensee shall, as part of detailed design and before the fuel management design is finalised, demonstrate that the fuel criteria required to ensure clad integrity during spent fuel interim storage operations can be met for limiting fuel rods, from the point of fuel leaving the UK HPR1000 spent fuel pool. Evidence should be provided to substantiate the selected criteria	4.5
AF-UKHPR1000-0004	The licensee shall validate the critical heat flux correlation limits and statistical parameters for the chosen critical heat flux correlation(s) in the sub-channel analysis code used to undertake fault analysis calculations. This should include an analysis of the underlying experimental data using that same code.	4.6

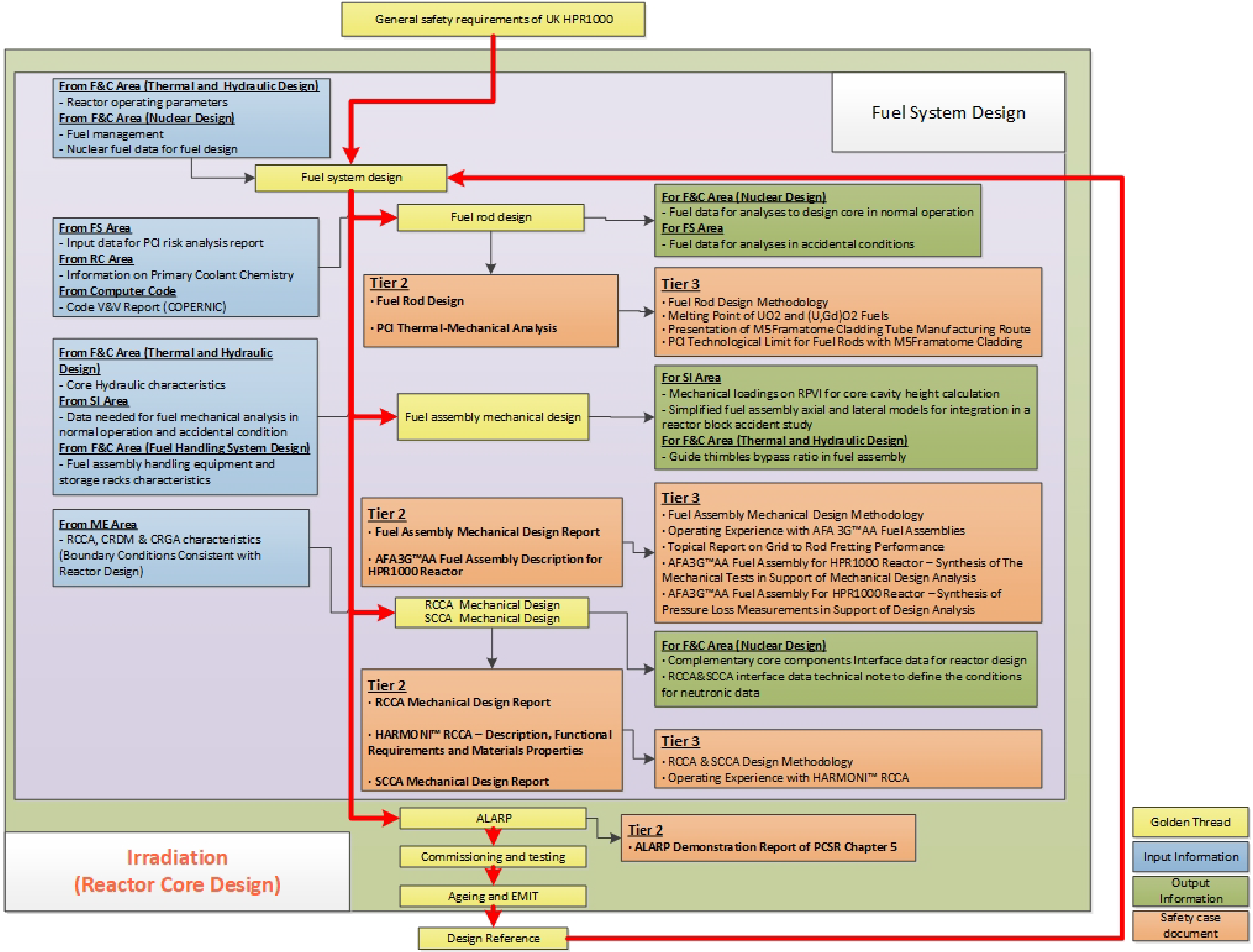
Number	Assessment Finding	Report Section
AF-UKHPR1000-0005	The licensee shall demonstrate that flow blockage due to clad ballooning will not result in a loss of coolable geometry in a large break loss of coolant accident. Fuel fragmentation, relocation and dispersal phenomena should also be addressed within the analysis, or their exclusion justified. Appropriate validation evidence should be provided for all analysis methods used. The analysis may use a reduced level of conservatism compared to that used for faults inside the UK HPR1000 design basis.	4.8
AF-UKHPR1000-0007	The licensee shall, as part of their operating procedures, implement a strategy for decision making in the event of a potential fuel failure being identified that defines the actions to be taken to reduce relevant risks to as low as reasonably practicable. The strategy should minimise the dispersal of nuclear material and limit further degradation of the cladding material.	4.11
