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Pre-Construction Safety Report

Chapter 5

Reactor Core

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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
CGN	China General Nuclear Power Corporation
CHF	Critical Heat Flux
CRDM	Control Rod Drive Mechanism
DBC	Design Basis Condition
DNB	Departure from Nucleate Boiling
DNBR	Departure from Nucleate Boiling Ratio
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
LOCA	Loss Of Coolant Accident
ONR	Office for Nuclear Regulation (UK)
OPEX	Operating Experience

PCER	Pre-Construction Environmental Report
PCI	Pellet-Cladding Interaction
PCSR	Pre-Construction Safety Report
PWR	Pressurised Water Reactor
RCCA	Rod Cluster Control Assembly
RCV	Chemical and Volume Control System [CVCS]
REN	Nuclear Sampling System [NSS]
RGP	Relevant Good Practice
RPV	Reactor Pressure Vessel
SAP	Safety Assessment Principle (UK)
SCC	Stress Corrosion Cracking
SFIS	Spent Fuel Interim Storage
SFR	Safety Functional Requirement
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. The design information of the handling, transport and storage are provided in Pre-Construction Safety Report (PCSR) Chapter 28. The design information of the SFIS is provided in PCSR Chapter 29. The key design information of the reactor core is presented in this chapter through all the Generic Design Assessment (GDA) steps.

In the previous step of the GDA process, the PCSR, the Pre-Construction Environmental Report (PCER) and the supporting references, which represent the design of the UK version of the Hua-long Pressurised Reactor (UK HPR1000) based on the STEP-12 fuel assembly, have been submitted or scheduled to be submitted to the UK regulators. Regarding the change of fuel types, the AFA 3G^{TM}AA fuel assembly from FRAMATOME is adopted in the design of the UK HPR1000 instead of STEP-12 in the following steps of the GDA assessment. The impact of the fuel change on the whole safety case is assessment in the *Fuel Change Impact Assessment* (see Reference [1]).

The present safety case of Reactor Core is produced based on the design reference version 2.1, as described in the UK HPR1000 Design Reference Report (*UK HPR1000 Design Reference Report,* Reference [2]).

5.2.1 Chapter Route Map

This chapter provides an introduction to the fuel system design, nuclear design and thermal-hydraulic design under Design Basis Conditions (see Chapter 4) in the UK HPR1000 nuclear power plant.

Claim 3: *The design and intended construction and operation of the UK HPR1000 will protect the workers and the public by providing multiple levels of defence to fulfil the fundamental safety functions, reducing the nuclear safety risks to a level 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 the Claim 3.3.1, this chapter developed 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, including fuel*

rod, fuel assembly and 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 for nuclear design derived from the general safety functions are identified. (see Sub-chapter 5.5);*

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

2) ***Argument 3.3.1.SC05.1-A2**: The reactor core specific design principles are identified based on 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, nuclear design and thermal and hydraulic design are identified and implemented based on RGP (ALARP Demonstration Report of PCSR Chapter 05,* Reference [3])*;*

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 methods including design codes and standards have been identified for the system :*

- ***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, nuclear design and thermal and hydraulic design have been identified (see Sub-chapter 5.3 - Codes and Standards);*

2) ***SC05SC05Argument 3.3.1.SC05.2-A2**: The system design has been analysed using the appropriate design methods and meets the design basis requirements (see Sub-chapter 5.4.3 - Design Evaluation, Sub-chapter 5.5.3 - Design Evaluation, Sub-chapter 5.6.3 - Design Evaluation):*

- ***Evidence 3.3.1.SC05.2-A2-E1**: The fuel rod, fuel assembly and RCCA design evaluations demonstrate that the design requirements are fulfilled so as to support 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 method and all the design bases in nuclear design 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 DNBR design basis, fuel*

temperature design basis, core flow design basis and hydrodynamic instability design basis are fulfilled. (see Sub-chapter 5.6)

