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Pre-Construction Safety Report
Chapter 5
Reactor Core

Text within this document that is enclosed within brackets '{…}' contains Sensitive Information and has been removed.

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5.1 List of Abbreviations and Acronyms

ABWR	Advanced Boiling Water Reactor
ACoP	Approved Code of Practice (UK)
AFCEN	French Association for Design, Construction and In-Service Inspection Rules for Nuclear Steam Supply System Components
Ag-In-Cd	Silver-Indium-Cadmium
ALARP	As Low As Reasonably Practicable
ANSI	American National Standards Institute
AP1000	Advanced Passive pressurised water reactor
ASME	American Society of Mechanical Engineers
BOC	Beginning Of Cycle
BCX	Beginning of Cycle, Equilibrium Xenon
CGN	China General Nuclear Power Corporation
CHF	Critical Heat Flux
CPR1000	Chinese Pressurised Reactor
CPR1000^{+}	Chinese Improved Pressurised Reactor
ACPR1000	Advanced Chinese Pressurised Reactor
CRDM	Control Rod Drive Mechanism
DBC	Design Basis Condition
DNB	Departure from Nucleate Boiling
DNBR	Departure from Nucleate Boiling Ratio
DR	Design Reference
EMIT	Examination, Maintenance, Inspection and Testing
EPR	European Pressurised Reactor
EOC	End Of Cycle
GDA	Generic Design Assessment
HPR1000	Hua-long Pressurised Reactor

HSE	Health and Safety Executive (UK)
IAEA	International Atomic Energy Agency
KRT	Plant Radiation Monitoring System [PRMS]
LCO	Limiting Condition of Operation
LOCA	Loss Of Coolant Accident
MOC	Middle Of Cycle
NFCC	Non-Fuel Core Component
ONR	Office for Nuclear Regulation (UK)
OPEX	Operating Experience
PCER	Pre-Construction Environmental Report
PCI	Pellet-Cladding Interaction
PCSR	Pre-Construction Safety Report
PMC	Fuel Handling and Storage System [FHSS]
PNS	Primary Neutron Source
PNSA	Primary Neutron Source Assembly
PWR	Pressurised Water Reactor
RCCA	Rod Cluster Control Assembly
RCP	Reactor Coolant System [RCS]
RCV	Chemical and Volume Control System [CVCS]
REN	Nuclear Sampling System [NSS]
RGL	Rod Position Indication and Rod Control System [RPICS]
RGP	Relevant Good Practice
RIC	In-core Instrumentation System [IIS]
RPE	Nuclear Island Vent and Drain System [VDS]
RPV	Reactor Pressure Vessel
RPN	Nuclear Instrumentation System [NIS]
SAP	Safety Assessment Principle (UK)
SCC	Stress Corrosion Cracking

SFIS	Spent Fuel Interim Storage
SFR	Safety Functional Requirement
SNS	Secondary Neutron Source
SNSA	Secondary Neutron Source Assembly
SSC	System, Structure and Components
TAG	Technical Assessment Guide (UK)
UK EPR	UK version of the European Pressurised Reactor
UK HPR1000	UK version of the Hua-long Pressurised Reactor
WENRA	Western European Nuclear Regulators Association

System codes (XXX) and system abbreviations (YYY) are provided for completeness in the format (XXX [YYY]), e.g. Chemical and Volume Control System (RCV [CVCS]).

5.2 Introduction

The Fuel & Core design is a combined concept including the design details on the fuel route, which is expected to satisfy the fundamental safety functions as follows:

a) Control of reactivity in the reactor and in the fuel storage facilities;

b) Removal of heat from the reactor and from the fuel storage facilities; and

c) Confinement of radioactive material, shielding against radiation and control of planned radioactive releases, as well as limitation of accidental radioactive releases.

The fuel route is divided into four sections, i.e. the handling & transport, the irradiation (reactor core), the storage and the Spent Fuel Interim Storage (SFIS). The purpose of this chapter is to introduce the reactor core design, which consists of the fuel system design, the nuclear design and the thermal and hydraulic design. Among these four sections in the fuel route, the design of the handling, transport and storage is presented in Pre-Construction Safety Report (PCSR) Chapter 28. The design of the SFIS is presented in PCSR Chapter 29. The design information of the reactor core is presented in this chapter.

The PCSR, Pre-Construction Environmental Report (PCER) and supporting references based on the STEP-12 fuel assembly, which represent the design of the UK version of the Hua-long Pressurised Reactor (UK HPR1000), have been submitted to the UK regulators. Following on from the change in fuel type to AFA 3G™ AA (Framatome) during GDA Step 3, the impact of the fuel change on the safety cases is assessed in Reference [1] *Fuel Change Impact Assessment*.

The present safety case of Reactor Core is produced based on the version 3 of the UK

HPR1000 Design Reference (DR3), as described in UK HPR1000 Design Reference Report, Reference [2].

5.2.1 Chapter Route Map

As mentioned in the previous section, PCSR Chapter 5 presents the reactor core design of the UK HPR1000, including the fuel system design, the nuclear design and the thermal and hydraulic design under Design Basis Conditions (defined in PCSR Chapter 4). In order to schematise the design logic of the reactor core design in this chapter, the chapter route map in form of Claim-Argument-Evidence is extracted from the overall route map and presented in the form of text instead of table for typeset reason.

Claim 3: *The design and intended construction and operation of the UK HPR1000 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 as low as reasonably practicable (ALARP).*

Claim 3.3: *The design of the processes and systems has been substantiated and the safety aspects of operation and management have been substantiated.*

Claim 3.3.1: *The design of the Fuel System and Reactor Core has been substantiated.*

To support Claim 3.3.1, this chapter presents five Sub-claims and a number of relevant arguments and evidences:

a) ***Sub-Claim 3.3.1.SC05.1:*** *The safety functional requirements (SFRs) or design bases have been derived for the reactor core design.*

1) ***Argument 3.3.1.SC05.1-A1****: The reactor core design bases have been derived from the safety analysis in accordance with the general design and safety principles (see Sub-chapter 5.4/5.5/5.6).*

- ***Evidence 3.3.1.SC05.1-A1-E1****: The criteria in fuel system design, including the fuel rod, the fuel assembly and the Rod Cluster Control Assembly (RCCA), are identified from the general safety function requirements (see Sub-chapter 5.4).*

- ***Evidence 3.3.1.SC05.1-A1-E2****: The design bases in nuclear design are identified from the general safety function requirements (see Sub-chapter 5.5).*

- ***Evidence 3.3.1.SC05.1-A1-E3****: The Departure from Nucleate Boiling Ratio (DNBR) design basis, the fuel temperature design basis, the core flow design basis and the hydrodynamic instability design basis in the thermal and hydraulic design are derived from the general safety functions (see Sub-chapter 5.6).*

2) ***Argument 3.3.1.SC05.1-A2****: The reactor core specific design principles are* *identified based on the relevant good practice (RGP) (see Sub-chapter 5.7).*

- ***Evidence 3.3.1.SC05.1-A2-E1**: The specific design principles of the fuel system design, the nuclear design and the thermal and hydraulic design are identified and implemented based on RGP* (see Reference [3] *ALARP Demonstration Report of PCSR Chapter 05*).

b) ***Sub-Claim 3.3.1.SC05.2:** The reactor core design satisfies the SFRs or design bases.*

1) ***Argument 3.3.1.SC05.2-A1**: Appropriate design methodologies including design codes and standards have been identified for the reactor core design.*

- ***Evidence 3.3.1.SC05.2-A1-E1**: According to design requirements and strategy of selection, appropriate design codes and standards of the fuel system design, the nuclear design and the thermal and hydraulic design have been identified (see Sub-chapter 5.3).*

2) ***SC05SC05Argument 3.3.1.SC05.2-A2**: The reactor core design has been analysed using the appropriate design methodologies to meet the relevant design requirements (see Sub-chapter 5.4.3/Sub-chapter 5.5.3/Sub-chapter 5.6.3).*

- ***Evidence 3.3.1.SC05.2-A2-E1**: The evaluations of the fuel rod design, the fuel assembly design and the RCCA design demonstrate that the design requirements are fulfilled so as to support relevant Safety Functions (see Sub-chapter 5.4).*

- ***Evidence 3.3.1.SC05.2-A2-E2**: The nuclear design evaluations are performed using the appropriate design methodology and the relevant design bases are satisfied. (see Sub-chapter 5.5).*

- ***Evidence 3.3.1.SC05.2-A2-E3**: The thermal and hydraulic design evaluations demonstrate that requirements of the DNBR design basis, the fuel temperature design basis, the core flow design basis and the hydrodynamic instability design basis are fulfilled. (see Sub-chapter 5.6).*

3) ***Argument 3.3.1.SC05.2-A3**: The reactor core design recognises interface requirements and effects from/to interfacing systems (see Sub-chapter 5.2.3/Sub-chapter 5.4.3/Sub-chapter 5.5.3/Sub-chapter 5.6.3).*

- ***Evidence 3.3.1.SC05.2-A3-E1**: The reactor core design has recognised interface requirements and effects from/to interfacing systems. (see Sub-chapter 5.2.2.3).*

c) ***Sub-Claim 3.3.1.SC05.3:** All reasonably practicable measures have been adopted to improve the design.*

1) ***Argument 3.3.1.SC05.3-A1****: The reactor core design meets the requirements of the relevant design principles (generic and system specific) and therefore of relevant good practice (see Sub-chapter 5.7).*

   - ***Evidence 3.3.1.SC5.3-A1-E1****: The main technical points of the fuel and core design for the UK HPR1000 are compared with the RGP and the current design are in compliance with the existing RGP* (see Reference [3] *ALARP Demonstration Report of PCSR Chapter 05*).

2) ***Argument 3.3.1.SC05.3-A2****: Design improvements have been considered and any reasonably practicable changes implemented (see Sub-chapter 5.7).*

   - ***Evidence 3.3.1.SC05.3-A2-E1****: The design improvements for reactor core design are identified and the reasonably practicable changes are implemented* (see Reference [3] *ALARP Demonstration Report of PCSR Chapter 05).*

d) ***Sub-Claim 3.3.1.SC05.4:*** *The reactor core performance will be validated by commissioning and testing.*

1) ***Argument 3.3.1.SC05.4-A1****: The reactor core has been designed to take benefit from a suite of pre-construction tests, to provide assurance of the initial quality of the manufacture (see Sub-chapter 5.8).*

   - ***Evidence 3.3.1.SC05.4-A1-E1****: The fuel assembly test is performed to ensure the mechanical performance.*

2) ***Argument 3.3.1.SC05.4-A2****: The reactor core has been designed to take benefit from a suite of commissioning tests, to provide assurance of the initial quality (see Sub-chapter 5.8).*

   - ***Evidence 3.3.1.SC05.4-A2-E1****: The core physics test is designed to ensure that the reactor is safe and operated in accordance with design.*

   - ***Evidence 3.3.1.SC05.4-A2-E2****: The test prior to initial criticality is designed to verify that proper coolant flow rates have been adopted in the core thermal and hydraulic analysis.*

   - ***Evidence 3.3.1.SC05.4-A2-E3****: The initial power and plant operation is designed to confirm that the peaking powers selected for use in the core thermal and hydraulic analysis are conservative.*

   - ***Evidence 3.3.1.SC05.4-A2-E4****: Component and fuel inspection is implemented to verify the uncertainty in the engineering hot channel factor in the design analyses is conservative.*

e) ***Sub-Claim 3.3.1.SC05.5:*** *The effects of ageing of the reactor core have been addressed in the design and suitable examination, maintenance, inspection, and* *testing are specified.*

1) ***Argument 3.3.1.SC05.5-A1****: An initial examination, maintenance, inspection and testing (EMIT) strategy has been developed for fuel system, identifying components that are expected to be examined, maintained, inspected and tested (see Sub-chapter 5.9).*

 - ***Evidence 3.3.1.SC05.5-A1-E1:*** *The Nuclear Sampling System (REN [NSS]) is applied to confirm that the radioactivity of primary coolant is maintained below the limit (see Sub-chapter 5.9).*

 - ***Evidence 3.3.1.SC05.5-A1-E2:*** *During the fuel unloading, visual inspection and online sipping tests (in case of the abnormal radioactivity levels) will be performed.*

In this chapter, the fuel and core design is justified by adopting appropriate methods to support the claims above and it is demonstrated that the risk of fuel failure due to the fuel and core design remains ALARP.

5.2.2 Chapter Structure

The structure of Chapter 5 is shown as follows.

- Sub-chapter 5.1 List of Abbreviations and Acronyms

This sub-chapter lists the abbreviations and acronyms that are used in this chapter.

- Sub-chapter 5.2 Introduction

This sub-chapter gives the route map, structure and interfaces with other chapters.

- Sub-chapter 5.3 Applicable Codes and Standards

This sub-chapter introduces the codes and standards applied in fuel system design, nuclear design and thermal-hydraulic design.

- Sub-chapter 5.4 Fuel System Design

This sub-chapter provides SFRs, design descriptions on fuel system design.

- Sub-chapter 5.5 Nuclear Design

This sub-chapter provides SFRs, design descriptions and design evaluations on nuclear design.

- Sub-chapter 5.6 Thermal and Hydraulic Design

This sub-chapter provides SFRs, design description and design evaluation on thermal and hydraulic design.

- Sub-chapter 5.7 ALARP Assessment

This sub-chapter presents the ALARP demonstration for PCSR Chapter 5.

- Sub-chapter 5.8 Commissioning and Testing

This sub-chapter lists the commissioning and testing activities related to fuel and core design.

- Sub-chapter 5.9 Ageing and EMIT

This sub-chapter introduces the EMIT activities related to fuel and core design.

- Sub-chapter 5.10 Failed Fuel Management Strategy

This sub-chapter introduces the design and measures implemented in the failed fuel management strategy.

- Sub-chapter 5.11 Source Term

This sub-chapter presents the source term related to fuel and core design.

- Sub-chapter 5.12 Concluding Remarks

This sub-chapter gives the concluding remarks for this chapter.

- Sub-chapter 5.13 References

This sub-chapter lists the supporting references of this chapter.

- Appendix 5A The Computer Codes Description

This appendix introduces the computer codes used in PCSR Chapter 5.

5.2.3 Interfaces with Other Chapters

The interfaces with other PCSR chapters are listed in the following table.

T-5.2-1 Interfaces between Chapter 5 and Other Chapters

PCSR Chapter	Interface
Chapter 1 Introduction	Chapter 1 provides the Fundamental Objective, Level 1 Claims and Level 2 Claims. Chapter 5 provides chapter claims and arguments to support the high level claims presented in Chapter 1.
Chapter 2 General Plant Description	Chapter 2 provides a brief introduction to the fuel and core. Chapter 5 provides a further description of the

PCSR Chapter	Interface
	reactor core mentioned in Sub-chapter 2.5.
Chapter 4 General Safety and Design Principles	Sub-chapter 4.4.3.2 provides the definition of Design Basis Conditions (DBCs) and safety functions related to Chapter 5.
Chapter 6 Reactor Coolant System	Chapter 6 provides the information of control rod drive mechanism and reflector. Chapter 5 provides the fuel and core design.
Chapter 8 Instrumentation and Control	Chapter 5 provides the functional requirements of Nuclear Instrumentation System (RPN [NIS]), In-core Instrumentation System (RIC [IIS]) and Rod Position Indication and Rod Control System (RGL [RPICS]). The SCCA design is designed to allow the insertion of instrumentation rod.
Chapter 10 Auxiliary Systems	Chapter 10 provides detailed design information of the RCV [CVCS].
Chapter 12 Design Basis Condition Analysis	Chapter 5 provides the acceptance criteria and performance data related to core and fuel under accidents for fault studies use. Chapter 12 provides the case-by-case analysis on the acceptance criteria for DBC-2, DBC-3 and DBC-4.
Chapter 13 Severe Accident Analysis	Chapter 5 provide the generic nuclear data for DEC-A and severe accident analysis. Chapter 13 provides the analysis of severe accident analysis.
Chapter 17 Structural Integrity	Chapter 5 Reactor Core describes fuel system design, nuclear design and thermal and hydraulic design. The relevant descriptions of irradiation surveillance requirements for the Reactor Pressure

PCSR Chapter	Interface
	Vessel (RPV) core shell and its radiation damage mechanism are discussed in Chapter 17. The dynamic motion of upper and lower core plates in LOCA condition shall be delivered and applied to the fuel design models.
Chapter 18 External Hazards	Chapter 18 provides list of external hazards, relevant design principles, design basis and safety assessment to identify potential risk information, and the ALARP demonstration from the external hazards point of view. Chapter 5 provides fuel system design applying external hazard protection design principles, which is used for external hazards safety assessment.
Chapter 21 Reactor Chemistry	Chapter 5 provides design requirements of the fuel and core, and the concentration of boron with fuel burnup and the thermal-hydraulic information for the fuel deposits. Chapter 21 provides the information about Reactor Chemistry regime which is related to fuel cladding corrosion.
Chapter 22 Radiological Protection	Chapter 5 provides reactor core design information used in source term design. Chapter 22 provides the generic aspects of source term and covers the various source terms for normal operation.
Chapter 23 Radioactive Waste Management	Chapter 5 provides the design of reactor core which contributes to minimise radioactive waste at source and generates unavoidable radioactive waste. Chapter 23 provides the management of radioactive waste generated from reactor core.
Chapter 28 Fuel Route and	Chapter 5 covers the design of Fuel Assembly

PCSR Chapter	Interface
Storage	(FA) and Non-Fuel Core Components (NFCCs), which is a key input to the design of the Fuel Handling and Storage System (PMC [FHSS]) Systems, Structures and Components (SSCs). The reactor power operation period is part of the overall fuel route described in Chapter 28. Chapter 28 provides a general introduction of fuel route and the safety demonstration of fuel handling and storage system.
Chapter 29 Interim Storage for Spent Fuel	PCSR Chapter 5 covers the fuel assembly design parameters and operation information, including size, weight, quantity, etc., which is the necessary information to spent fuel disposability assessment and BQF design. Chapter 29 provides the introduction of spent fuel interim storage, including the spent fuel management strategy, general requirements, optioneering considerations, etc.
Chapter 30 Commissioning	Chapter 30 provides the arrangements and requirements for commissioning aligned with SSC design requirements, which is associated with Sub-chapter 5.8 Commissioning and Testing.
Chapter 31 Operational Management	Chapter 5 provides the fuel and core design which the operating limits and conditions prescribed in Chapter 31 derived from. Chapter 31 presents the arrangement of EMIT, operating limits and conditions for core design.
Chapter 33 ALARP Evaluation	The ALARP approach presented in Chapter 33 has been applied in Chapters 5 to perform the ALARP demonstration for the structure, system and component designs, which supports the overall ALARP demonstration addressed in Chapter 33.

5.3 Applicable Codes and Standards

The principles for selection of applicable design codes and standards for reactor core design consider the design characteristics, the UK regulatory expectations, the requirements of guidance documents and the engineering practice (see Chapter 4.4.7 Codes and Standards).

The following principles are applied during the selection process:

a) Adopt international good practice or RGP accepted by UK Regulatory authorities;

b) Adopt the latest version of codes and standards. Gap analysis is carried out for the selection of an older version when the latest version is available;

c) Priority is given to codes and standards specific to the nuclear industry to ensure a balance between conservative design and security is achieved;

d) The codes and standards have been applied to other reactor types from previous GDAs.

According to the design requirements and strategy of selection, the codes and standards listed below are applied to the UK HPR1000 reactor core design.

a) The analysis of codes and standards for the fuel system design is based on the function, structure and material characteristics of the fuel components. The following list used for the fuel system design is taken from *Suitability Analysis of Codes and Standards in Fuel Design* (see Reference [4]).