AF-UKHPR1000-0008	The licensee shall include in the validation base for its chosen nuclear physics codes and models a range of comparisons with measured physics test data. This should include axial power distributions and Rod Cluster Control Assembly bank differential worth curves, from reactor core and fuel designs that are as close as practicable to those of UK HPR1000. If practical the measured data should come from HPR1000 plant(s). The comparisons should include measured data from cores containing both fresh and partially-burnt fuel.	4.13
AF-UKHPR1000-0009	The licensee shall, in the verification and validation evidence underlying the UK HPR1000 thermal hydraulic sub-channel code, present code-to-code comparisons for sub-channel local parameter predictions.	4.13
AF-UKHPR1000-0010	The licensee shall justify the UK HPR1000 temperature regulation Rod Cluster Control Assembly bank insertion limits and core axial offset limits together to demonstrate that the consequences of faults for fuel integrity have been reduced to as low as reasonably practicable.	4.14

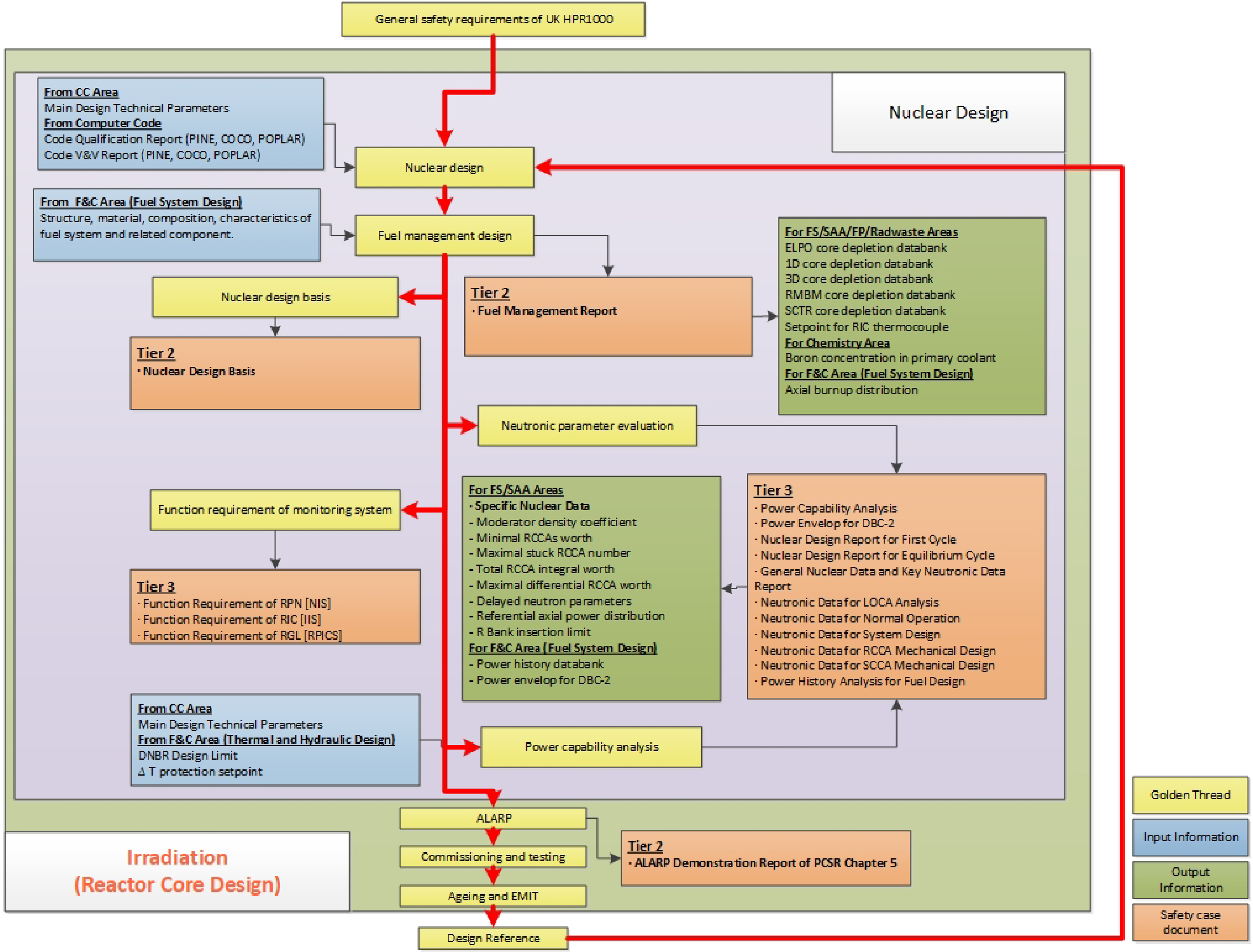
Number	Assessment Finding	Report Section
AF-UKHPR1000-0126	The licensee shall ensure that relevant fuel and core related implementable requirements are included in site-specific operating documentation and underpinned by the safety case. This should include, but not be limited to, those requirements relating to operating rules, examination, maintenance, inspection and testing requirements, commissioning tests and rules for core reload design which feature in the generic safety case but which are not yet identified and managed as implementable requirements.	4.12