   [1] IAEA, Safety Standards: Design of the Reactor Core for Nuclear Power Plants Safety Guide, No. NS-G-1.12, 2005 edition.

   [2] IAEA, Specific Safety Requirements - Safety of Nuclear Power Plants: Design Specific Safety Requirements, No. SSR-2/1, 2016 edition.

   [3] AFCEN, Design and Construction rules for Fuel Assemblies of PWR Nuclear Power Plants, RCC-C, 2018.

   [4] AFCEN, Design and Construction Rules for Mechanical Components of PWR Nuclear Islands, RCC-M, 2017.

   [5] ASME, ASME's Boiler and Pressure Vessel Code (BPVC) Section III - Rules for Construction of Nuclear Facility Components - Division 1 - Subsection NB - Class 1 Components, BPVC-III NB, 2019.

   [6] ASME, ASME's Boiler and Pressure Vessel Code (BPVC) Section III - Rules for Construction of Nuclear Facility Components - Division 1 - Subsection NG - Core Support Structures, BPVC-III NG, 2019.

   [7] US NRC, Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition – Reactor, NUREG-0800, Chapter 4 Reactor, Section 4.2 Fuel System Design Review Responsibilities Rev. 3 (Formerly issued as NUREG-75/087).

   [8] ANSI, Light Water Reactors Fuel Assembly Mechanical Design and Evaluation,

ANSI 57.5, 1996 (R2006).

b) The nuclear design principles are analysed in accordance with the requirements identified from related codes and standards, which provide clarifications of the definitions of technical glossaries, nuclear design bases and the methods, conditions and acceptance criteria for reactor core physics tests. The following list of codes and standards used for the nuclear design is taken from the *Suitability Analysis of Codes and Standards in Fuel and Core Design* (see Reference [5]):

[1] IAEA, Design of the Reactor Core for Nuclear Power Plants, No. NS-G-1.12, 2005 edition.

[2] IAEA, Safety of Nuclear Power Plants: Design, No.SSR-2/1, 2016 edition.

c) The codes and standards for the thermal and hydraulic design are applied in accordance with general technical principles, definitions of related glossaries, thermal design bases, hydraulic design bases, determination principles of the design limits, the pressure drop and the hydraulic load. The following list of codes and standards used for the nuclear design is taken from *Suitability Analysis of Codes and Standards in Fuel and Core Design* (see Reference [5]):

[1] IAEA, Safety of Nuclear Power Plants: Design, No.SSR-2/1, 2016 edition.

5.4 Fuel System Design

This sub-chapter describes the SFRs that shall be fulfilled in the fuel system design. The fuel rod design covers DBC-1, DBC-2 and frequent DBC-3 (DBC-3 with the frequency between 10^{-2} /r y and 10^{-3} /r y), while the discussions on acceptance criteria in infrequent DBC-3 and DBC-4 refer to Chapter 12. The fuel assembly design, RCCA design and SCCA mechanical design covers all DBCs.

The AFA 3G^{TM}AA fuel assembly is adopted in the UK HPR1000.

5.4.1 Safety Functional Requirement

The fuel system, including the fuel rod, the fuel assembly, the RCCA and the SCCA, shall be properly designed to fulfil the safety functions provided in Chapter 4.

For DBC-1, DBC-2 and frequent DBC-3, the following SFRs are identified:

a) The nuclear design, thermal-hydraulic design and fuel system design ensure that the heat released in the fuel can be removed by the reactor coolant (Safety Function H2 - Remove heat from the core to the reactor coolant);

b) The nuclear design and fuel system design ensure the control of core reactivity, the nuclear chain reaction could be stopped, and the reactor would be able to return to a safe state using two diverse shutdown systems (Safety Functions R1 - Maintain core reactivity control, R2 - Shutdown and maintain core sub-criticality and R3 - Prevention of uncontrolled positive reactivity insertion into the core);

c) The design and performance of the fuel system shall preclude the release of radioactive material during operation under DBC-1, DBC-2 and frequent DBC-3 by maintaining the integrity of fuel cladding (Safety Function C1 - Maintain integrity of the fuel cladding to ensure confinement of radioactive material).

During start-up and shutdown, the SFRs identified above remain applicable. The justification of these SFRs shall take into account the maximum power changes which the fuel assembly and RCCA experience.

Fuel failure (defined as penetration of the fuel rod cladding which is the fission product barrier) is not expected during DBC-1, DBC-2 and frequent DBC-3.

For infrequent DBC-3 and DBC-4, the following SFRs are identified:

a) Fuel system design ensures the preservation of an assembly array geometry to enable the insertion of RCCAs to shut down the reactor (Safety Functions R1, R2 and R3);

b) Fuel system design ensures the preservation of an assembly array geometry to enable the cooling of the reactor core (Safety Function H2).

The risk of fuel failure is considered to be ALARP in infrequent DBC-3 and DBC-4. The related ALARP assessment refers to Sub-Chapter 12.15 of Chapter 12.

5.4.2 Design Description

The fuel system includes the fuel assembly, RCCA and SCCA, a detailed description of the fuel assembly is given in Reference [6] (*AFA 3G^{TM}AA Fuel Assembly Description for HPR1000 Reactor*), the RCCA description is provided in Reference [6] (*HARMONI RCCA - Description, Functional Requirements and Material Properties*), and the structural features of the SCCA are demonstrated in Reference [8] (*SCCA – Description, Functional Requirements and Materials Properties*).

5.4.2.1 Fuel Assembly

The assembly is made up of 264 fuel rods supported by an orthogonal structure with a 17×17 square array (F-5.4-1).

The skeleton consists of:

- 1 top nozzle,

- 1 bottom nozzle,

- 24 guide thimbles,

- 1 instrumentation tube,

- 8 structural grids (6 of them being mixing grids),

- 3 mid-span mixing grids.

The instrumentation tube is located in the centre and provides a channel for insertion of an in-core neutron detector.

The guide thimbles provide channels for insertion of different types of core components whose type depends on the position of the particular fuel assembly in the core.

The fuel rods are loaded into the skeleton to form the fuel assembly, in such a way that there is an axial clearance between the fuel rod ends and the top and bottom nozzles in order to accommodate the differential elongation of the skeleton and the fuel rods during operation.

5.4.2.1.1 Fuel Rod

The first cycle of the UK HPR1000 reactor is made up of six types of fuel assembly which differ in the UO_2 enrichment and the number of gadolinium rods. The fuel management in the subsequent cycles are made up of three types of fuel assembly which differ in the number of gadolinium rods.

The UO_2 rods are filled with cylindrical uranium dioxide pellets with chamfered edges, fabricated by cold pressing then sintering. The dishes are machined into each pellet at the upper and lower faces to reduce the axial expansion of the fuel stack.

A plenum is provided at the top end of the fuel rod to accommodate fission gas release. A stainless steel helical spring holds the pellet column in place during transportation operations preceding loading into the reactor and during handling operations.

The pellet-cladding gap and the plenum volume are designed to take into account the release of fission gases, differential thermal expansion between cladding and pellet and the swelling of the pellets.

The rod is helium-pressurized, which improves the conductivity of the pellet-cladding gap and enables fuel temperature to be kept down and fission gas release to be restricted.

The rod end plugs were designed for better insertion of the fuel rods in all on-site repair situations. The cladding and the end plugs are joined together by the USW (Upset Shape Welding) process. The end plugs are made of Zirconium alloy (Zircaloy-4 or $M5_{Framatome}$).

The UO_2-Gd_2O_3 fuel rod only differs from the UO_2 rod in the composition of pellets.

5.4.2.1.2 Top Nozzle and Hold-down System

The top nozzle assembly functions as the upper structural element of the fuel assembly, the coolant outlet plenum, and a partial protective housing for the rod cluster control assembly (RCCA) or other core components.

It consists of a welded square structure (made of AISI 304 L) comprising an adaptor plate and a top plate interconnected by a thin enclosure and 4 multi-leaf springs (made of alloy 718) packs held in place by 4 attachment screws and protected by sockets machined in the top plate.

The adaptor plate is provided with slots for coolant flow. The choice of a 1/8 symmetrical array and of triangular and oblong slots provides an increase in flow area while reducing the thickness of the adaptor plate.

The centre of the adaptor plate presents a hole to accommodate instrumentation tube, which provides a channel for the passage of the in-core detector.

The adaptor plate also features machined holes for connecting the nozzle to the guide thimbles and providing a channel for the core component rods. It distributes the transmitted loads to the guide thimbles and limits any axial shifting of the fuel rods.

The top nozzle skirt is a thin-walled enclosure; it forms the coolant divergence zone and connects the adaptor plate to the top plate.

The top plate has a large square opening in the centre to permit access for the RCCA spider assemblies, holddown systems and tools for handling the assembly in the shop or on site. This opening also permits access to all the connections between the guide thimbles and the adaptor plate. It channels the coolant flow through the upper core plate towards the upper internals. Two pads located on two diagonally opposite corners of the top plate accommodate the alignment pins on the upper core plate and provide lateral positioning of the fuel assembly.

Holes are machined into the other two pads to accommodate and secure the four spring packs. They protect the spring leaf ends and attachment screws during handling operations.

The holddown spring screws are made captive by lock wires welded to the pads. The free end of the upper leaf is bent back towards the bottom. It passes through the bottom leaves. Its « key » shape allows it to lock into a special-purpose slot in the top plate. These arrangements ensure that in the very unlikely case of failure of these springs in their stressed area, the failed leaf remains captive in the upper nozzle and does not risk disrupting the motion of the RCCAs in the various operating conditions.

The springs exert sufficient force to counteract the hydraulic upflow forces. In normal flow conditions, the assembly is kept in contact with the lower core plate (axial holddown of the assembly). This system also absorbs the differential elongation between assembly and internals during changes of temperature and under irradiation.

5.4.2.1.3 Bottom Nozzle

The anti-debris bottom nozzle ensures the distribution of the coolant through the fuel assembly, supports the vertical loads imposed to the structure, limits downward fuel rod movement and ensures fuel assembly protection against debris.

It is made up of a ribbed structure with 4 feet topped with a thick anti-debris device (made of AISI 660). The legs form a plenum for the inlet coolant flow towards the fuel assembly.

The ribbed structure (made of AISI 304 L) is designed to accommodate the loads transmitted by the guide thimbles. It acts as a housing for the guide-thimble attachment screws. It supports the anti-debris device and provides an outer enclosure compatible with handling requirements.

Indexing and positioning of the fuel assembly is controlled by alignment holes in two diagonally opposite nozzle legs which mate with the locating pins in the lower core plate.

The guide thimbles are firmly attached to the ribbed plate by socket head screws.

The 3 mm-thick anti-debris plate features 3.3×3.3 mm square cutouts and 0.45 mm wide ligaments.

Two pins, made captive by a spot weld, secure the anti-debris plate to the ribbed structure during installation and removal sequences. The anti-debris plate is also attached by the 24 guide-thimble lower connections. The upper face of the anti-debris plate has tapered recesses for centering the guide-thimble end plugs during nozzle repositioning.

Chamfers on the outer edges of the nozzle facilitate the insertion of the assemblies into the reactor during loading operations.

5.4.2.1.4 Grid

The grids ensure that the fuel rods are regularly spaced relatively to each other throughout fuel assembly lifetime. The grids are of type AFA 3G™AA and are divided into 2 categories:

- 8 structural grids,
- 3 mid-span mixing grids.

The structural grids are of two types:

- The bottom and top end grids have no mixing vanes,
- The 6 mixing grids feature mixing vanes in the upper part, designed to improve coolant mixing.

They consist of recrystallized $M5_{Framatome}$ straps to which hairpin springs are fitted, made of quenched and aged alloy 718.

The inner and outer straps are assembled to form an array of 289 cells, 25 of which receive the guide thimbles and instrumentation tube. The 264 remaining cells receive the fuel rods. Within a given cell, each rod is held in place by a double system of springs and dimples which act in 2 perpendicular planes. The dimples are obtained by forming in the straps. The alloy 718 springs are hairpin-shaped.

In order to still enhance its thermal-hydraulic performance, the AFA 3G^{TM}AA fuel assembly features 3 mid span mixing grids (so-called MSMG), located mid-way along the three highest heated spans of the assembly. The MSMGs have a coolant mixing function only. They are made of straps stamped and formed from recrystallized $M5_{Framatome}$ alloy strips.

5.4.2.1.5 Guide Thimble

The guide thimbles of the AFA 3G^{TM}AA assembly are of the MONOBLOC type. The guide thimbles are structural members which also provide channels for the neutron absorber rods or neutron source assemblies. The guide thimble is one-piece of $M5_{Framatome}$ alloy.

The inner diameter of the upper part of the guide thimble provides an annular area sufficiently large to permit rapid insertion of the control rod during a scram and to accommodate the flow of coolant during normal operation. The inner diameter of the guide thimble is reduced in its lower part. It acts as a dashpot to slow down the motion of the control rod at its travel limit.

The outer diameter remains constant throughout the tube.

The guide thimble features flow holes located above the dashpot to enable fluid flow during normal operation and to accommodate the outflow of water during the rapid insertion of the control rod.

A plug is welded to the bottom end of the guide thimble and drilled with a threaded hole for connection to the bottom nozzle. A threaded sleeve is swaged to the top of the guide-thimble and is used to fasten it to the top nozzle.

5.4.2.1.6 Instrumentation Tube

The instrumentation tube of each fuel assembly is used as a channel for in-core neutron detectors. It is also made of $M5_{Framatome}$ alloy. This tube exhibits a constant thickness and inner diameter throughout its length which are equal to those of the current part of the guide thimble. The instrumentation tube is attached to the grids in the same way as the guide, however it is only constrained at the top and bottom nozzle locations.

5.4.2.2 Rod Cluster Control Assembly

There are two types of RCCA for UK HPR1000 reactor:

- black RCCA with 24 absorber rods filled with Ag-In-Cd,
- grey RCCA with:
  - 8 absorber rods identical to the black RCCA absorber rod,
  - 16 stainless steel rods which are filled with stainless steel spacers (also called inert rods).

Figure 1 provides the main characteristics of HARMONI RCCA.

Each RCCA is composed of:

- a supporting structure in the form of a spider assembly coupled to a drive shaft which is actuated by a control rod drive mechanism (CRDM) mounted on the reactor vessel head,
- 24 rods (absorber or stainless steel rods).

5.4.2.3 Stationary Core Component Assembly (SCCA)

The SCCAs are designed to be irradiated in the UK HPR1000 reactor, there are 3 types of core components, including:

- TPA: Thimble Plug Assembly
- PNSA: Primary Neutron Source Assembly
- SNSA: Secondary Neutron Source Assembly

5.4.2.3.1 TPA

Each TPA consists of:

a) A supporting structure in the form of a hold-down assembly, which rests on the top nozzle adaptor plate

   The hold-down assembly is composed of:

   1) The base plate perforated for the passage of the primary coolant. Holes are drilled in the plate to accommodate thimble plugs. This plate rests on the top nozzle adapter plate, leaving space for water to flow between the two plates;

   2) The spring guide welded to the base plate which allows the passage of the instrumentation;

   3) Two helical coil springs (inner and outer springs);

   4) The yoke, held in position and guided by two pins which ride in slots in the spring guide. The yoke fits around the spring guide and compresses the hold-down springs during normal operation.

b) A bundle of 24 rods called thimble plugs who are securely fastened to a base-plate by a nut which is then locked in place by a welded pin.

   The thimble plug is a short solid rod used to fill the end of vacant assembly guide thimbles, in order to limit the fuel assembly by-pass flow.

5.4.2.3.2 PNSA and SNSA

Each PNSA consists of:

a) a hold-down assembly identical to TPA;

b) 22 thimble plugs identical to TPA;

c) one PNS rod (Primary Neutron Source rod);

d) one SNS rod (Secondary Neutron Source rod).

Each SNSA consists of:

a) a hold-down assembly identical to TPA (see §2.1);

b) 20 thimble plugs identical to TPA;

c) 4 SNS rod (Secondary Neutron Source rod).

PNS rod and SNS rod consist in a cladding tube closed at its extremities by two welded plugs. The bottom of the thimble plugs and the lower end plug of the long rods are bullet-nosed to facilitate the rods insertion into the guide thimbles of the fuel assembly.

5.4.3 Design Evaluation

As indicated in Sub-chapter 5.4.1, the fuel system is designed to satisfy the SFRs identified in Sub-chapter 4.4.

The bounding analysis for fuel rod performance is performed in DBC-1, DBC-2 and frequent DBC-3 (see Reference [9] to [11]). Detailed discussion on acceptance criteria in DBC-2, DBC-3 and DBC-4 are performed case-by-case in Sub-chapter 12.5.1 of Chapter 12.

The analysis for fuel assembly, RCCA and SCCA performance is performed in all DBCs (see Reference [12] to [14]).

5.4.3.1 Fuel Rod

The design assessment for the fuel rod addresses the following potential physical phenomena:

a) Irradiation densification and swelling;

b) Fuel temperature;

c) Fission gas release;

d) Irradiation creep and growth;

e) Pellet-Cladding Interaction (PCI)-Stress Corrosion Cracking (SCC);

f) Creep collapse;

g) Strains and stresses;

h) Fatigue;

i) Oxidation and hydriding; and

j) Vibration and fretting wear.

Based on the physical phenomena shown above, the design criteria are applied to preclude fuel failure during operation in DBC-1, DBC-2 and frequent DBC-3. It should be noted that the vibration and fretting wear is analysed in Sub-chapter 5.4.3.2.3 while the irradiation growth of fuel rods is analysed in Sub-chapter 5.4.3.2.4. The fuel rod performance code COPERNIC is applied in the fuel rod design evaluation. COPERNIC is a best-estimate code that predicts the thermal-mechanical behavior of a single fuel rod. More information about COPERNIC can be seen in the Appendix 5A. All the following points in fuel rod design evaluation are emphasised as demonstrating that the design requirements are fulfilled for the fuel rods so as to support Safety Functions H2 and C1 listed in Sub-chapter 5.4.1.

5.4.3.2.1 Fuel Temperature (Safety Functions H2 and C1)

The maximum pellet temperature shall remain lower than the fuel melting point. The aim of this criterion is to prevent fuel melt conditions, which could cause volume variation due to phase change (and dispersion of fuel particles), resulting in severe duty on the cladding. Taking into account the variation of fuel melting point with burn-up, the analysis in steady-state and frequent fault conditions shows that an adequate margin exists between the maximum fuel temperature obtained and the melting point calculated (see Reference [9]).

5.4.3.2.2 Fuel Rod Internal Pressure (Safety Function C1)

During DBC-1, the internal pressure due to the fission gas release and initial pressurisation shall be less than the value which would lead to an increase or a re-opening of the pellet to cladding diametric gap by cladding tensile creep. The criterion precludes the outward cladding creep rate from exceeding the fuel swelling rate, and therefore, ensures that the gap does not re-open during steady state operation. Considering the uncertainties which are linked to the models or the manufacturing parameters and the penalty of the operating transients, the maximum internal pressure in DBC-1 maintains lower than the limiting pressure (see Reference [9]).

5.4.3.2.3 Cladding Stress (Safety Function C1)

In steady-state conditions, only low stresses are generated in the cladding. Before the pellet comes into contact with the cladding, the maximum absolute stress level (due to the pressure difference between the inside and outside the rod) is low. After the contact, tensile stresses develop in the cladding due to fuel swelling. Even so, the cladding stress remains small (see Reference [9]).

During an overpower transient, the combined effects of fuel pellet expansion and the presence of corrosive fission product in the gap, such as iodine, could lead to the PCI-SCC of cladding. The risk of fuel failure induced by PCI-SCC is estimated in frequent fault and the results show that the cladding remains at an adequate PCI margin based on the technical limit of $M5_{Framatome}$ alloy (see References [10] and [11]).

5.4.3.2.4 Cladding Strain (Safety Function C1)

Following the closure of the pellet to cladding diametric gap, fuel swelling and thermal expansion and the interaction between pellet and cladding induce cladding strains along circumferential direction. Criteria on the uniform circumferential strain are defined in order to avoid strain type fuel failure. The analysis shows that the maximum cladding permanent strain is estimated to be negative and verified below the limit of 1% in steady-state condition, the maximum total strain induced by frequent fault with uncertainties remains below the limit of 2% in frequent fault (see Reference [9]).

5.4.3.2.5 Cladding Corrosion and Hydriding (Safety Functions H2 and C1)

Oxidation and hydriding are directly related to fuel performance during operation. Oxidation degrades the cladding thermal conductivity while the hydriding leads to a decrease in the cladding ductility and impact toughness. According to the results of upper-bound analysis, the maximum cladding corrosion thickness is observed at the end of life with the upper-bound value is confirmed to be lower than the required limit of 100 μm (see Reference [9]).

5.4.3.2.6 Cladding Stability (Safety Function C1)

At the beginning of life, the maximum coolant pressure shall neither lead to the collapse nor to the plastic strain of the cladding. The freestanding criterion is validated at the beginning of life for a rod in hot-zero-power reactor. The results show that the criterion is complied with (see Reference [9]).
At the level of axial gaps, which could be formed within the fuel column, the combined effects of the differential pressure across the cladding wall and of the cladding creep, could lead to cladding collapse (the increase of ovality results in the cladding circumferential buckling). The fuel rod design, in particular the fuel rod pressurisation and by using of stable fuel during irradiation, avoids any risk of cladding collapse (see Reference [9]).

5.4.3.2.7 Fuel Column Stability (Safety Function C1)

The spring located in the rod plenum must prevent any fuel stack shifting in the cladding when the rod is subjected to a maximum acceleration of 4g during shipping operations before irradiation. The minimal compression exerted by the spring on the pellet stack is estimated to be 4 times greater than the weight of the fuel column to guarantee that the criterion is met (see Reference [9]).

5.4.3.2.8 Cladding Fatigue (Safety Function C1)

The reactor operating conditions lead to alternate loadings, which eventually impose cladding stress cycling and fatigue. For the base-load mode of operation, it is not necessary to verify the fatigue criterion since the number of power and temperature cycles is very limited. For the daily load follow, the stress levels that could be reached in the grid following transients are not sufficient to cause significant cladding fatigue (see Reference [9]).

5.4.3.2.9 Fuel Rod Design Summary

As shown above, all the fuel rod design criteria are met with margins taking into account the operating conditions. The design of the fuel rod therefore maintains its structural integrity and its capability to transfer heat into the coolant (Safety Functions H2 and C1) in DBC-1, DBC-2 and frequent DBC-3. Detailed discussion on acceptance criteria in DBC-2, DBC-3 and DBC-4 are performed case-by-case in Sub-chapter 12.5.1 of Chapter 12.

More information on control of corrosion is provided in Chapter 21, as setting and maintaining an appropriate Reactor Chemistry regime is vital to obtaining good fuel performance.

5.4.3.2 Fuel Assembly

The mechanical integrity of a fuel assembly is evaluated to withstand the mechanical stresses as a result of:

a) Fuel handling and loading;

b) Power variations;

c) Temperature gradients;

d) Hydraulic loads, induced by the core flow and hold-down forces required to maintain core geometry;

e) Irradiation (e.g. radiation induced growth and swelling);

f) Vibration and fretting induced by coolant flow;

g) Creep deformation;

h) External events such as earthquakes; and

i) Postulated faults such as a loss of coolant accident (LOCA).

Considering all the mechanical stress caused by the phenomenon shown above, each component was evaluated by design loads, which is defined as the most conservative load in each DBC; the design criteria are provided to preclude fuel assembly damage during all design basis conditions. The mechanical performance of AFA 3G™AA are evaluated in Reference [12], are emphasized as demonstrating that the design requirements are fulfilled for the fuel assemblies so as to support Safety Functions R1, R2, R3, C1 and H2.

5.4.3.2.1 Fuel Assembly Lift-off (Safety Functions R1and R3)

The hold-down assembly is designed to prevent fuel assembly lift-off in reactor cold startup, normal operation and hot shutdown conditions.

The hold-down forces of fuel assembly are determined by SYSMA code, which is developed by Framatome and introduced in Appendix 5A.

As shown in Reference [12], it is indicated that the hold-down force produced by leaf-springs could withstand the hydraulic lift force, the design requirements are fulfilled for the hold-down system so as to ensure the fuel assemblies seating on lower core plate.

The evaluation gives that the minimum contact force between fuel assembly bottom nozzle and lower core plate is greater than recommended margin in normal operation. Therefore, the hold-down spring force of fuel assembly could withstand the hydraulic force with the consideration of uncertainties.

5.4.3.2.2 Fuel Assembly Structure Components Integrity

5.4.3.2.2.1 Top Nozzle (Safety Functions R1 and H2)

Top nozzle plays an important role in component structure. It shall be properly designed to withstand all the loads transmitted by hold-down system, handling and shipping load induced by inertia. Therefore, it shall incorporate all necessary features for the installation of the hold-down system springs and withstand applied forces.

The mechanical design of top nozzle is evaluated by finite element analysis, which is SYSTUS code introduced in Appendix 5A. The results in Reference [12] show that the membrane stresses and membrane plus bending stress of ligaments of adaptor plate are much lower than the stress criteria as defined in ASME Section III Division 1, Subsection NG 3221, as given in Section 5.3.

5.4.3.2.2.2 Connection (Safety Functions R1 and H2)

The purpose of evaluation is to prove that connections could withstand all design loads, ensuring the integrity of fuel assembly. The connections include screw connection between top nozzle and guide thimbles, welding connection between grids and guide thimbles, screw connections between bottom nozzle and guide thimble plugs.

Mechanical strength is evaluated to show the design loads on this connection stay within the allowable values determined experimentally.

The evaluation in Reference [12] shows that the connection in fuel assembly can keep mechanical integrity in operation.

5.4.3.2.2.3 Bottom Nozzle (Safety Functions R1and H2)

The stress analysis is performed by finite element software, which is SYSTUS code introduced in Appendix 5A, using a complete modelling of the AFA 3G bottom nozzle (ribbed structure and filter plate).

The calculations are performed for the bounding situation of handling the fuel assembly equipped with its RCCA under 4g acceleration.

On this basis, the deflection calculated at the centre of the plate is about 0.16 mm. The membrane stresses and membrane + bending stresses in the ribbed structure of the AFA 3G^{TM}AA bottom nozzle are given in Reference [12]. They are well below the design criteria.

Thus, the mechanical strength of the ribbed structure of AFA 3G^{TM}AA bottom nozzle for the UK HPR1000 reactor is verified with margins.

5.4.3.2.2.4 Guide Thimble and Instrumentation Tube (Safety Functions R1, R2, R3)

The mechanical integrity of the guide thimble has to be checked with respect to the design criteria. This covers the acceptable stresses, axial stability, acceptable RCCA drop characteristics.

Guide thimbles and instrumentation tube play an important role in fuel assembly, not only act as the main skeleton of fuel assembly, but also provide the channel for RCCA dropping during a shutdown situation. Therefore, axial dimensional stability and mechanical stress shall be ensured to support the fuel rods.

The pressure load in guide thimbles and the impacting load on top nozzle induced by the insertion of RCCA is analysed by SAM code.

The maximum stresses determined by considering the minimum tube sections and shows that the mechanical integrity of the guide thimble is well within the acceptable criteria. It is to be noted that the presence of the MONOBLOC guide thimble with the increased-section of the dashpot led to high design margins in the lower part. And the results show that the buckling strength of the AFA 3G^{TM}AA guide thimbles is well within the allowable margins.

5.4.3.2.2.5 Grid (Safety Functions R1 and H2)

The main functions of grids are to support the fuel rods and to promote mixing of coolant.

The design justification analyses concerning the grid cover the following aspects:

- A check that it keeps its integrity and does not undergo any general deformation under design loads (fabrication, shipping, normal operation and handling), and that it provides sufficient support of the fuel rod that satisfies the dedicated criteria and contributes to the vibration response of the fuel rod.
- Its behaviour is assessed under accident loads in order to ensure that it complies with the safety criteria.

The results show that the fuel rod is always correctly supported by the grids throughout the fuel assembly lifetime.

5.4.3.2.3 Grid to Rod Fretting Wear (Safety Function C1)

The coolant flows past the fuel assembly with high velocity, acting as an energy source to fuel rod vibrating excitation, which may lead to the fretting wear at grid-to-rod contact points. The fuel assembly design shall preclude the fuel rod failure due to grid-to-rod fretting (GTRF) during normal operation. Two possible types of flow-induced rod vibration mechanisms are identified:

a) Normal flow-induced vibration which results from coolant turbulent flow, which is unavoidable in normal operational condition.

b) Abnormal flow-induced vibration results from high-speed lateral flow, including vortex shedding induced instability and the fluid elastic instability.

The analysis consists firstly in assessing analytically that the fuel rod vibratory behaviour exhibits no risk of instability that could cause sudden deterioration of the cladding due to the degradation of the contact interactions against the grid cell supports. This analysis is performed with VIBUS code, which is introduced in Appendix 5A.

The resistance of AFA 3G^{TM}AA fuel assembly to flow-induced vibrations and grid-to-rod fretting is supplemented by the Reference [15]. This latter also presents the main robustness elements in support of the AFA 3G^{TM}AA fuel assembly performance for the UK HPR1000 reactor relative to the risk of grid-to-rod fretting wear. In this regard, Reference [15] details the full-size fuel assembly mock-up tests performed under bounding flow conditions and the operational experience which supports the AFA 3G^{TM}AA performance.

Analyses and tests prove that grid-to-rod fretting performance is acceptable. It is not expected to experience fretting wear issues during normal operation.

5.4.3.2.4 Rod Growth (Safety Function C1)

The studies were intended to:

- Verify the axial dimensions of assembly to core plate gaps and gaps between nozzles and rod ends to minimise the effects of differential thermal elongation and irradiation.

- Verify the dimensions of assembly-to-assembly gaps, allowing for the growth of the AFA 3G^{TM}AA grid M5$_{Framatome}$ straps.

The gaps between fuel rods and nozzles cannot be closed to prevent interfering, which could result in fuel failure. The design evaluation addresses the irradiation creep and growth of fuel rod and fuel assembly.

The criterion is met, which is proven by the evidence of no axial elongation greater than clearances between rods and nozzles, taking account of manufacturing uncertainties, maximum rod growth and minimum fuel assembly growth induced by irradiation.

5.4.3.2.5 Coolability and Insertion of RCCAs (Safety Function R2)

This consists in verifying that the grid dimensional stability conditions are complied, even assuming a quadratic combination (SRSS) of the lateral effects of the LOCA and

Safety Shutdown Earthquake (SSE).

The dynamic response of fuel assemblies in reactor core are evaluated by CASAC code, which is introduced in Appendix 5A.

The maximum impacting forces on the structural grids and the MSMG during the LOCA and SSE condition are evaluated. The maximum forces on the structural grids and the MSMG are below the AFA 3G™AA grid buckling strengths, which are determined by the grid dynamic buckling tests, therefore. Integrity of the structural grids is confirmed.

The maximum stresses in the guide thimbles result from the SRSS combination of the stresses due to the vertical effects of the LOCA and SSE and those induced by the lateral effects of the LOCA and SSE. The evaluation shows that the maximum stresses in the AFA 3G™AA guide thimbles during LOCA and seismic comfortably meet the criteria for mechanical integrity.

The verification of guide thimble stability against elastic and plastic buckling is performed on the spans. The results are shown in Reference [12]. It is checked that the maximum stresses, in the most loaded span, comply with the elastic and plastic dynamic buckling criteria. The axial stability of AFA 3G™AA guide thimbles during the LOCA and seismic conditions is ensured with margin.

The mechanical strength of the bottom nozzle ribbed plate is verified by a finite element calculation. The membrane and membrane + bending stresses are calculated, by use of a cross-multiplication for the maximum impact load, which is defined using the SRSS combination of the LOCA and SSE maximum axial load on bottom nozzle. The top nozzle loading is negligible compared to other design loads (no impact with the upper core plate).

5.4.3.2.6 Fuel Assembly Mechanical Design Summary

The fuel assembly design evaluations demonstrate that the design requirements are fulfilled for the fuel assemblies in order to support Safety Functions R1, R2, R3, C1 and H2.

5.4.3.3 Rod Cluster Control Assembly

The justification of the RCCA considers the following issues:

a) Cladding stresses;

b) Thermal stability of absorber materials;

c) Irradiation stability of absorber materials and the cladding; and

d) Compatibility between RCCA and fuel assembly.

The RCCA evaluations in Reference [13] show that the design requirements have been satisfied in order to support Safety Functions R1, R2 and R3.

5.4.3.3.1 Internal pressure and cladding stresses (Safety Functions R1, R2 and R3)

The internal pressure is a result of the primary circuit coolant pressure imposed on the cladding in normal operation condition, overpressure in dashpot and the internal pressure of rod the under RCCA misalignment up to rod drop condition. It is the cladding load which can cause cladding stresses.

During normal, transient and accident conditions, the cladding stresses are calculated and the corresponding results are less than the stress limits.

5.4.3.3.2 Irradiation stability of absorber materials and the cladding (Safety Functions R1, R2 and R3)

The absorber must be held upright to enable it to control reactivity and to maintain its integrity within the cladding. However, irradiation results in creep and swelling on the absorber, which may bring breakage to the cladding and make RCCAs jammed in the guide thimble. Therefore, to maintain the integrity of cladding, the cladding swelling due to the irradiation shall be below the limit.

RCCAs lifetime verification has been calculated without exceeding the swelling limit, assuring the irradiation stability of absorber materials and the cladding.

5.4.3.3.3 Thermal stability of absorber materials (Safety Functions R1, R2 and R3)

The maximum absorber temperature must remain less than the melting point. The aim of this criterion is to prevent melt conditions, which would break the absorber integrity.

Maximum absorber temperatures in DBC-1 and DBC-2 are calculated and the criterion can be met.

5.4.3.3.4 Performance under insertion (Safety Functions R1, R2 and R3)

The RCCAs must be compatible with the fuel assemblies under insertion. The aim of this compatibility is to cover the top of the fissile column and for non-interference with fuel assembly screw shoulder. At the same time, the stress of spider assembly under insertion must be less than the stress limits, in order to maintain the integrity.

Considering the uncertainties, RCCAs are compatible with the fuel assemblies under insertion and the maximum stress of spider assemblies maintains a margin to the limits.

5.4.3.3.5 Rod Cluster Control Assemblies design summary

As shown above, the design criteria of RCCAs are met with margins which demonstrate the performance of RCCAs. Therefore, RCCAs could maintain the structural integrity taking into account the operating conditions, which is enough for RCCAs mechanical design to support Safety Functions R1, R2 and R3.

Evaluations of control rod drive mechanism also have essential influence on it. Chapter 6 could be referred to when required to justify the effective control.

5.4.3.4 Stationary Core Components Assembly

Analyses are performed to evaluate the stresses in the springs, the yoke arm, the welds and the rods in Reference [14].

The analysis of the rod temperature is performed with a finite element software. It is a structure analysis software based on the finite elements method. Others analyses are made with analytical methods.

The justification of the SCCA considers the following issues:

a) Stress for structural members, including the hold-down system (except for the springs) and the SCCA rods claddings;

b) Stress for connecting elements, including; and

c) Stress for hold-down springs.

5.4.3.4.1 Hold-down system integrity

The strength of the hold-down springs is verified during the whole SCCA lifetime by stress analysis. The stress values obtained are compared to the criteria.

The minimum compression load is independent of the SCCA type (hold-down system and deflections loss identical for TPA, PNSA and SNSA). The most unfavorable case is obtained for the TPA end of life (result envelop of the PNSA and SNSA) on fresh fuel assembly.

The SCCA hold-down system verification is performed for the envelop situation in operation and during handling. This situation is obtained for a fresh SCCA inserted in an EOL fuel assembly.

The stresses calculated in cold conditions and in reactor condition indicated that the stresses of hold-down springs, the yoke arm, the weld between the hub and the base plate and the pins of yoke are well below the stress criteria, their integrity can be maintained.

5.4.3.4.2 Threaded connection rod/base plate

The loads applied on this connection are:

- The maximum load in handling and shipping. The static equivalent load corresponding is an axial acceleration of 6g.
- Operating loads: the charge applied on the rod is due to the hydraulic forces.
- Manufacturing load: the minimum preload in the connection must be higher than the external efforts to prevent wear of the threaded connection.

The analysis of the threaded connection rod/base plate consists in determining the stress caused by the tightening torque applied during manufacturing. Maximum stresses are calculated for the residual tightening torque value, after having welded the nut, and are then compared to the criterion.

These results are presented in Reference [14], all criteria are met.

5.4.3.4.3 Rod integrity

This section presents the analyses performed to check the functional requirements concerning SCCA rods through its lifetime of each PNSA and SNSA, which correspond to the in-core residence time of the source assemblies.

5.4.3.2.3.1 Evaluation of SCCA rods materials temperatures

The temperatures of the components of the SCCA rods are calculated with finite element software. These calculations consist in thermal steady state analyses.

The calculation results show that the maximum temperatures of each component for PNSA and SNSA are well below the associated criteria in the SCCA rods.

5.4.3.2.3.2 Evaluation of gaps

For PNSA, the radial gaps between the alumina spacer and the cladding inner diameter on the one hand and the axial gap between the alumina spacer stack and the top end plug on the other hand are calculated in cold and hot conditions. Minimal radial gaps in cold and hot conditions are of the same order of magnitude and remain positive.

The axial gap of the primary source rod is calculated in cold and hot conditions, and the minimal axial gap is obtained in cold conditions and remains positive.

For SNSA, the radial gap between the Sb-Be pellet and the cladding is calculated, and the radial gap in the secondary source rod is calculated in cold and hot conditions. Minimal radial gaps in cold and hot conditions are remain positive.

5.4.3.2.3.3 Verification of cladding and end plug weld

The highest stress intensities in the cladding are obtained in hot conditions at beginning of life for the primary source rod and in hot conditions at end of life for the secondary source rod. Stress criteria are verified.

The verification of welds stresses between the plug and the cladding shows that criteria are respected.

The mechanical strength of the seal weld is verified considering:

- the differential pressure is applied of the whole surface of the weld;
- a safety factor for welding;
- the minimal seal weld thickness.

The analysis results in Reference [14] compares the stress values with their respective criteria for the secondary source rod seal weld.

5.4.3.2.3.4 Verification of cladding circumferential stability

It is verified that the circumferential geometry of the cladding tube remains stable during the in-core residence of the primary and secondary source rods. This guarantees acceptable conditions for cooling of the rods and prevents from any risk of jamming of the rods that could complicate handling operations of the SCCAs during outages.

For the primary source rod, there are no risks of collapse before the first cycle in the non-supported zone of the cladding.

For the secondary source rods, there are no risks of collapse before 16 calendar years in the non-supported zone of the cladding.