AF-UKHPR1000-0127	The licensee shall demonstrate that core coolability and Rod Cluster Control Assembly insertion are not challenged due to mechanical damage to UK HPR1000 fuel assemblies and core components in a large break loss of coolant accident. The analysis may use a reduced level of conservatism compared to that used for faults inside the UK HPR1000 design basis.	4.8

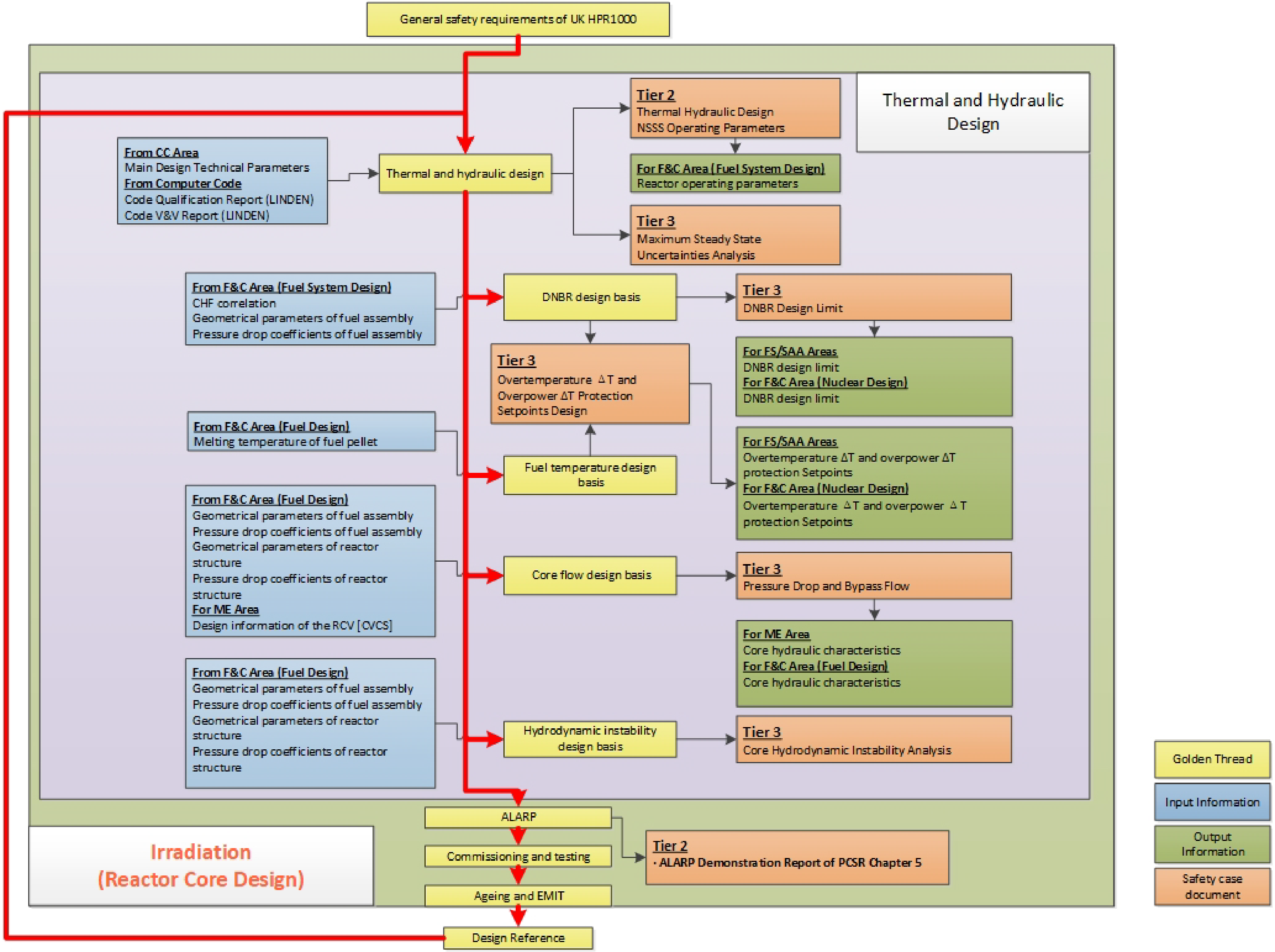
Annex 3

PCSR Chapter 5 Document Hierarchies (information extracted from Ref. 24)

(begins over page)


General safety requirements of UK HPR1000
Fuel System Design
From F&C Area (Thermal and Hydraulic Design)
- Reactor operating parameters
From F&C Area (Nuclear Design)
- Fuel management
- Nuclear fuel data for fuel design
Fuel system design
Fuel rod design
For F&C Area (Nuclear Design)
- Fuel data for analyses to design core in normal operation
For FS Area
- Fuel data for analyses in accidental conditions
From FS Area
- Input data for PCI risk analysis report
From RC Area
- Information on Primary Coolant Chemistry
From Computer Code
- Code V&V Report (COPERNIC)
Tier 2
• Fuel Rod Design
• PCI Thermal-Mechanical Analysis
Tier 3
• Fuel Rod Design Methodology
• Melting Point of UO2 and (U,Gd)O2 Fuels
• Presentation of M5Framatome Cladding Tube Manufacturing Route
• PCI Technological Limit for Fuel Rods with M5Framatome Cladding
From F&C Area (Thermal and Hydraulic Design)
- Core Hydraulic characteristics
From SI Area
- Data needed for fuel mechanical analysis in normal operation and accidental condition
From F&C Area (Fuel Handling System Design)
- Fuel assembly handling equipment and storage racks characteristics
Fuel assembly mechanical design
For SI Area
- Mechanical loadings on RPVI for core cavity height calculation
- Simplified fuel assembly axial and lateral models for integration in a reactor block accident study
For F&C Area (Thermal and Hydraulic Design)
- Guide thimbles bypass ratio in fuel assembly
From ME Area
- RCCA, CRDM & CRGA characteristics (Boundary Conditions Consistent with Reactor Design)
Tier 2
• Fuel Assembly Mechanical Design Report
• AFA3G™AA Fuel Assembly Description for HPR1000 Reactor
Tier 3
• Fuel Assembly Mechanical Design Methodology
• Operating Experience with AFA 3G™AA Fuel Assemblies
• Topical Report on Grid to Rod Fretting Performance
• AFA3G™AA Fuel Assembly for HPR1000 Reactor – Synthesis of The Mechanical Tests in Support of Mechanical Design Analysis
• AFA3G™AA Fuel Assembly For HPR1000 Reactor – Synthesis of Pressure Loss Measurements in Support of Design Analysis
RCCA Mechanical Design
SCCA Mechanical Design
For F&C Area (Nuclear Design)
• Complementary core components Interface data for reactor design
• RCCA&SCCA interface data technical note to define the conditions for neutronic data
Tier 2
• RCCA Mechanical Design Report
• HARMONI™ RCCA – Description, Functional Requirements and Materials Properties
• SCCA Mechanical Design Report
Tier 3
• RCCA & SCCA Design Methodology
• Operating Experience with HARMONI™ RCCA
ALARP
Tier 2
• ALARP Demonstration Report of PCSR Chapter 5
Commissioning and testing
Ageing and EMIT
Design Reference
Irradiation
(Reactor Core Design)
Golden Thread
Input Information
Output Information
Safety case document


General safety requirements of UK HPR1000
Nuclear Design
From CC Area
Main Design Technical Parameters
From Computer Code
Code Qualification Report (PINE, COCO, POPLAR)
Code V&V Report (PINE, COCO, POPLAR)
Nuclear design
From F&C Area (Fuel System Design)
Structure, material, composition, characteristics of fuel system and related component.