5.4.3.4.4 SCCA mechanical design summary

As shown above, the design criteria of SCCAs are met with margins, which demonstrate the performance of SCCAs. Therefore, SCCAs could maintain the structural integrity taking into account manufacturing loadings, in operating conditions and handling operations.

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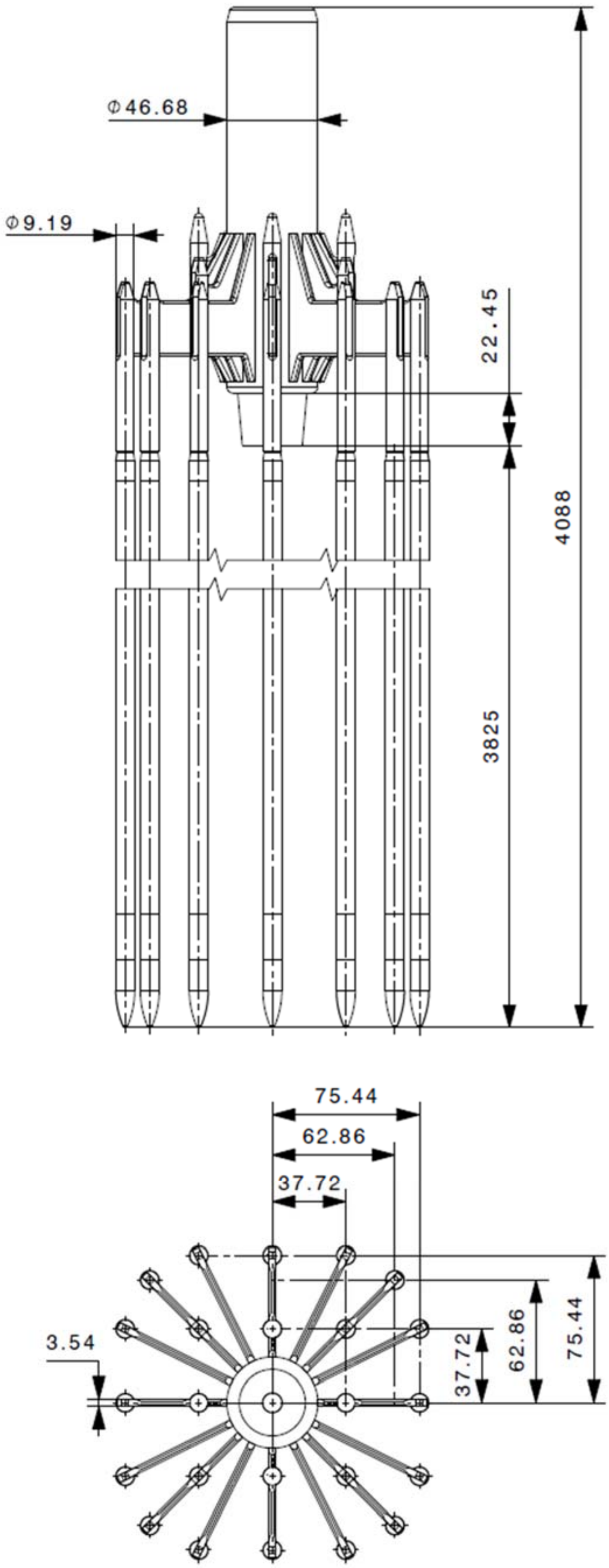
F-5.4-1 Fuel Assembly



F-5.4-2 RCCA – Main characteristics

5.5 Nuclear Design

5.5.1 Safety Functional Requirement

In this sub-chapter, the design bases for nuclear design and reactivity control systems are identified. The specified design bases derived from the safety functions listed in Sub-chapter 4.4.4 are identified.

Under DBC-1, margins are guaranteed between the plant operation parameters and the set-points for actuation of automatic or manual protective actions (Safety Function C1). Under DBC-2 and frequent DBC-3, protective actions are triggered, resulting in automatic or manual shutdown (Safety Functions R1 and R2). After the necessary corrective actions, the reactor is able to restore DBC-1. Fuel failure does not occur under DBC-1, DBC-2 and frequent DBC-3 (Safety Function C1).

5.5.2 Design Description

5.5.2.1 Reactor Core Design Description

5.5.2.1.1 Main Description

The reactor core is filled with 177 fuel assemblies. At cold conditions, the active core height is 365.76 cm, the equivalent diameter is 323 cm and the height/diameter ratio is 1.13. The main global parameters for the reactor core are shown in Table T-5.5-1. The core is surrounded by the metal reflector. The metal reflector structure is located inside the core barrel and sits on the lower support plate. It adopts an all-welded structure, which is formed by a series of W-shaped plates, C-shaped plates and ribbed plates. Details of the metal reflector presented in PCSR Chapter 6.

The first cycle (Cycle 1) adopts three types of the fuel assemblies which differ in ^{235}U enrichments so as to flatten the in-core radial power distribution. The fuel assemblies with lower enrichments are arranged adjacent to each other in a chequered pattern. The fuel assemblies with the highest enrichment are arranged at the periphery.

The transition from Cycle 1 to Equilibrium Cycle is expected to take two transition cycles to extend the cycle length from 12 months (Cycle 1) to 18 months (transition cycles and Equilibrium Cycle). During a refuelling outage, 1/3 to 1/2 of the fuel assemblies will be replaced with fresh fuel assemblies. Figures F-5.5-1 to F-5.5-4 show the (re)loading patterns of Cycle 1, Cycle 2, Cycle 3 and Equilibrium Cycle. For Cycle 2, Cycle 3 and Equilibrium Cycle, the ^{235}U enrichment of the fresh fuel is 4.45%.

In the UO_2-Gd_2O_3 fuel rod, burnable absorber material (Gd_2O_3) is blended within UO_2 to flatten the power distribution and to reduce the soluble boron concentration particularly at Beginning of Cycle (BOC). During power operation, the depletion of the burnable absorbers introduces positive reactivity, compensating for the negative reactivity due to the fuel depletion and the accumulation of fission products.

In practice, the core reloading pattern, including the quantity and arrangement of fresh fuel assemblies, depends on the energy requirement and the power histories of previous cycles.

During power operation, the fission products are accumulated along with the fuel depletion. The effect of fissile material depletion and fission product accumulation are partially compensated by the build-up of plutonium produced by the non-fission absorption of ^{238}U. At BOC, the reactor core has adequate excess reactivity to compensate for the depletion of the fissile material and the accumulation of fission product poisons. The excess reactivity is controlled by soluble boron and burnable absorbers in the core.

Considering that high soluble boron concentration can result in a positive moderator temperature coefficient, the use of burnable absorbers significantly reduces soluble boron concentration in the primary coolant so as to reduce the moderator temperature coefficient, especially at BOC where the soluble boron concentration is high. The reactivity insertion due to the burnable absorber depletion can be compensated by boron dilution. Figure F-5.5-5 presents the comparison of core depletion curves with/without burnable absorber rods based on the loading pattern of Cycle 1. In addition, the use of burnable absorber rods also flattens the in-core radial power distribution. Figure F-5.5-6 shows the layouts of the fuel assembly which represent the burnable absorber rod arrangement in a fuel assembly 17×17 array.

5.5.2.1.2 Means of Control

5.5.2.1.2.1 Reactivity Control

Core reactivity is controlled by chemical poisons dissolved in the coolant, RCCAs and burnable absorber rods as described below.

a) Chemical Poisons

Soluble boron, as boric acid, is used to control relatively slow reactivity changes associated with:

1) The moderator temperature defect during the transient from ambient temperature at cold shutdown to the hot operating temperature at zero power;

2) Transient xenon and samarium poisoning, following power changes or RCCA motions;

3) The excess reactivity required to compensate for the effects of fissile inventory depletion and the accumulation of long-life fission products; and

4) Burnable absorber depletion.

b) Rod Cluster Control Assembly

The number of RCCAs is shown in Table T-5.5-1. The RCCAs are grouped into three banks based on different functions:

1) Power compensating banks, including G1, G2, N1, N2;

2) Temperature regulating bank (R bank); and

3) Shutdown banks, including SA, SB, SC, SD.

Generally, the power compensating banks and the temperature regulating bank are also called “control banks”.

The arrangement of RCCA banks is shown in Figure F-5.5-7. The RCCAs are used to achieve shutdown state and compensate for fast reactivity changes associated with:

1) The required shutdown margin at hot zero power state, under one stuck RCCA (with maximum reactivity value) condition;

2) The reactivity compensation when power changes (power defects including Doppler and moderator effects induced reactivity changes);

3) The abnormal perturbation of boron concentration, coolant temperature or xenon concentration (with rods not exceeding the allowable rod insertion limits); and

4) Fast reactivity variation resulting from the load changes.

In order to maintain shutdown margin, insertion limit is set. The R bank position is monitored and the operator is notified by an alarm if the limit is approached.

Before the start-up, the shutdown banks are withdrawn before the control banks. During the power rise from zero to full power, the control banks are moving upwards sequentially with a prescribed overlap. The motion of RCCA banks is performed using the control rod drive mechanism (CRDM). The information of CRDM equipment design is presented in Sub-chapter 6.5.3.

c) Burnable Absorber Rod

The burnable absorber rods are used to control the excess reactivity combined with other means of reactivity control, flatten the radial power distribution and to prevent the moderator temperature coefficient from being positive at power operation. The use of burnable absorber rods reduces the critical boron concentration in the primary coolant at BOC and the gadolinium in the burnable absorber rods is depleted at a sufficiently slow rate so as to ensure the moderator temperature coefficient is non-positive throughout the cycle life as discussed in Sub-chapter 5.5.3.2.

5.5.2.1.2.2 Control of Power Distribution

a) DBC-1

During power changes, the power compensation banks are inserted/withdrawn to compensate for the reactivity variation. The four power compensation banks move in an order of G1-G2-N1-N2 in a prescribed overlap to minimise the axial power distribution perturbations due to the RCCA motions.

Boric acid is used to compensate for reactivity changes due to xenon poisoning during load following combined with small adjustments of control rod RCCA insertion.

At the power operation, the bank R is used for refined reactivity control and axial power shape adjustment. The R bank has significant negative reactivity worth to make a rapid reactivity adjustment during reactivity transients. At power operation, the bank R is handled within an operation band on the top of core (higher than the insertion limit) to minimise xenon transient effects on axial power shape. detailed information of the bank R operation band and the insertion limit is provided in *Nuclear Design Report for First Cycle* and *Nuclear Design Report for Equilibrium Cycle* (see References [16][17]).

Ex-core detectors, which are calibrated periodically by in-core detectors, monitor ΔI and instant power level. These parameters are supervised by the operators to ensure that nuclear design limits are met during operation.

The normal operating domain is divided into two regions, Region I and Region II, as shown in Figure F-5.5-8. The operating strategy is to limit ΔI within Region I in order to prevent it from deviating too far away from its reference value. However, a temporary entry into Region II is acceptable.

b) DBC-2

Under DBC-2, the extreme power distributions which lead to high maximum linear power density may appear. In this case, fuel rod integrity is ensured by limiting the centreline pellet temperature. This temperature limit corresponds to a limited maximum linear power density value at elevation z. Considering that ΔI is a function of instant power level, a limit to the maximum power level is set to ensure the axial power distribution is limited to prevent the fuel melting. Under DBC-2, fuel rod integrity is ensured through overpower ΔT and overtemperature ΔT protection.

5.5.2.1.3 Stability

5.5.2.1.3.1 Introduction of Stability

Total power oscillations are inherently stable due to the negative power coefficients maintained in the reactor core. Therefore, the spatial power oscillations in the core are readily detected and suppressed at a constant power level.

5.5.2.1.3.2 Stability Control and Surveillance

Xenon-induced spatial power oscillations appear generally after power changes or control rod motions. The radial and azimuthal xenon oscillations are self-dampening due to the negative reactivity feedback. The axial xenon oscillations are dampened by the RCCAs motion and the operation measures. The operator handles the RCCAs to maintain the axial power difference (ΔI) within the normal operating domain. The normal operating domain is divided into two regions, Region I and Region II. Under DBC-1, the reactor core is operated within Region I. In certain ranges of power, the temporary departure into Region II is also allowed, then the operator ensures that the reactor returns to Region I (ΔI is defined in Sub-chapter 5.5.2.2.3). If ΔI exceeds the boundary of the normal operating domain, the power level is automatically reduced.

The Xenon-induced spatial power oscillations are indicated by ΔI and the power tilt which are monitored by in-core and ex-core detection systems. This information can be displayed to the operator so as to monitor and intervene if necessary. The signals from the in-core and ex-core detectors and partially from the protection system are available for the operators to supervise these spatial power oscillations. The loop temperature sensors, pressuriser pressure indication and measured axial offset are provided for the overpower ΔT and overtemperature ΔT protections, which ensure the design limits are met.

In the reactor core, the online monitoring system processes information provided by the fixed in-core detectors, thermocouples and loop temperature measurements, which ensures that the radial power distribution is continuously monitored.

As mentioned above, the radial and azimuthal oscillations resulting from spatial xenon effects are stable. Both of them are self-damping without any operating or protecting actions due to the negative reactivity feedback. The provisions for the protection against non-symmetric perturbations in radial power distribution caused by equipment malfunctions (including control rod drop, rod misalignment and asymmetric loss of reactor coolant flow) are discussed in Chapter 12.

5.5.2.2 Important Parameter Description

5.5.2.2.1 Total Heat Flux Hot Channel Factor

The heat flux hot channel factor F_Q is defined as the ratio of maximum local linear power density of the fuel rod to the average linear power density of the fuel rod.

Without regard to densification effect and uncertainty,

$$F_Q = \frac{\text{Maximum linear power density of fuel rod}}{\text{Average linear power density of fuel rod}}$$

A total uncertainty factor for maximum linear power can be used to include the uncertainties and penalties defined below:

$$F_Q^T = F_Q \times F_I^{F_Q}$$

Actually, F_Q is calculated using the synthetic as follows:

$$F_Q = \max_{onz} Q\left(z\right) \text{ (without uncertainty)}$$

$$F_Q^T = \max_{onz} Q^T(z) \quad \text{(with uncertainty)}$$

where $Q_{(z)}$, the maximum linear power at elevation *z*, is defined as the ratio of the maximum linear power density at elevation *z* to the average linear power density and can be determined by the following formula:

$$Q^T(z) = \max_{x,y}[P(x,y,z)] \times F_I^{F_Q}$$

where:

$P(x,y,z)$ is the core 3D power distribution;

$F_I^{F_Q}$ is total uncertainty factor for maximum linear power, taking account of the uncertainties and penalties as follows:

F_U^N, nuclear factor,

F_Q^E, engineering factor,

F_B, rod bow factor,

F_{Xe}, xenon factor,

F_{cal}, calorimetric factor (under DBC-1).

The design limit of F_Q is shown in Table T-5.5-2.

5.5.2.2.2 Nuclear Enthalpy Rise Hot Channel Factor

The nuclear enthalpy rise hot channel factor $F_{\Delta H}^N$ is defined as the ratio of maximum fuel rod power to the average fuel rod power, with rod power defined as the integral of linear power along the rod.

$$F_{\Delta H}^{cal} = \frac{\text{Maximum fuel rod power}}{\text{Average fuel rod power}}$$

Allowing for the uncertainty:

$$F_{\Delta H}^{N} = F_{\Delta H}^{cal} \times F_{I}^{F_{\Delta H}}$$

The uncertainty $F_I^{\Delta H}$ includes the sub-factors as follows:

F_U^N , nuclear factor,

F_m , method and misalignment factor,

F_{Xe} , xenon factor.

The design limit of $F_{\Delta H}^N$ is shown in Table T-5.5-2.

5.5.2.2.3 Axial Offset

The axial offset is defined as:

$$AO = \frac{\Phi_t - \Phi_b}{\Phi_t + \Phi_b};$$

$$\Delta I = AO \times P_r$$

Φ_t and Φ_b are fluxes on the upper and lower halves of the core and P_r is relative power.

5.5.3 Design Evaluation

5.5.3.1 Fuel Burnup

Fuel burnup refers to the quantity of energy output from the fissile material in the fuel. It also provides a quantitative measure of the fuel irradiation time in the nuclear core.

Initial excess reactivity in the fuel, although not a design basis, is sufficient to maintain core criticality at full power to compensate for negative reactivity induced by xenon, samarium and other fission products. The end of cycle is reached when the concentration of soluble boron approximates to 10 ppm (natural boron).

The maximum discharge burnup of the fuel assembly and the fuel rod for each cycles are within the range proven in the fuel design analyses (Safety Function C1). Meanwhile, the fuel shall provide sufficient excess reactivity throughout the cycle length until the target discharge burnup is met. Based on the fuel management, the results on discharge burnup of all the cycles are within the burnup design limits which are shown in T-5.5-2, the evaluation results are shown in *Fuel Management Report* (see Reference [18]).

5.5.3.2 Reactivity Feedback

There are two main effects which provide the feedback to a rapid introduction of positive reactivity: The Doppler effect and the flux spectrum effect. The Doppler effect relates to the resonance absorption effect due to fuel temperature variation, and the flux spectrum effect is caused by the variation of moderator density. These reactivity effects are usually characterised by reactivity coefficients. The enrichment of fuel in the UK HPR1000 is lower than 5%, which ensures the Doppler coefficient remains negative so as to provide a rapid negative reactivity feedback to the nuclear power or fuel temperature rise. The negative moderator temperature coefficient provides a feedback to the coolant temperature or void fraction variations. The moderator temperature coefficient remains negative at power operation by reducing the soluble boron concentration using the burnable absorber rods. These approaches ensure the core provides negative reactivity feedback to any power/temperature rises (Safety Functions R1 and R2).

Since these reactivity coefficients varies through the fuel cycle, they are limited in prescribed ranges. The upper/lower limits of these ranges are used as interfaces in fault studies as conservative assumptions. These design limits for different reactivity coefficients are provided in Table T-5.5-3. The calculated results, including Doppler coefficient, moderator temperature coefficient and moderator density coefficient are shown in *Nuclear Design Basis* (see Reference [19]).

5.5.3.2.1 Fuel Temperature (Doppler) Coefficient

The fuel temperature (Doppler) coefficient is defined as the quantity of reactivity insertion due to per degree of fuel temperature increase. It is primarily a measure of the Doppler broadening of ^{238}U, ^{239}Pu and ^{240}Pu resonance absorption peaks. Doppler broadening effect of other isotopes, for example ^{236}U and ^{237}Np, is also taken into account, but their contributions to Doppler effect are much smaller than ^{238}U, ^{239}Pu and ^{240}Pu. The effective resonance absorption cross sections of fuel increase with the rise of fuel temperature, which provide negative feedback to the fuel temperature. The integral of the Doppler power coefficient with core power variation is defined as Doppler power defect, refers to the contribution of the Doppler effect to integral reactivity insertion due to the power change.

5.5.3.2.2 Moderator Coefficient

The moderator coefficient is used to quantify the reactivity variation due to the change in specific of coolant parameters such as density, temperature and void fraction. The coefficients are thus named moderator density, temperature and void coefficients.

5.5.3.2.2.1 Moderator Temperature and Density Coefficients

The moderator temperature coefficient (moderator density coefficient) is defined as the change in reactivity per degree variation of moderator temperature (moderator density respectively).

The soluble boron used in the reactor as a means of reactivity control also has an effect on moderator density coefficient because the soluble boron density decreases when the coolant temperature rises and this phenomenon introduces positive reactivity. Therefore, if the soluble poison concentration is high enough, the value of the moderator temperature coefficient becomes positive. The use of burnable absorbers reduces the initial concentration of soluble boron to maintain moderator temperature coefficient negative at operating temperature. The moderator coefficient becomes more negative with the increase of core burnup due to the reduction of soluble boron concentration.