Fuel management design
Nuclear design basis
Tier 2
· Nuclear Design Basis
Tier 2
· Fuel Management Report
For FS/SAA/FP/Radwaste Areas
ELPO core depletion databank
1D core depletion databank
3D core depletion databank
RMBM core depletion databank
SCTR core depletion databank
Setpoint for RIC thermocouple
For Chemistry Area
Boron concentration in primary coolant
For F&C Area (Fuel System Design)
Axial burnup distribution
Neutronic parameter evaluation
Function requirement of monitoring system
Tier 3
· Function Requirement of RPN [NIS]
· Function Requirement of RIC [IIS]
· Function Requirement of RGL [RPICS]
For FS/SAA Areas
· Specific Nuclear Data
- Moderator density coefficient
- Minimal RCCAs worth
- Maximal stuck RCCA number
- Total RCCA integral worth
- Maximal differential RCCA worth
- Delayed neutron parameters
- Referential axial power distribution
- R Bank insertion limit
For F&C Area (Fuel System Design)
- Power history databank
- Power envelop for DBC-2
Tier 3
· Power Capability Analysis
· Power Envelop for DBC-2
· Nuclear Design Report for First Cycle
· Nuclear Design Report for Equilibrium Cycle
· General Nuclear Data and Key Neutronic Data Report
· Neutronic Data for LOCA Analysis
· Neutronic Data for Normal Operation
· Neutronic Data for System Design
· Neutronic Data for RCCA Mechanical Design
· Neutronic Data for SCCA Mechanical Design
· Power History Analysis for Fuel Design
From CC Area
Main Design Technical Parameters
From F&C Area (Thermal and Hydraulic Design)
DNBR Design Limit
Δ T protection setpoint
Power capability analysis
ALARP
Commissioning and testing
Ageing and EMIT
Design Reference
Tier 2
· ALARP Demonstration Report of PCSR Chapter 5
Irradiation
(Reactor Core Design)
Golden Thread
Input Information
Output Information
Safety case document


General safety requirements of UK HPR1000
Thermal and Hydraulic Design
From CC Area
Main Design Technical Parameters
From Computer Code
Code Qualification Report (LINDEN)
Code V&V Report (LINDEN)
Thermal and hydraulic design
Tier 2
Thermal Hydraulic Design
NSSS Operating Parameters
For F&C Area (Fuel System Design)
Reactor operating parameters
Tier 3
Maximum Steady State
Uncertainties Analysis
From F&C Area (Fuel System Design)
CHF correlation
Geometrical parameters of fuel assembly
Pressure drop coefficients of fuel assembly
DNBR design basis
Tier 3
DNBR Design Limit
Tier 3
Overtemperature ΔT and
Overpower ΔT Protection
Setpoints Design
For FS/SAA Areas
DNBR design limit
For F&C Area (Nuclear Design)
DNBR design limit
From F&C Area (Fuel Design)
Melting temperature of fuel pellet
Fuel temperature design basis
For FS/SAA Areas
Overtemperature ΔT and overpower ΔT protection Setpoints
For F&C Area (Nuclear Design)
Overtemperature ΔT and overpower ΔT protection Setpoints
From F&C Area (Fuel Design)
Geometrical parameters of fuel assembly
Pressure drop coefficients of fuel assembly
Geometrical parameters of reactor structure
Pressure drop coefficients of reactor structure
For ME Area
Design information of the RCV [CVCS]
Core flow design basis
Tier 3
Pressure Drop and Bypass Flow
For ME Area
Core hydraulic characteristics
For F&C Area (Fuel Design)
Core hydraulic characteristics
From F&C Area (Fuel Design)
Geometrical parameters of fuel assembly
Pressure drop coefficients of fuel assembly
Geometrical parameters of reactor structure
Pressure drop coefficients of reactor structure
Hydrodynamic instability design basis
Tier 3
Core Hydrodynamic Instability Analysis
ALARP
Commissioning and testing
Ageing and EMIT
Design Reference
Tier 2
· ALARP Demonstration Report of PCSR Chapter 5
Irradiation
(Reactor Core Design)
Golden Thread
Input Information
Output Information
Safety case document