5.5.3.2.2.2 Moderator Void Coefficient

The moderator void coefficient is defined as the change in reactivity with one percent change in the moderator void fraction. The effect of moderator void coefficient is taken into account in the shutdown margin (see Sub-chapter 5.5.3.5).

5.5.3.3 Control of Power Distribution

The power capability analysis is performed to prevent the Departure from Nucleate Boiling (DNB) and to ensure the fuel rod integrity. The design limits are imposed as follows:

a) Under DBC-1, the total heat flux hot channel factor F_Q^T does not exceed the design limit;

b) Under DBC-2, including the maximum overpower condition, the linear power density is limited to prevent the fuel from melting;

c) Under DBC-1 and DBC-2, any power distribution does not lead to DNB; and

d) The fuel management design ensures that the linear power density and the burnup in fuel rod are consistent with the assumptions applied in fuel rod mechanical integrity analysis.

For DBC-1, power capability analysis is performed to ensure that, for all the design cycles, the maximum linear power $Q_l^T(z)$ is enveloped by the LOCA limit along the active core height. The LOCA limit is shown in Figure F-5.5-9 and the evaluated results are given in *Nuclear Design Basis* (see Reference [19]). The results show that all the transients in the normal operating domain complying with the operation limit for the operating regions do not overstep the assumptions used for LOCA analyses.

For DBC-2, the power capability analysis is performed to ensure that the fuel melting limit is met thereby ensuring that all the transients which do not trigger the ΔT protection do not lead to fuel melting. The penalty functions of ΔT protection channel is shown in Table T-5.5-4. The overpower ΔT protection channel ensures the linear power density does not exceed the fuel melting limit:

$$Q_{II}^{T}(z) \leqslant F_Q^F$$

where

- $Q_{II}^{T}(z)$ is the maximum axial power at elevation z for all transients under DBC-2,

- $F_Q^F = \dfrac{\text{Linear power density limit}}{\text{Average linear power density}} = 3.286$.

The evaluation results are given in *Nuclear Design Basis* (see Reference [19]).

For the accidents in which the axial power distribution is only slightly perturbed, reference axial power distributions are applied in the calculation of DNBR, which is given in *Nuclear Design Basis* (see Reference [19]). These reference axial power distributions are proven to be the most conservative axial power distribution in terms of DNBR under DBC-1. Under DBC-2, all transients which do not trigger the overtemperature protection satisfy the DNBR design limit. The evaluation results are given in *Nuclear Design Basis* (see Reference [19]).

Otherwise, the fuel management design is optimised to keep the maximal fuel assembly and fuel rod burnup below the design limits (see Sub-chapter 5.5.3.1).

5.5.3.4 Xenon Stability

In the UK HPR1000, xenon-induced power oscillations generally appear after power changes or control rod motions.

5.5.3.4.1 Radial and Azimuthal Xenon-Induced Power Oscillation

Radial and azimuthal xenon-induced power oscillations occur due to control rod misalignment or the rod drop accident. For the UK HPR1000 design, since the moderator temperature coefficient and Doppler power coefficient are not positive at power operation, the core maintains negative reactivity feedback. This characteristic ensures that the radial and azimuthal xenon-induced power oscillations are self-damping. Simulations have been performed at Beginning of Cycle, Equilibrium Xenon (BCX), Middle of Cycle (MOC) and End of Cycle (EOC) for each designed cycle to show the self-damping ability against the radial and azimuthal xenon-induced power oscillations. The analysis is performed in simulating power shape disturbances and the free xenon-induced power oscillations at the specific burnups (BCX, MOC and EOC) for each cycle. The results show the radial and azimuthal xenon-induced power oscillations in the UK HPR1000 reactor core can be self-damping throughout each designed cycle.

5.5.3.4.2 Axial Xenon-Induced Power Oscillation

At power operation, because of power change or control rod motion, the axial power distribution can be disturbed and axial xenon-induced power oscillations appear. Different from the case of radial and azimuthal xenon-induced power oscillations, the free axial oscillations are not convergent in some conditions. Therefore, specific measures are performed to prevent or control the divergent axial power oscillations. In the analysis, calculations are performed at various burnups for each designed cycle to simulate the axial xenon-induced power oscillations in the UK HPR1000 reactor and analyse the stability against them. According to the analysis result, the axial oscillations can be prevented or controlled with prediction, RCCA banks and operating rules. If the axial xenon-induced power oscillation is not successfully dampened within the normal operating domain, the safety of the core is ensured by the protection system. If ΔI increases until the ΔT protection is triggered, the reactor will trip to keep the reactor safe.

5.5.3.5 Controlled Reactivity Insertion Rate

The maximum reactivity insertion rate due to withdrawal of RCCAs at power or boron dilution is limited. Under DBC-1, the limit for maximum reactivity insertion rate due to withdrawal of control RCCAs is set to ensure the linear power density does not exceed the maximum limit and the DNBR design limit is met under the overpower condition (Safety Functions R3 and C1).

The maximum reactivity insertion rate due to uncontrolled RCCA bank withdrawal is determined by the maximum rod withdrawal speed and the maximum differential reactivity worth of RCCA banks. Under DBC-1, the maximum reactivity insertion rate is lower than the design limit.

The reactivity insertion rate is calculated with conservative axial power and xenon distribution. The xenon burnout rate is significantly lower than the reactivity insertion rate under DBC-1. The design limit of controlled reactivity insertion rate is shown in Table T-5.5-2.

5.5.3.6 Shutdown Margin

In the UK HPR1000, the reactor trip can be achieved with rapid insertion of RCCA banks. Therefore adequate shutdown margin is maintained at power operation state or shutdown states respectively. In the analyses in which the reactor trip is taken into account, the RCCA with the highest reactivity worth is stuck out of the core (stuck rod criterion) (Safety Functions R2 and R3).

The RCCAs provide sufficient negative reactivity to achieve reactor trip and compensate for the power defect effect from full power to zero power. The reactivity feedback resulting from power drop consists of contributions from the Doppler effect, the moderator effect, the flux redistribution effect, the moderator void effect, specific uncertainties and allowances. Shutdown margin is satisfied throughout the cycle length from BOC to EOC. The design limit of shutdown margin respectively for BOC and EOC are given in Table T-5.5-5. The evaluation results are presented in *Nuclear Design Basis* (see Reference [19]).

5.5.3.7 Sub-Criticality

Sufficient sub-criticality is maintained during refuelling state and in fuel storage to prevent unexpected criticality (Safety Functions R2 and R4).

5.5.3.7.1 Criticality during Refuelling State

The criteria related to the core criticality during refuelling are shown as follows:

a) $K_{eff} < 0.99$ with all rods out; and

b) $K_{eff} < 0.95$ with all rods in.

The calculation of criticality during refuelling state is given in the *Nuclear Design Basis* Reference [19].

5.5.3.7.2 Criticality for Fuel Storage

The criticality analysis for the fresh fuel storage racks and the spent fuel storage pool in UK HPR1000 is based on the following criticality safety principles:

a) Wherever significant amount of fissile materials may be present, there shall be a system of safety measures to minimise the likelihood of unplanned criticality.

b) A criticality safety case shall incorporate the double contingency approach.

The criteria are met for fresh fuel assembly storage in the fresh fuel storage rack and fuel assembly storage in the spent fuel pool in the UK HPR1000.

a) $k_{\mathrm{eff}} < 0.95$ for fresh fuel assemblies in storage rack under normal conditions;

b) $k_{\mathrm{eff}} < 0.98$ for fresh fuel assemblies in the storage rack in the most unfavourable conditions; and

c) $k_{\mathrm{eff}} < 0.95$ for fuel assemblies storage in the spent fuel pool in the most unfavourable conditions.

The considerations and assumptions used are listed as follows:

a) Fuel assemblies have the highest enrichment and have the maximum reactivity without control rods or burnable absorber rods;

b) The Fuel assembly array is transversely infinite and is encompassed by selected conservative reflector;

c) The neutron absorption of guide thimble, instrument tube and fuel cladding in the fuel assemblies is considered. Guide thimble and instrument tube are open-ended in the calculation model, and the interior of these tubes is filled with the same material as the surroundings;

d) The soluble boron acid for neutron absorption in the water is not considered except for spent fuel storage in a fuel assembly drop accident;

e) The water temperature is chosen to generate the maximum reactivity in case of flooded conditions;

f) The thermal neutron scattering treatment S (α,β) was applied in pool water;

g) The applicable uncertainties and tolerances（in terms of design, geometrical and material specifications, manufacturing tolerances, nuclear data） are considered for fresh fuel and spent fuel;

h) The unfavourable conditions are analysed by sensitivity analysis, including change of temperature, corrosion of neutron absorber for spent fuel storage; and

i) Formation of fuel debris condition and deformation of spent fuel condition are both specially considered in fuel assemblies dropped accident.

Fuel storage in the new fuel storage rack and in the spent fuel pool are introduced in PCSR Chapter 28.6.3, and the interim storage for spent fuel is introduced in PCSR Chapter 29 Sub-chapter 29.2.

The detailed information of fresh fuel and spent fuel criticality analysis is given in the *Criticality Analysis of Fuel Storage* (see Reference [20]).

5.5.3.8 Vessel Irradiation

Neutrons generated in the reactor core can leak from the active region. When these neutrons with high energy irradiate structural material, it causes irradiation damage and degradation of structural material. Fast neutrons (energy > 1 MeV) are particularly critical to the embrittlement of the reactor pressure vessel which is critical for the safe operation. However, the structural materials, which are located between the core and the pressure vessel, including the metal reflector structure, the core barrel and relevant water gap, serve to reduce neutron flux density originating from the core.

In the UK HPR1000, the fuel assemblies with high burnup are loaded at the periphery of the reactor core so as to reduce the neutron leakage from the core and the irradiation of the RPV (except the first cycle). The distribution of the neutron fluxes in various structural components varies considerably from core to reactor vessel. The fast neutron flux at internal surface of vessel can reach 1.4×10^{10} $n \cdot cm^{-2} \cdot s^{-1}$ based on core parameters and power distribution in the equilibrium cycle (see *Fuel Management Report* and *Primary Shielding Calculations Report*, References [18][21]), which can be used for long term radiation damage estimation. Further information concerning the RPV is discussed in Chapter 17.

T-5.5-1 (1/3) Reactor Core Description

Core	
Equivalent diameter, cm	323
Average active height of the core fuel, cm	365.76
Height/diameter ratio	1.13
Fuel assemblies (cold condition)	
Number	177
Fuel rod array	17×17
Number of fuel rods per assembly	264
Lattice pitch, cm	1.26
Overall dimensions of assembly, cm×cm	21.4×21.4
Number of guide thimbles per assembly	24
Number of instrumentation tube per assembly	1

T-5.5-1 (2/3) Reactor Core Description

Fuel rod (cold condition)	
Number	46728
Outside diameter, mm	9.5
Diametric gap, mm	0.17
Thickness of the cladding, mm	0.57
Fuel pellet	
Material	Sintered UO_2
Density of UO_2 (% of theoretical density)	95
Enrichment of fuel for the UO_2 assemblies	
(% by weight ^{235}U, Cycle 1)	
• Zone 1	1.80%
• Zone 2	2.40%
• Zone 3	3.10%
Enrichment of fuel for the UO_2 assemblies	
(% by weight ^{235}U, Equilibrium Cycle)	4.45%
Control Rod	
Composition (% by weight)	80% Ag, 15% In and 5% Cd
Cladding material	Type 316L stainless steel

T-5.5-1 (3/3) Reactor Core Description

Black RCCA	
Number of black RCCAs	56
Number of absorber rods in a black RCCA	24
Grey RCCA	
Number of grey RCCAs	12
Number of absorber rods in a grey RCCA	8
Number of stainless steel rods in a grey RCCA	16
Burnable absorber rods	
{Number of assemblies with burnable absorber rods	
Material	
^{235}U enrichment, %	
• Cycle 1	
• Equilibrium Cycle	
Gd_2O_3 mass fraction, %	
• Cycle 1	
• Equilibrium Cycle	
Gd_2O_3 theoretical density, g/cm^3}	
Excess reactivity	
Maximal assembly k_{inf} (cold, clean core, zero boron)	
• Cycle 1	1.402
• Equilibrium Cycle	1.386
Maximal core k_{eff} (cold, zero power, BOC, zero boron)	
• Cycle 1	1.212
• Equilibrium Cycle	1.232

T-5.5-2 Nuclear Design Objectives and Limits[1]

Maximum discharge burnup limit for fuel rod, MWd/tU	57000
Maximum discharge burnup limit for fuel assembly, MWd/tU	52000
Average linear power density at nominal power, W/cm	179.5
{Maximum linear power $Q^T(z)$ (under DBC-1)	{ }
Total heat flux hot channel factor, F_Q^T (under DBC-2)}	{ }
Nuclear enthalpy rise hot channel factor (at hot full power), $F_{\Delta H}^N$	1.65
Maximal Reactivity insertion rate, pcm/s	55

[1] The information in this table is provided in References [18] and [19]

T-5.5-3 Design Limits of Nuclear Design Parameters[2]

Reactivity coefficients	Unit	Limit
Moderator temperature coefficient (at power)	pcm/℃	≤ 0
Moderator density coefficient (G1G2N1 inserted)	pcm/(g.cm^{-3})	< 0.580×10^{5}
Doppler temperature coefficient	pcm/℃	-4.65 ~ -1.80
Doppler power coefficient	pcm/%FP	Figure F-5.5-10
Maximum boron differential reactivity worth (natural boron)	pcm/ppm	-19.0
Effective delayed neutron fraction	/	0.00750 ~0.00440
Neutron lifetime	μs	31.0
Maximum differential reactivity worth of bank R	pcm/step	15.0 (Beginning of Cycle, equilibrium Xenon) 21.0 (EOC)

2 The detailed information presented this table is provided in Reference [22]

T-5.5-4 Penalty Functions of Overpower ΔT Protection Channel (for Safety Analysis)[3]

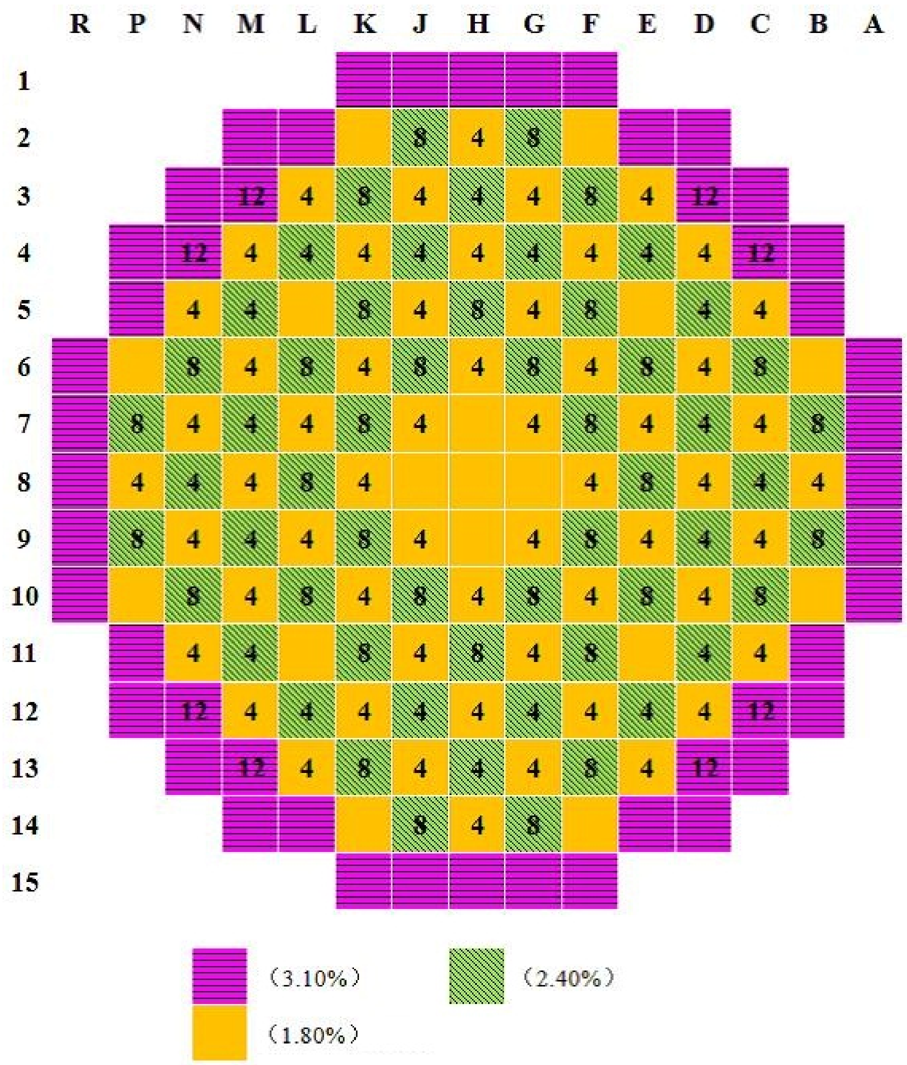
{ }

[3] { }

T-5.5-5 Shutdown Margin[4]

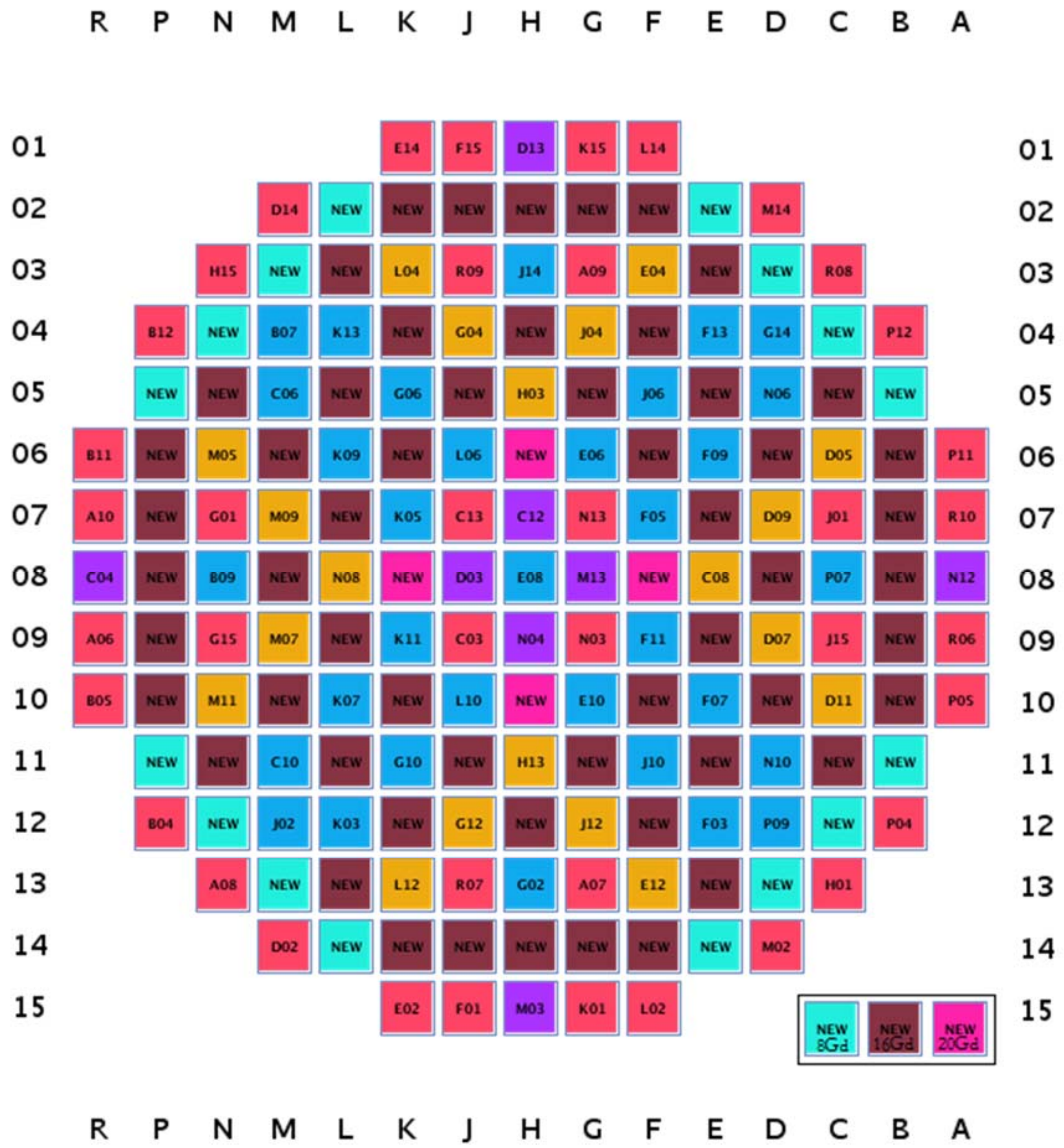
	Condition	Limit
Shutdown margin (pcm)	BOC	2000
	EOC	3300

[4] The information in this table is provided in Reference [19]



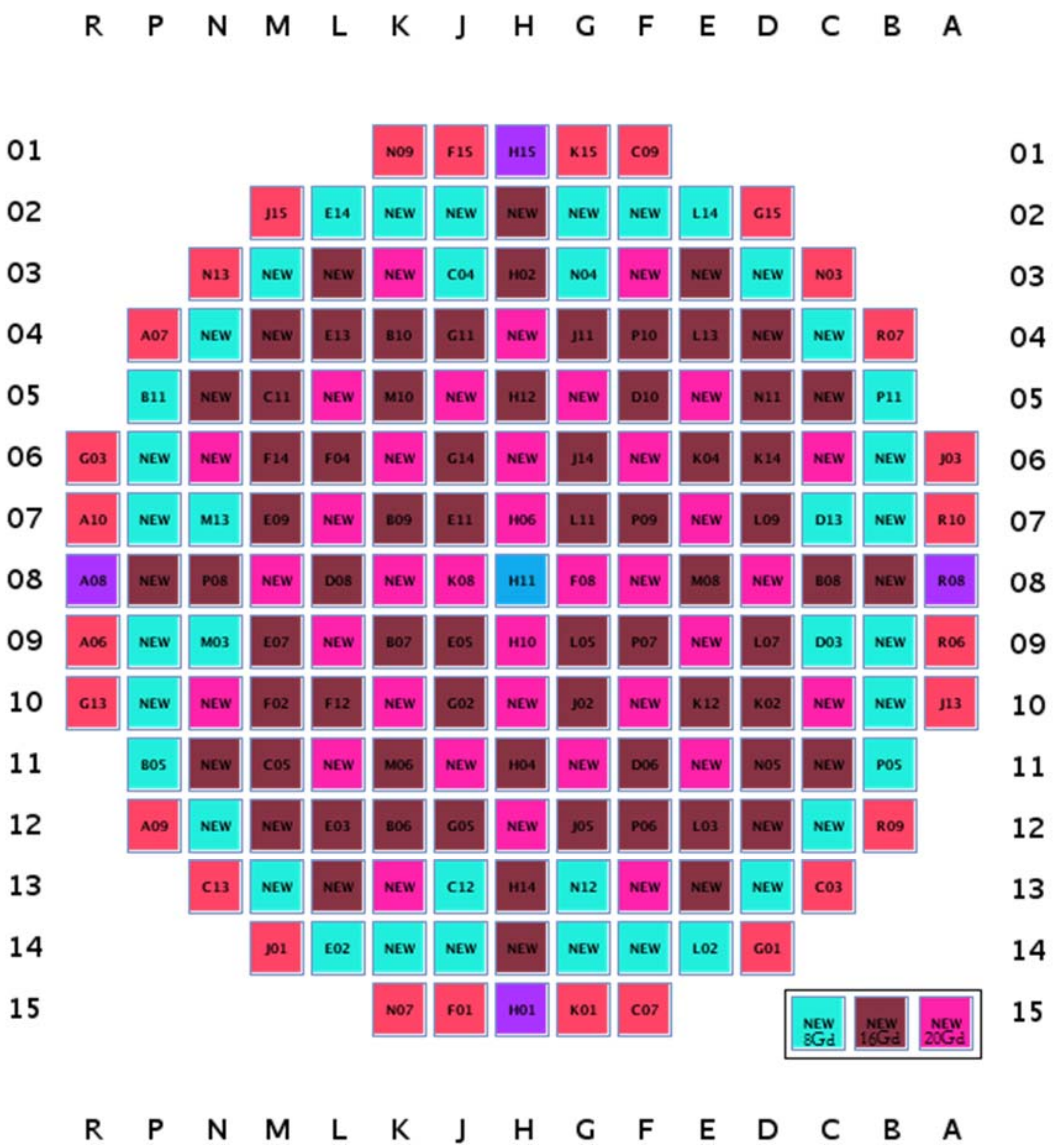
Note: The numbers on the assemblies indicate the number of burnable absorber rods.

F-5.5-1 Loading Pattern of Cycle 1



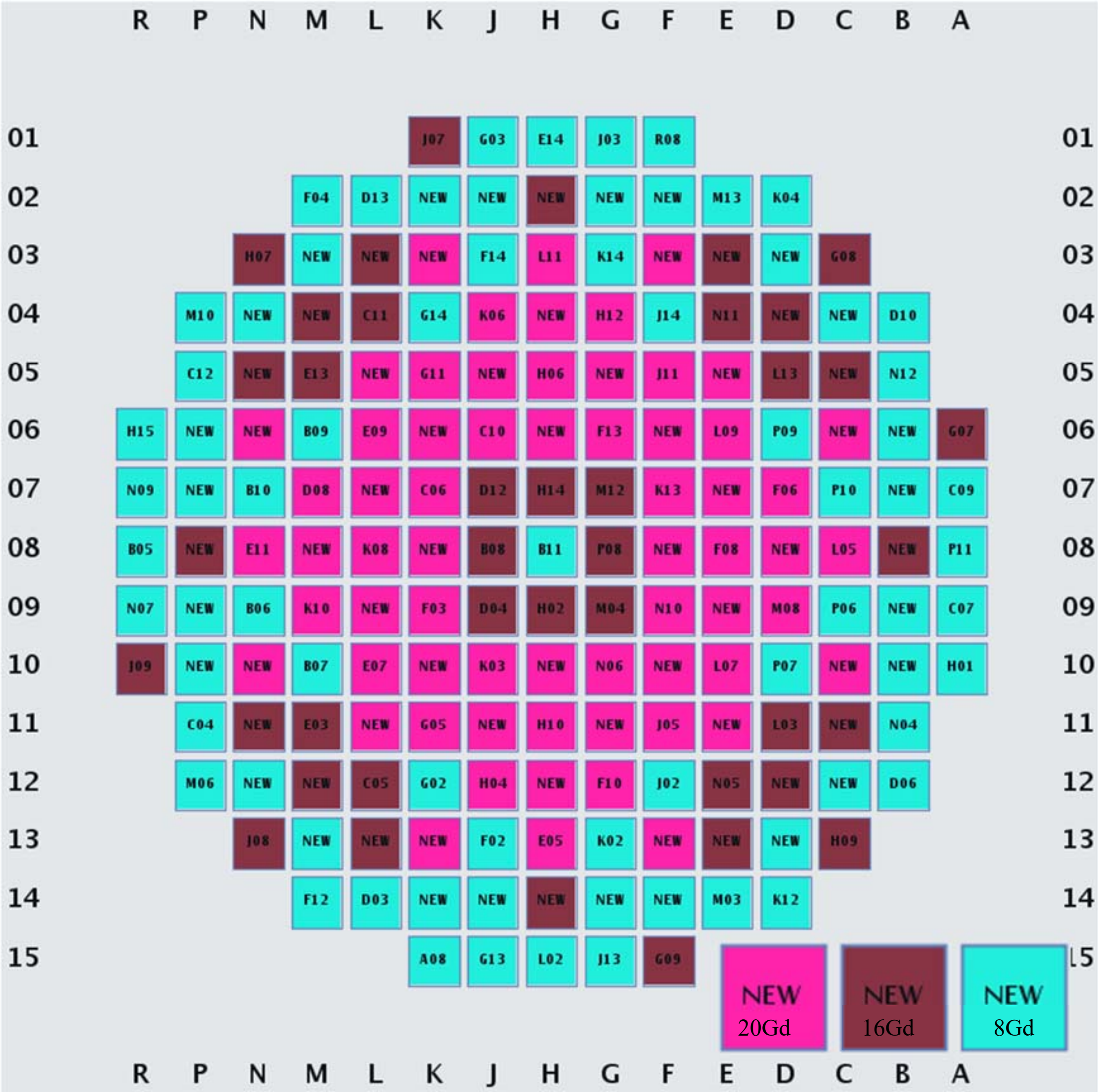
Note: The enrichment of new fuel assemblies is 4.45%.

F-5.5-2 Reloading Pattern of Cycle 2



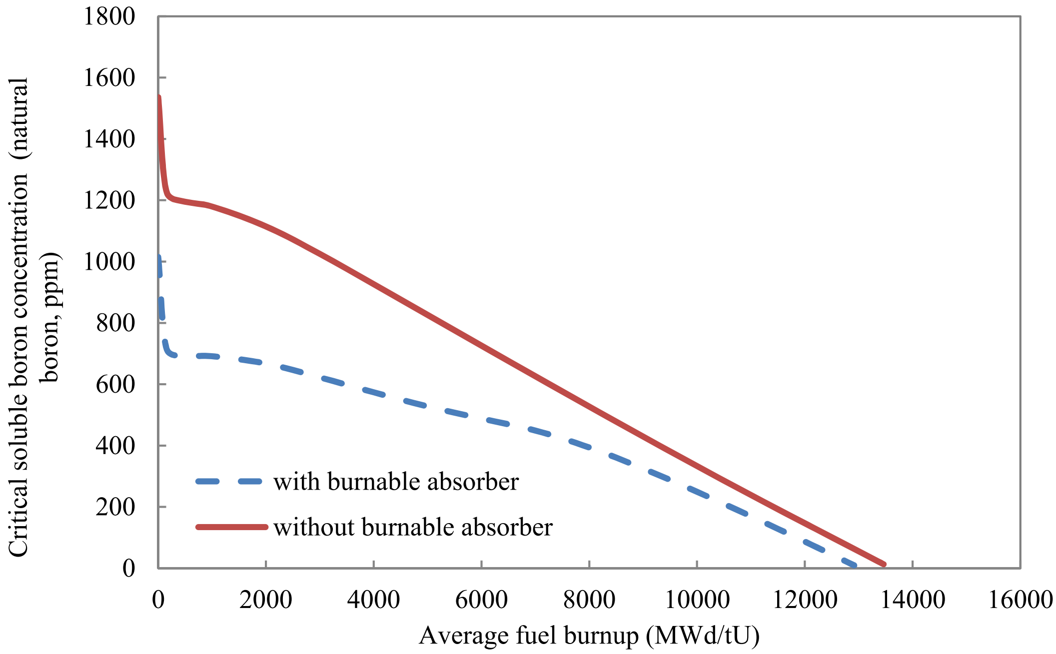
Note: The enrichment of new fuel assemblies is 4.45%.

F-5.5-3 Reloading Pattern of Cycle 3



Note: The enrichment of new fuel assemblies is 4.45%.

F-5.5-4 Reloading Pattern of Equilibrium Cycle



F-5.5-5 Critical Soluble Boron Concentration of Cycle 1 with and without Burnable Absorber

F-5.5-6 Burnable Absorber Rod Layout in Fuel Assemblies

	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A
01															
02						N2		R		N2					
03					SA		SC		SC		SA				
04				N1		G2		G1		G2		N1			
05			SA		SD		SB		SB		SD		SA		
06		N2		G2		R		N1		R		G2		N2	
07			SC		SB		SD		SD		SB		SC		
08		R		G1		N1				N1		G1		R	
09			SC		SB		SD		SD		SB		SC		
10		N2		G2		R		N1		R		G2		N2	
11			SA		SD		SB		SB		SD		SA		
12				N1		G2		G1		G2		N1			
13					SA		SC		SC		SA				
14						N2		R		N2					
15															

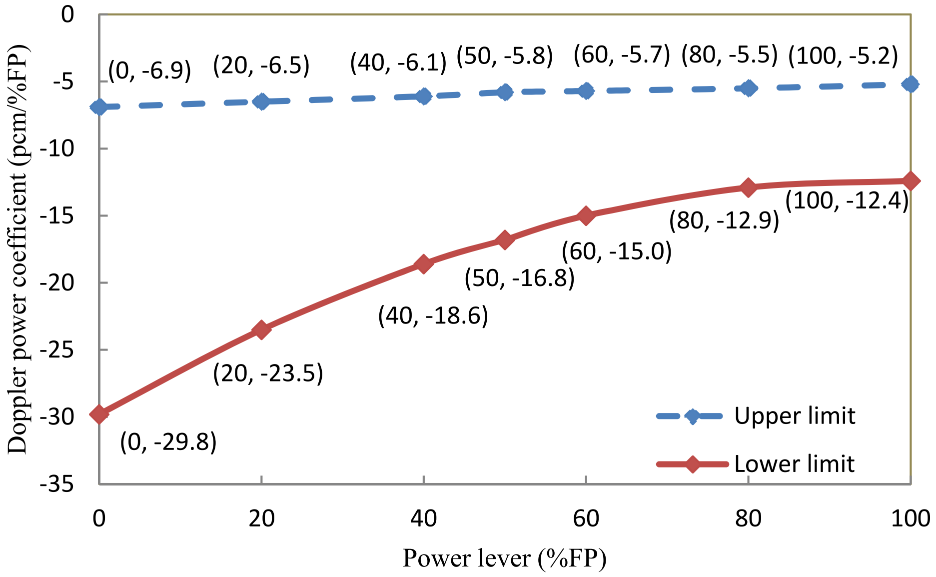
Power compensating banks	G1 G2 N1 N2	4 8 8 8
Temperature regulating banks	R	8
Shutdown RCCAs	SA SB SC SD	8 8 8 8

Note: G1 and G2 banks consist of grey RCCAs. The other banks consist of black RCCAs.

F-5.5-7 Arrangement of RCCA Banks

F-5.5-8 Normal Operating Domains (for Safety Analysis)

F-5.5-9 LOCA Limit (DBC-1)



F-5.5-10 Limit of Doppler Power Coefficient

5.6 Thermal and Hydraulic Design

5.6.1 Safety Functional Requirement

The thermal and hydraulic design of the reactor core shall comply with the following Safety Function Requirements, as defined in Chapter 4:

a) Remove heat produced in the fuel via the coolant fluid for all design basis conditions (Safety Functions H2 and H4 - Maintain heat removal from fuel stored outside the RCS but within the site); and

b) Ensure containment of radioactive substances under DBC-1, DBC-2 and frequent DBC-3 (fuel rod integrity) (Safety Function C1).

The following performance and safety criteria requirements are established for the thermal and hydraulic design of the fuel:

a) Fuel failure is not expected under DBC-1, DBC-2 or frequent DBC-3; and

b) Fraction of fuel failure is limited under infrequent DBC-3 and DBC-4 to ensure the reactor is taken to the safe state.

5.6.2 Design Description

In RPV, the primary coolant flows through the following parts in turn:

- Inlet nozzles;
- Downcomer;
- Lower plenum, including the flow distribution device;
- Bottom support plate;
- Core;
- Upper core plate;
- Upper plenum;
- Outlet nozzles.

Values of parameters related to fuel temperature and linear power density are presented in Table T-5.6-1 (*NSSS Operating Parameters,* Reference [24]) for all coolant loops in operation. The reactor is designed to ensure neither Departure from Nucleate Boiling (DNB) nor fuel centreline melting under DBC-1 and DBC-2. The overtemperature ΔT trip signal protects the core against DNB, and the overpower ΔT trip signal prevents the core against excessive power. In Chapter 12, the core thermal response under DBC-2 is described.

The objectives of reactor core thermal-hydraulic design are to determine the maximum heat removal capability in all flow sub-channels and to ensure that the core safety limits are not exceeded with the consideration of hydraulic and nuclear effects. The thermal-hydraulic design considers local variations in dimensions, power generation, flow redistribution and mixing (Safety Functions H2, H4 and C1).

5.6.3 Design Evaluation

The following design bases have been established for the thermal and hydraulic design of the reactor core to satisfy the SFRs identified in Sub-chapter 5.6.1.

5.6.3.1 Departure from Nucleate Boiling Design Basis

There is at least a 95% probability that DNB does not occur on the limiting fuel rods under DBC-1 and DBC-2, at a 95% confidence level.

DNB is a type of boiling crisis that takes place when a vapour film forms on the wall surface, which leads to a rapid decrease in heat transfer and the temperature of the wall surface continues to increase.

By preventing DNB, adequate heat transfer from the fuel cladding to the reactor coolant can be ensured; thereby fuel failure due to inadequate cooling can be prevented. This provides a way for the deterministic safety analysis to demonstrate how the results provide a challenge to the structural integrity of the fuel (Safety Function C1). The maximum fuel rod surface temperature is not a design basis since the difference between maximum fuel rod surface temperature and coolant temperature is very small during operation in the nucleate boiling region. Limits provided by the reactor control and protection systems are such that this design basis is met for transients associated with DBC-1 and DBC-2, including overpower transients. The DNBR is defined as follows:

$$DNBR = \frac{q''_{DNB.N}}{q''_{loc}}$$

$$q''_{DNB.N} = \frac{q'_{CHF}}{F}$$

Where: $q''_{DNB.N}$: The predicted heat flux considering the influence of axial heat flux distribution

q'_{CHF} : Uniform Critical Heat Flux (CHF) predicted by the CHF correlation

F : The shape factor of non-uniform axial heat flux distribution

q''_{loc} : The actual local heat flux

FC2000 CHF correlation and W3 CHF correlation are used to calculate the expected critical heat flux. FC2000 CHF correlation is used downstream of the first mixing grid of fuel assembly because FC2000 is suitable for AFA 3G^{TM} AA fuel assemblies equipped with Mid Span Mixing Grid (MSMG), fuels assemblies retained for the UK HPR1000 reactor (*FC2000 CHF Correlation,* Reference [25]). W3 CHF correlation is used upstream of the first mixing grid of fuel assembly (*UK HPR1000 - W3 CHF*

Correlation, Reference [26]).

The minimum calculated DNBR shall be greater than the DNBR design limit to ensure fuel integrity.

5.6.3.1.1 Statistical DNBR Design Limit

For most DBC-2 accidents, the DNBR design limit is determined by using the FC2000 CHF correlation and statistical method. The statistical method uses the statistics theory to comprehensively consider correlation uncertainty, plant thermal-hydraulic parameters uncertainty, code uncertainty, and transient calculation uncertainty.

Since the fuel rod bow has an adverse effect on the DNBR safety analysis, the DNBR design limit takes into account the effect of the rod bow penalty. Rod bow in relation to DNBR is described in Sub-chapter 5.6.3.1.4.4.

The statistical DNBR design limit is { } (see Reference [23] *Thermal Hydraulic Design*).

5.6.3.1.2 Deterministic DNBR Design Limit

For accidents where limiting thermal-hydraulic conditions are outside the validity domain of the statistical method, a deterministic analysis shall be performed with plant parameter uncertainties applied to the initial conditions of the plant transient. Minimum DNBR shall be compared to the deterministic DNBR design limit including the rod bow penalty.

$$\text{Deterministic DNBR design limit}=\frac{\text{Owen criterion}}{1-\text{rod bow penalty}}$$

The Owen criterion of FC2000 CHF correlation and the deterministic DNBR design limits with FC2000 CHF correlation are described in Reference [23]. The deterministic DNBR design limits with FC2000 CHF correlation is { } (see Reference [23]).

The design limits of the W3 CHF correlation and the deterministic DNBR design limits with W3 CHF correlation are also described in Reference [23]. The deterministic DNBR design limits with W3 CHF correlation are as follows (provided in Reference [23]):

{ }

5.6.3.2 Fuel Temperature Design Basis

Under DBC-1 and DBC-2, there is at least a 95% probability at a 95% confidence level that the fuel pellet temperature shall be below its melting temperature (Safety Function H2).

The melting temperature of uranium dioxide that is not irradiated is 2810℃ (*Thermal*

Hydraulic Design, Reference [23]). And the actual melting temperature of uranium dioxide is affected by a number of factors. Among these factors, it is the irradiation that has the greatest impact. The melting temperature of uranium dioxide decreases 32℃ per 10,000MWd/tU (*Thermal Hydraulic Design,* Reference [23]). The melting temperature of uranium dioxide used in design is 2590℃ (*Thermal Hydraulic Design,* Reference [23]).

By precluding fuel pellet melting, the fuel geometry is preserved and possible adverse effects of molten fuel pellet on the cladding are eliminated.

5.6.3.3 Core Flow Design Basis

The minimum value of thermal design flowrate that pass through the fuel rod region of the core is 93.5% of the available flow, and this is effective for fuel rod cooling (Safety Function H1).

Core cooling evaluations are based on the thermal design flowrate (minimum flowrate) entering the Reactor Pressure Vessel (RPV). A total of 6.5% of the flowrate is taken as the maximum bypass flowrate. This includes RCCA guide thimble and instrumentation tube cooling flow, leakage flow through the metal reflector structure, core peripheral assemblies bypass flow, head cooling flow, and leakage flow to the RPV outlet nozzles.

5.6.3.4 Hydrodynamic Instability Design Basis

Modes of operation associated with DBC-1 and DBC-2 do not lead to hydrodynamic instability (Safety Functions H2 and C1).

Hydrodynamic instability in the nuclear reactor is not desired, as the thermal-hydraulic conditions changes due to hydrodynamic instability may result in the critical heat flux lower than that in steady and continuous flow conditions, or cause undesirable forced vibration to reactor internals.

5.6.3.5 Departure from Nucleate Boiling Ratio

The minimum DNBR of the limiting flow channel is located downstream of the location of peak heat flux (hot spot). This is because of the increase of enthalpy rise downstream.

The influence of typical cell and guide tube cold wall cell, the uniform and non-uniform heat flux distributions, and the changes of rod heating section length and lattice spacing are considered in FC2000 CHF correlation and W3 CHF correlation.

The sub-channel analysis code LINDEN is used to analyse the flow distribution in the core and the local conditions in the hot channel.

5.6.3.5.1 CHF Correlation Description

The FC2000 CHF correlation development was based exclusively on critical heat flux data from tests performed on Framatome 17x17 fuel assemblies with and without Mid Span Mixing Grids. This correlation based on local fluid conditions accounts directly for both typical and thimble cold wall cell effects, uniform and non-uniform heat flux profiles, and variations in rod heated length and in grid spacing (*FC2000 CHF Correlation,* Reference [25]).

W3 CHF correlation has been established by L. S. Tong based on experimental data of CHF tests performed in simple geometries, like raw tubes and annular spaces with heated wall(s) (*UK HPR1000 - W3 CHF Correlation,* Reference[26]).

The validity domain of FC2000 CHF correlation is described in Reference [25] *FC2000 CHF Correlation*, and the validity domain of W3 CHF correlation is described in Reference [26] *UK HPR1000 - W3 CHF Correlation*.

5.6.3.5.2 Mixing Effect between Sub-channels

In a rod bundle, the flow channels formed by four adjacent fuel rods are open to each other through the gap between two adjacent fuel rods. There is a cross-flow between channels due to the pressure difference. The mixing effect between sub-channels can reduce enthalpy rise in the hot channel.

The exchange of turbulent momentum and enthalpy between the channels can be calculated by LINDEN.

5.6.3.5.3 Engineering Hot Channel Factor

5.6.3.5.3.1 Definition of Hot Channel Factor

The total hot channel factors for heat flux and enthalpy rise are defined as the maximum to core average ratios of these quantities. The heat flux hot channel factor considers the local maximum linear heat generation rate at the hot spot, and the enthalpy rise hot channel factor involves the maximum integrated value along the hot channel. The engineering factors take into account the manufacturing variation in fuel rod and fuel assembly materials and geometry. Two types of engineering hot channel factors F_Q^E and $F_{\Delta H}^E$ are defined below.

5.6.3.5.3.2 Heat Flux Engineering Hot Channel Factor

The heat flux engineering hot channel factor F_Q^E is used to calculate the maximum heat flux on the fuel rod surface. This factor is determined by statistically combining the impacts on the heat flux from the tolerances of the fuel pellet diameter, density, enrichment, eccentricity and fuel rod diameter. The measured manufacturing data for the 17×17 fuel rods are used for validation and verification, and the manufacturing data of 95% of the limit fuel rods cannot exceed this design value at 95% confidence level.

5.6.3.5.3.3 Enthalpy Rise Engineering Hot Channel Factor

The enthalpy rise engineering hot channel factor $F_{\Delta H}^{E}$ is determined by statistically combining the influences of manufacturing tolerances for fuel density and enrichment on enthalpy rise. $F_{\Delta H}^{E}$ is a direct multiplier of the hot channel enthalpy rise.

5.6.3.5.4 Flow Distribution

When the hot channel enthalpy rise is calculated, the effects of core coolant flow on distribution results need to be considered. These effects are discussed below.

5.6.3.5.4.1 Inlet Flow Maldistribution

Inlet flow maldistribution in core thermal performances is discussed in Sub-chapter 5.6.3.3.3. A design basis of 5% reduction in coolant flow to the hot assembly is used in the sub-channel analysis.

5.6.3.5.4.2 Flow Redistribution

It is considered that local or general boiling increases the channel flow resistance which reduces the hot channel flowrate. The effect of the non-uniform power distribution is inherently considered in the sub-channel analysis for every operating condition which is evaluated.

5.6.3.5.4.3 Flow Mixing

A sub-channel mixing model is incorporated in LINDEN and is used in the reactor design. The mixing vanes included in the spacer grid design induce additional flow mixing between the various flow channels in a fuel assembly, as well as between adjacent assemblies. This mixing reduces the enthalpy rise in the hot channel caused by a local power peak or an unfavourable mechanical deviation.

5.6.3.5.4.4 Effect of Rod Bow on DNBR

The effect of fuel rod bow is considered in the DNBR safety analysis. In order to offset the effect of rod bow, the rod bow penalty factor is added in the calculation of DNBR design limits.

The maximum rod bow penalty considered in the DNBR safety analysis is determined with an assembly average burn-up of 28,000 MWd/tU (*Thermal Hydraulic Design,* Reference [23]). For burn-ups greater than 28,000 MWd/tU, the effect of $F_{\Delta H}$ decrease on DNBR can compensate for the effect of rod bow penalty increase on DNBR. (*Thermal Hydraulic Design,* Reference [23])

5.6.3.6 Linear Power Density

The core average and maximum linear power density are given in Table T-5.6-1.

5.6.3.7 Core Hydraulic

The core hydraulic design supports the core flow basis of providing a minimum flowrate of 93.5% of the available flow.

5.6.3.7.1 Core and Reactor Pressure Vessel Pressure Drop

The pressure drop is caused by viscosity of fluid and geometric changes in the flow channel. The fluid is assumed to be incompressible, turbulent and single-phase. These assumptions are used in the calculation of the pressure drop in core and RPV in order to determine the loop flow in the reactor coolant system. Two-phase flow is not considered in the calculation of the pressure drop in core and RPV, as the average void fraction of the core is negligible in the design.

The two-phase flow is considered in the thermal analysis of core sub-channel. The pressure drop of the core and RPV is calculated using the following formula:

$$\Delta P_L = (K + f\frac{L}{De})\frac{\rho V^2}{2}\cdot 10^{-6}$$

Where: ΔP_L : Unrecoverable pressure drop, MPa

ρ : Fluid density, kg/m^3

L : Length, m

De : Equivalent diameter, m

V : Fluid velocity, m/s

K : Form loss coefficient, dimensionless

f : Friction loss coefficient, dimensionless

For each component of the core and RPV, a constant fluid density is assumed. Due to the complicated geometrical shape of the core and RPV, it is hard to obtain a precise analysis value for the coefficients of form loss and friction resistance. Therefore, experimental values of these coefficients shall be obtained through hydraulic simulation of geometrically similar models.

The core pressure drop includes those of the fuel assemblies, lower support plates and upper core plates. They are calculated according to the nominal flow under the actual operation conditions of the power plant.

The characteristics of core pressure drop are determined according to the hydraulic tests carried out for 17×17 fuel assemblies over a wide range of Reynolds numbers. The pressure drop of the other parts of RPV except the core is obtained with form loss correlation obtained according to the hydraulic test data.

5.6.3.7.2 Bypass Flow

The following flow paths for core bypass flows are considered:

a) Flow through the spray nozzles into the upper head for head cooling purposes;

b) Flow entering into the RCCA guide thimbles and the instrumentation tubes to cool the control rods, the thimble plug rods and neutron sources;

c) Leakage flow from the RPV inlet nozzle directly to the RPV outlet nozzle through the gap between the RPV and the barrel;

d) Flow through the metal reflector structure for the purpose of cooling these components, but considered useless for core cooling; and

e) Flow in the gaps between the fuel assemblies on the core periphery and the adjacent metal reflector structure.

The maximum or minimum design value of the above bypass flow is used in the core thermal-hydraulic design in a conservative method.

5.6.3.7.3 Inlet Flow Distribution

The inlet flow distribution is non-uniform. A 5% reduction of the hot assembly inlet flow is assumed, which is proved to be conservative by inlet flow distribution test.

Investigations with LINDEN involving decreasing the flow rate through a limited inlet area of the core indicate that there is a rapid redistribution within one-third of the core height and that consequently the inlet flow maldistribution has a negligible impact on the hot channel DNBR, which occurs at the upper part of the core. This flow redistribution is due to the redistribution of fluid velocities.

5.6.3.7.4 Friction Factor Correlation

The friction factor f is expressed as follows:

$$f = f_{sp}\, Y(\alpha, G, \varnothing)$$

Where f_{sp} concerns single phase flow and Y (α, G, ∅) is a corrective factor for two-phase flow. α is void fraction. G is mass velocity. ∅ is wall heat flux. The two-phase correlation is only used on the sub-channel analysis and not in the design of the normal operation core flow rate and pressure drop. Then single phase factor is defined as:

$$f_{sp} = f_{iso} A(\varnothing)$$

Where f_{iso} deals with isothermal conditions and $A(\varnothing)$ takes into account heat flux effects (viscosity decreases near the rod).

5.6.3.8 Hydrodynamic and Flow Power Coupled Instability

Thermohydrodynamic instabilities are undesirable in the nuclear reactor, because they may change the thermal-hydraulic conditions thus resulting in a DNB heat flux lower than that in steady and continuous flow conditions, or cause undesirable forced vibration to reactor internals.

The Ledinegg type of static instability and the density wave type of dynamic instability are considered for the UK HPR1000 plant operation.

5.6.3.8.1 Static Instability

Ledinegg instability refers to a sudden change of flow from one steady state to another. This instability occurs when the slope of the reactor coolant system pressure drop - flow rate curve ($(\partial \Delta p / \partial G)_{internal}$) becomes algebraically lower than the loop supply (pump head) pressure drop - flow rate curve ($(\partial \Delta p / \partial G)_{external}$). The criterion for stability is thus:

$$(\partial \Delta p / \partial G)_{internal} \geq (\partial \Delta p / \partial G)_{external}$$

The head curve of reactor coolant pump has a negative slope, i.e. $(\partial \Delta p / \partial G)_{external} < 0$ while the pressure drop-flow curve of reactor coolant system during its operation under DBC-1 and DBC-2 has a positive slope, i.e. $(\partial \Delta p / \partial G)_{internal} > 0$. Therefore, Ledinegg instability will not occur.

5.6.3.8.2 Dynamic Instability

The mechanism of density wave oscillations in a heated channel can be described briefly as an inlet flow fluctuation that produces an enthalpy perturbation. This perturbs the length and the pressure drop of the single-phase region and causes steam quality or void perturbations in the two-phase region of an ascending fluid. The steam quality and length perturbations in the two-phase region create two-phase pressure drop perturbations. However, since the total pressure drop across the core is maintained by the characteristics of the fluid system external to the core, then the two-phase pressure drop perturbation feeds back to the single-phase region. These resulting perturbations can be either attenuated or self-sustained.

A simple method has been developed by Ishii for parallel closed channel systems to evaluate whether a given condition is stable with respect to a density wave type of dynamic instability. The application of this method to the UK HPR1000 indicates that a large margin to density wave instability exists. The method of Ishii applied to the UK HPR1000 design is conservative due to the parallel open channel feature of the UK HPR1000 core. For such core, there is little resistance to lateral flow leaving the flow channels of high power density. There is also energy transfer from high power density channels to lower power density channels. This coupling with cooler channels leads to the judgment that an open channel configuration is more stable than the above closed channel configuration under the same boundary conditions.

The flow mixing between channels shows that open channels are more stable than closed ones under the same restrictions. Therefore, hydrodynamic instability will not occur in the UK HPR1000.

5.6.3.9 Uncertainties

5.6.3.9.1 Uncertainties in Pressure Drops

The pressure drops of core and RPV are based on the best estimate flow. The uncertainties of these parameters are based on the test results.

5.6.3.9.2 Uncertainties due to Inlet Flow Maldistribution

The influence of non-uniform distribution of core inlet flow used in core thermal-hydraulic analysis on uncertainties is discussed in Sub-chapter 5.6.3.3.3.

5.6.3.9.3 Uncertainty in DNB Correlation

The uncertainty of DNB correlation is based on standard deviation and average value of the ratios of measured CHFs to CHFs predicted by correlation.

5.6.3.9.4 Uncertainties in DNBR Calculations

The uncertainties in the DNBR calculated by sub-channel analysis due to nuclear peaking factors are accounted for by applying conservative values of the nuclear peaking factors and including measurement error allowances. Meanwhile, conservative values for the engineering hot channel factors are used, as described in Sub-chapter 5.6.3.1.3. In addition, flow distribution is considered in a penalising way as discussed in Sub-chapter 5.6.3.1.4.

5.6.3.9.5 Uncertainties in Flowrates

The thermal design flow which includes the uncertainties between estimation and measurement is used in the core thermal performance calculation.

5.6.3.9.6 Uncertainties in Hydraulic Loads

The hydraulic load on the fuel assembly is calculated based on the pump overspeed transients, in which the flow generated is 20% greater than the mechanical design flow. The mechanical design flow is greater than the best estimate flow under actual operation conditions of the power plant.

5.6.3.9.7 Uncertainty in Mixing Coefficient

The conservative value of the mixing coefficient k_T is introduced in LINDEN for reactor calculations.

5.6.3.10 Summary of Thermal-Hydraulic Evaluation

Sub-Chapter 5.6.3 describes the thermal-hydraulic design bases, DNBR, linear power density, core hydraulic, hydrodynamic and flow power coupled instability, and uncertainties. This sub-chapter demonstrates that the thermal-hydraulic design of the UK HPR1000 can meet the design bases.

T-5.6-1 (1/2) Reactor Thermal and Hydraulic Characteristics of UK HPR1000

Design parameters	
Reactor thermal power, MWt	3150
Heat generated in fuel, %	97.4
System pressure (nominal value), MPa	15.5
$F_{\Delta H}^{N}$	1.65
Coolant flowrate	
Total thermal design flowrate, m^3/h	72,000
Effective flowrate for heat transfer, m^3/h	67,320
Effective flow area for heat transfer, m^2	4.33
Average flow rate along fuel rods, m/s	4.32
Coolant temperature (based on thermal design flowrate)	
Nominal inlet temperature, ℃	288.6
Average temperature rise in the RPV, ℃	36.8
Average temperature rise in the core, ℃	39.1
Average temperature in the core, ℃	308.1
Average temperature in the RPV, ℃	307.0

T-5.6-1 (2/2) Reactor Thermal and Hydraulic Characteristics of UK HPR1000

Heat transfer	
Heat transfer surface area of the core, m^2	5094.7
Average surface heat flux, W/cm^2	60.22
Maximum surface heat flux under nominal conditions, W/cm^2	147.54
Average linear power density, W/cm	179.5
Peak linear power density during normal conditions, W/cm	439.8
Peak linear power density caused by overpower transients/operator errors (assuming maximum overpower of 120%FP),W/cm	≤ 590
Power density kW/l (core)	102.5
Specific power, kW/kgU	38.78
Fuel centre temperature	
Fuel centre melting temperature, ℃	2590

5.7 ALARP Assessment

Based on the Reference [27], the scope of Generic Design Assessment (GDA) is identified in fuel and core design, including the fuel system design, nuclear design, thermal-hydraulic design and fuel handling and storage (before loading and after loading).

The ALARP demonstration for fuel system is presented in Reference [28] by Framatome. And the ALARP demonstration for fuel handling and storage is presented in Reference [29]. Therefore, this sub-chapter mainly presents the ALARP demonstration of nuclear design and thermal-hydraulic design.

5.7.1 Holistic ALARP Assessment

5.7.1.1 Evolution of the HPR1000

The UK HPR1000 technology, developed by China General Nuclear Power Corporation (CGN), is based on improvements of the Chinese Pressurised Reactor (CPR1000), Chinese Improved Pressurised Reactor (CPR1000^{+}) and Advanced Chinese Pressurised Reactor (ACPR1000). The overall evaluation of the UK HPR1000 is introduced in *HPR1000 R&D History*, Reference [30]. The nuclear design and the thermal-hydraulic design of UK HPR1000 are mature design.

The fuel system design evolution of UK HPR1000 is presented in Reference [28] by Framatome.

5.7.1.2 Compliance with RGP

RGP is typically defined in the following non-exhaustive list of sources (see Reference [31]):

- Safety Assessment Principles (SAPs) and Technical Assessment Guides (TAGs) of Office for Nuclear Regulation (ONR);
- International Atomic Energy Agency (IAEA) Safety Standards;
- Recognised design codes and standards;
- Approved Codes of Practice (ACoPs);
- Western European Nuclear Regulators Association (WENRA) Safety Reference Levels for reactors, decommissioning, and the storage of radioactive waste and spent fuel.

The following SAPs and TAGs are related to nuclear design and thermal-hydraulic design.

a) Safety Assessment Principles for Nuclear Facilities, Revision 1 (2020), ONR;

b) Technical Assessment Guides related to fuel and core design:

- Safety of Nuclear Fuel in Power Reactors, NS-TAST-GD-075 Revision 3 (2020), ONR.

The key codes and standards are presented in Sub-chapter 5.3. Among these codes and standards, 2 of them are related to nuclear design and thermal-hydraulic design, and 6 of them are related to fuel system design (see Reference [5] and [32]). The codes and standards related to fuel handling and storage (before loading and after loading) are presented in mechanical engineering design area.
This sub-chapter only presents the compliance of nuclear design and thermal-hydraulic design with RGP. The compliance of fuel system design is presented in Reference [28] by Framatome. And the compliance of fuel handling and storage (before loading and after loading) is presented in Reference [29].

The analysis shows that nuclear design and thermal-hydraulic design are well compliant with RGP.

5.7.1.3 OPEX Review

OPEX from European Pressurised Reactor (EPR), Advanced Passive 1000 (AP1000), Advanced Boiling Water Reactor (ABWR) and CPR1000 has been considered in order to optimise UK HPR1000 design.

The sources of OPEX mainly include the following aspects:

a) Lessons learnt from previous GDA projects

   1) PCSR and its supporting documents of UK EPR, UK AP1000 and UK ABWR;

   2) Regulatory Queries (RQs), Regulatory Observations (ROs) and Regulatory Issues (RIs) issued by ONR for UK EPR, UK AP1000 and UK ABWR;

   3) Assessment reports issued by ONR for UK EPR, UK AP1000 and UK ABWR.

b) International OPEX sharing from authority websites, for example, Nuclear Institute and Nuclear Energy Agency, etc.

c) Engineering and design documents of CPR1000.

The review against OPEX that identified contains the thermal-hydraulic characteristics and the core dimension optimization (see Reference [3]). The analysis shows that nuclear design and thermal-hydraulic design are well compliant with OPEX.

5.7.1.4 Risk Assessment

With the compliance analysis with RGP and OPEX, no gap or risk has been identified in nuclear design or thermal-hydraulic design.

Risks from other areas shall also be considered in ALARP analysis. For fuel and core topic area, only risks form fault study topic area is received. In the fault study topic area, during the analysis process of no fuel failure analysis for frequent faults, the analysis results show that all frequent transients, except the Inadvertent Opening of One Steam Generator Relief Train or of One Safety Valve, are demonstrated no fuel failure (see Reference [33]). For the Inadvertent Opening of One Steam Generator Relief Train or of One Safety Valve, the improvement plan is to optimize the setpoint of overpower ΔI penalty function. With this improvement, this transient can also be demonstrated no fuel failure (see Reference [34]).

And an ALARP assessment for DNB analysis has been done. After this assessment, several gaps and shortfalls have been identified. Some potential improvements to the core design that may reduce the predicted number of fuel failures in faults are also identified. The assessment also shows that DNB analysis is ALARP. The detailed information is shown in *Supporting Report on ALARP Assessment for DNB Analysis*, Reference [35].

5.7.2 Specific ALARP Assessment

In this topic area, a category one modification is considered. It is the modification on setpoint of “Overpower ΔT Reactor Trip” signal (see Reference [23] and Reference [34]).

5.7.3 ALARP Conclusion

This sub-chapter presents the ALARP demonstration of nuclear design and thermal-hydraulic design for fuel and core design topic area. And the current design is well compliant with RGP and OPEX. In addition, in the fault study topic area, all frequent transients are demonstrated no fuel failure. A category one modification on setpoint of “Overpower ΔT Reactor Trip” signal is considered during this demonstration. And DNB analysis is also demonstrated ALARP. Therefore, the core nuclear design and the core thermal-hydraulic design of the UK HPR1000 are ALARP (see Reference [3]).

5.8 Commissioning and Testing

5.8.1 Reactor Core Physics Test

Nuclear design calculations guarantee that the reactor core physics parameters do not exceed the safety values. Reactor core physics tests check that the reactor core physics parameters are consistent with design predictions and thereby ensure that the core will be operated as per the design intent.

5.8.2 Tests Prior to Initial Criticality

Reactor coolant flow tests are performed following fuel loading after plant startup. The results of the successive enthalpic balances performed allow for the determination of the coolant flow rates at reactor operating conditions. These tests verify that proper coolant flow rates have been used in the core thermal and hydraulic analysis.

5.8.3 Initial Power and Plant Operation

Core power distribution measurements are made at several core power levels at the start of each cycle and are compared with predicted values. These tests are used to confirm that conservative peaking factors are used in the core thermal-hydraulic analysis. Tests are also undertaken each month, and compared with predicted power distributions.

5.8.4 Component and Fuel Inspection

Fabrication measurements critical to thermal and hydraulic analyses are obtained to verify that the uncertainty included in the engineering hot channel factor in the design analysis is conservative.

Further detailed site specific arrangements for the UK HPR1000 commissioning and testing activities will be presented during the Nuclear Site Licensing phase in conjunction with the site license.

5.9 Ageing and EMIT

Fuel assembly mock-up tests including mechanical tests (see Reference [36]) and flow loop tests (see Reference [37]) are run when justified by design changes to the assembly structure. Their aim is to either acquire the experimental data needed for some studies (data for accident analysis) or to globally test the behaviour of an assembly in a flow loop (vibration response, hydraulic compatibility and endurance).

As presented in Chapter 31, the fuel rod is designed to accommodate the in-pile conditions such as exceedingly high internal fission gas pressure, fuel and cladding temperatures, and cladding stresses. Since power ramp rate plays a key role in maintaining fuel integrity during DBC-1 and, DBC-2 and frequent DBC-3, the change rate of linear power has been restricted during plant operation to maintain the integrity of fuel rods.

As recommended in Chapter 21, the fuel rod integrity will be confirmed in-service mainly by REN [NSS], designed for detection, monitoring and sampling of the primary circuit. This system monitors the radioactivity of the primary coolant. Providing the radioactivity of the primary coolant remains below the acceptable limit, it can be concluded there is no loss of fuel rod integrity.

During fuel unloading, the fuel assemblies will be required to undergo an online sipping test whenever abnormal radioactivity levels within the primary coolant are detected. Visual inspection will also be required to examine items including cladding surface and structural integrity of the grid.

5.10 Failed Fuel Management Strategy

The safe operation of the fuel shall always be ensured during all operational states of the nuclear power plant. A systematic approach has been taken for the design of the UK HRP1000 to ensure the safety of the fuel and to manage the failed fuel in case of any fuel failures. This includes the mature design of the fuel to prevent the fuel failure, radioactivity monitoring and sampling analysis to detect any failures, and operational means to react to the failure to reduce the risks to the power plant safety and environment to the level of ALARP/BAT.

5.10.1 Component and Fuel Inspection

An advanced and proven fuel assembly design with rich operating experience, which is AFA 3G™AA fuel system and provided by Framatome, has been adopted to ensure that no failure is occurred during reactor operation. The performance of fuel assembly, fuel rod, RCCA and SCCA are demonstrated in relevant Framatome documentations.

5.10.2 Detection of Fuel Failure

Radioactivity monitoring and sampling analysis are included in the design to detect fuel failure. Several radionuclides are chosen as the indicators of fuel failure, as shown in *Generic Water Chemistry Specification (LCO)*, Reference [38]. Liquid samples from the Reactor Coolant System (RCP [RCS]) are collected by the REN [NSS] and the activity of these nuclides are analysed in the laboratories afterwards, as described in *Design Substantiation Report on Sampling and Monitoring Systems: Nuclear Sampling System*, Reference [39].

5.10.3 Sampling Analysis

In-process monitoring to detect fuel failure during power operation is provided by the Plant Radiation Monitoring System (KRT [PRMS]). A gamma-sensitive detector located on the let-down line of Chemical and Volume Control System (RCV [CVCS]) and a gamma-sensitive detector located on the Nuclear Sampling System (REN [NSS]) line connected to the primary circuit. Two alarm levels are set for each monitoring channel. When the level 1 alarm threshold is exceeded, the operator will closely monitor any increase in the measured value. If the level 2 alarm threshold of both monitoring channels is exceeded, it initiates automatic closure of the containment isolation valves of the RCV [CVCS], the REN [NSS] and the Nuclear Island Vent and Drain System (RPE [VDS]).

5.10.4 Operational Means

Following shutdown (including fall-back to shutdown) of the reactor, refuelling is carried out to remove and replace the fuel. During the refuelling stage, the irradiated fuel assemblies being unloaded from the reactor core are tested by the on-line sipping test, which is installed on the refuelling machine in the reactor building. It is used during unloading to qualitatively detect whether the fuel assembly is failed or not. Meanwhile, the off-line sipping test facility, which is installed in the spent fuel pool, is used to carry out quantitative test of the suspicious fuel assemblies. If it is confirmed that the fuel assembly is failed, it will be transferred to a special failed fuel assembly storage cell for storage. The fuel unloading, inspection and storage operations are presented in Subsection 6.1.1.2 of *Fuel Handling and Storage System Design Manual Chapter 6 System Operation and Maintenance*, Reference [40].

5.11 Source Term

In DBC-1, DBC-2 and frequent DBC-3 there shall be no fuel failures due to design basis transients, therefore the contribution to the source term is from activation of fuel rod and fuel assembly materials and coolant interactions. The source term for this interaction is covered by reactor chemistry in Chapter 21.

However, in DBC-1, DBC-2 and frequent DBC-3, there remains a possibility that there could be random fuel failures resulting from manufacturing defects or operational issues. These fuel failures may or may not be detected during operation (depending on the magnitude of the failure), however the potential releases from the failures are within the capability of Chemical and Volume Control System (RCV [CVCS]) to manage, as described in Chapter 10, with the radiological aspects discussed in Chapter 22.

For operation in in frequent DBC-3 and DBC-4 the fuel and core response is shown in Chapter 12, which provides the contribution to the source term. The source term as a whole is discussed in more detail in Chapter 22.

5.12 Concluding Remarks

This chapter presents the safety and design basis used in the reactor core design of the UK HPR1000. The fuel system design, nuclear design and thermal and hydraulic design have been discussed and the reactor core design description has been provided. All the design bases are derived from the safety functions for the UK HPR1000 as discussed in Chapter 4. Evidence provided demonstrates that these principles are satisfied by the design of the UK HPR1000.

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Appendix 5A Chapter 5 Computer Code Description

There are several computer codes used in Chapter 5, each computer code is as described below:

T-5A-1 Computer Code List

Computer Codes	Sub-chapter
COPERNIC	5.4.3.1
SYSMA	5.4.3.2.1
CARAFE	5.4.3.2.1
DETIGE	5.4.3.2.1
SYSTUS	5.4.3.2.2.1, 5.4.3.2.2.3
SAM	5.4.3.2.2.4
VIBUS	5.4.3.2.3
CASAC	5.4.3.2.5
PCM	5.5.3
POPLAR	5.5.3
LINDEN	5.6.3

a) Fuel Rod Performance Codes

- Fuel Rod Design Code - COPERNIC

COPERNIC is a best-estimate code that predicts the thermal-mechanical behaviour of a single fuel rod in a pressurised water reactor (PWR). It is used to verify that a fuel rod design. For a given reactor, operating conditions and fuel management, meets the design and safety criteria at all times.

It contains a consistent set of physical models for the analysis of PWR fuel in normal and off-normal conditions with regard to thermal, mechanical and fission-gas aspects.

The code has a modular structure and includes a set of stand-alone subprograms, each describing a single physical phenomenon. The subprograms are called by a driver program that controls overall progress of the analysis. Special numerical subroutines control the time step and accelerate the convergence of the iterative processes.

COPERNIC is applicable to calculate PWR fuel rod behaviour with the fuel of UO_2, MOX and UO_2-Gd_2O_3, and the cladding of Zircaloy-4 and $M5_{Framatome}$ alloy.

The validation of COPERNIC relies upon the measurement/prediction comparison relating to the experience feedback built up by Framatome through experimental programs and reactor monitoring programs. The agreement of thermal, mechanical, internal pressure and corrosion parameters of fuel rod proves COPERNIC's capacity to simulate fuel rod thermomechanical behaviour within its validity range.

The detailed information about COPERNIC is given in Reference [41] *Verification and Validation of the fuel system design Software – COPERNIC*.

- Fuel Assembly Thermal Hydraulic Code

  - CARAFE

    The coolant flow through the core induces hydrodynamic forces acting on the fuel assembly in the vertical direction. The fuel assembly buoyancy force is acting in the same direction.

    Both forces are counterbalanced by the fuel assembly weight and the fuel assembly holddown system.

    The vertical hydraulic force is the result of the hydraulic resistance of the fuel assembly components, and the inlet and outlet impulses.

    The computation of hydraulic forces is based on the DELPHYNE methodology. The three principal parts of the method are:

    a core calculation to determine the pressure difference, the density and the flow rate in each assembly. The model takes into account a core inlet flow distribution, the local pressure drop coefficients from the lower core plate to the upper core plate of each fuel assembly and the outlet pressure field. This is performed with the FLICA III-F code.

    a post-processing to determine the hydrodynamic force on each assembly, based on the core calculation. This is performed with the CARAFE code.

    a calculation of the effect of uncertainties on the hydrodynamic force. The impact of each parameter is determined and all uncertainties are statistically combined to get a global uncertainty. This is performed with the CARAFE code.

    The validation of the CARAFE code is assessed in the Reference[42]. The thermal hydraulic conditions encountered in UK HPR1000 reactor are within the range of application of the code CARAFE.

  - DETIGE

    The DETIGE code is a thermal-hydraulic code which enables predicting the flow behaviour in guide tubes.

    The DETIGE code is used to determine:

    1) the bypass flow rate through the guide tube,

    2) the hydraulic forces acting on the cluster's rods,

    3) the thermal behaviour of the coolant in the guide tube.

The DETIGE code solves the continuity, the momentum and the energy equations for a fluid system composed by two parallel channels: one for the core and the other for the guide thimble.

The validation of the DETIGE code is assessed in the Reference [43]. The thermal hydraulic conditions encountered in UK HPR1000 reactor are within the range of application of the code DETIGE.

- Fuel Assembly Mechanical Design Code

The qualification report of the fuel assembly mechanical design codes supporting the UK HPR1000 are the following with the corresponding reference. However, those documents are available for an audit at fuel vender's office.

For fuel assembly mechanical design code CASAC, which is a general purpose structural analysis code designed to solve a wide range of mechanical problems.

This software is dedicated to the study of structures composed of slender parts, concentrated masses and connecting elements featuring linear or non-linear behaviour, the modelling capabilities also include super elements consisting of mass and stiffness matrices expressed at the connected node degrees of freedom.

The CASAC code is appropriate for the UK HPR1000 mechanical design calculations.

T-5A-2 Fuel Assembly Mechanical Design Code

Code	Function	Reference	Title
VIBUS 2 .2	Fuel rod vibratory analysis	FS1-0002849 Rev. 1.0	VIBUS 2.2 Qualification note - Fuel rod vibration analysis
SYSTUS 19	Finite Elements Mechanical analysis code	NEER-F.DC.10296 Rev. H	SYSTUS computer program: verification and physical validation synthesis report
SYSMA 3.8	Analysis of the Fuel Assembly hold-down system (leaf springs)	FFDC01746 Rev. 2.0	SYSMA Version 3.8 – Description and assessment report
SAM 5.6	Analysis of RCCA drop into the dashpot	FFDC03290 Rev. A	SAM Software – Synthesis qualification report
CASAC 5.3	Fuel assembly dynamic response to LOCA and seismic condition	FS1-0034319 Rev 1.0	CASAC 5.3 - Qualification Summary Report

b) PCM

PCM is a nuclear design code package in which PINE and COCO are used in this chapter.
PINE is an advanced Pressurised Water Reactor (PWR) lattice calculation code, and COCO is a three-dimensional (3-D) core calculation code. PINE generates two-group parameter tables for macroscopic cross-sections and the assembly discontinuity factors, which COCO uses to calculate these parameters.

- PINE

PINE performs 2-D lattice calculation for single assembly and multiple assemblies of PWR and generates two-group parameter tables. The parameters include diffusion coe fficients, macroscopic cross-section, surface dependent discontinuity factors, xenon and samarium microscopic densities, flux shape factor for power reconstruction and kinetic parameters.
PINE uses multi-group cross-section databank of IAEA WIMS-D update program.
The physical models of PINE include resonance calculation, transport calculation, leakage correction and burnup calculation.
The equivalence principle is applied to carry out resonance calculation. The Method of Characteristics is applied to perform two-dimensional heterogeneous transport calculation. B1 approximation is applied to take into account the leakage effect. PINE uses two different advanced burnup calculation strategies, which are Linear Rate method and Log Linear Rate method.
Detailed information about PINE is given in Reference [44] *PINE - A Lattice Physics Code: Qualification Report* and Reference [48] *PINE - A Lattice Physics Code: Verification and Validation Report*.

- COCO

COCO is used for PWR nuclear reactor design. The main functions include loading pattern design, critical boron concentration search, evolution calculation, control rod worth assessment, reactivity coefficients calculation, shutdown margin calculation, etc. COCO is also used to perform transient calculations such as Reactivity Induced Accidents.
The solver of COCO is based on Nodal Expansion Method which can handle 2-D and 3-D geometries. The Nodal Expansion Method solver can provide flux distribution in full core and 1/4 core geometries. Furthermore, the Nodal Expansion Method solver is accelerated using Coarse Mesh Finite Difference Method.
The feedback of COCO includes a closed-channel thermal-hydraulic model, which is responsible for moderator temperature and density, and a fuel temperature calculation model.
Both microscopic and macroscopic burnup models are developed. The former focuses on the fission products, minor actinides, etc. The latter handles the intra-node burnup distribution. In macroscopic burnup, nodal surface burnup is calculated to correct cross-

sections.

Detailed information about COCO is given in Reference [45] *COCO - A 3-D Nuclear Design Code: Qualification Report* and Reference [49] *COCO - A 3-D Nuclear Design Code: Verification and Validation Report*.

c) POPLAR

POPLAR is a 1-D neutron diffusion-depletion code. POPLAR is used to perform bite calculation, calibration calculation, xenon depletion calculation, transient xenon calculation, control rod reactivity worth calculation and control rod cross-section modification. Furthermore, POPLAR is used for transient calculation.

POPLAR obtains relevant input of the core from COCO, and the tables of few-group parameters from PINE.

The physical models of POPLAR include cross-section interpolation, 3-D to 1-D conversion, two-group 1-D diffusion solver, leakage correction, thermal feedback and 1-D control rod insertion.

Detailed information about COCO is given in Reference [46] *POPLAR - A 1-D Core Calculation Code: Qualification Report* and Reference [50] *POPLAR - A 1-D Core Calculation Code: Verification and Validation Report*.

d) LINDEN

LINDEN is a sub-channel analysis code which is used for thermal-hydraulic design and safety analysis of reactor core. It calculates the thermal-hydraulic parameters of coolant in reactor core under various conditions, such as pressure, mass velocity, quality and void fraction, etc. Based on the calculated thermal-hydraulic parameters, the DNB of reactor core can be predicted by using a specific CHF correlation.
The flow model in LINDEN is a four-equation model combined with a drift-flux correlation, which takes into account the slip velocity between liquid and vapour phases under two-phase flow. The four-equation model includes a mixture mass equation, a mixture energy equation, a mixture momentum equation and a liquid energy equation. Among them, the liquid energy equation is used to simulate the thermal non-equilibrium of liquid phase during sub-cooled boiling.

The detailed information about LINDEN is given in Reference [47] *LINDEN - A Subchannel Analysis Code: Qualification Report* and Reference [51] *LINDEN - A Subchannel Analysis Code: Verification and Validation Report*.