3) ***Argument 3.3.1.SC05.2-A3****: The system analysis recognises interface requirements and effects from/to interfacing systems (see Sub-chapter 5.2.3 - Interfaces with other parts of PCSR, Sub-chapter 5.4.3 - Design Evaluation, Sub-chapter 5.5.3 - Design Evaluation, Sub-chapter 5.6.3 - Design Evaluation):*

   - ***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 - ALARP):*

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

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

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

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

1) ***Argument 3.3.1.SC05.4-A1****: The system 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 - Commissioning and Testing):*

   - *{*

     }

2) ***Argument 3.3.1.SC05.4-A2****: The system has been designed to take benefit*

from a suite of commissioning tests, to provide assurance of the initial quality of the build (see Sub-chapter 5.8 - Commissioning and Testing).

- ***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 used 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 the conservative peaking factors are used in the core thermal and hydraulic analysis;*
- ***Evidence 3.3.1.SC05.4-A2-E4**: Component and fuel inspection is designed to verify the uncertainty included 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 system 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 - Ageing and EMIT).*

- ***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 fuelling unloading, the visual inspection and the online sipping test (in case of the abnormal radioactivity levels) will be performed.*

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 Source Term

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

- Sub-chapter 5.11 Concluding Remarks

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

- Sub-chapter 5.12 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

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 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 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 related to core and fuel under accidents. The safety functional requirements are derived under DBC-3 and DBC-4. The fuel failure under frequent fault, core thermal response under DBC-2 is described are provided in Chapter 12
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 Vessel (RPV) core shell and its radiation damage mechanism will be discussed in Chapter 17.

PCSR Chapter	Interface
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. Chapter 21 provides the chemistry regime for the integrity of fuel cladding.
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 Storage	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

PCSR Chapter	Interface
	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	Reactor core design is discussed in Chapter 5. Chapter 31 presents the arrangement of 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 nuclear core design considered the design characteristics, UK regulatory expectations, requirements of guidance documents and engineering practice (see Chapter 4.4.7 Codes and Standards).

The following principles are applied during the selection process:

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

b) Adopted the latest version of codes and standards. {

}

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 are applied to other reactor types from previous GDAs.

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

a) The analysis of codes and standards for the fuel system design is based on

function, structure and material characteristics of fuel components. The following list of codes and standards 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 declared 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.

c) The codes and standards for the thermal and hydraulic design are predominantly analysed in accordance with general technical principles, definitions of related glossaries, thermal design bases, hydraulic design bases, determination principles of design limits, pressure drop and 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 should be satisfied in the fuel system design. The fuel rod design covers DBC-1 and DBC-2 while the discussions on DBC-3 and DBC-4 are presented in Chapter 12. The fuel assembly and RCCA mechanical design covers all DBCs.

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

5.4.1 Safety Functional Requirement

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

For DBC-1 and DBC-2, the following SFRs have been identified:

a) The nuclear design, thermal-hydraulic design and fuel system design ensure that the heat produced 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 in DBC-1 and DBC-2 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 (the detailed information about the fuel failure during frequent fault is provided in Chapter 12).

For DBC-3 and DBC-4, the following SFRs have been 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).

5.4.2 Design Description

The following design descriptions are taken from *AFA 3GAA Fuel Assembly Description for HPR1000 Reactor* and *HARMONI RCCA - Description, Functional Requirements and Material Properties* (see Reference [6] and [7]).

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 UK HPR1000 reactor first core 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 reloads 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 the 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 incore 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^{TM}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 will 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).

5.4.3 Design Evaluation

As indicated in Sub-chapter 5.4.1, the fuel system is designed to satisfy the SFRs identified in Chapter 4, which corresponds to fuel rod performance in DBC-1 and DBC-2 (The evidence to support the fuel rod performance in DBC-3 and DBC-4 is provided in Chapter 12) and to fuel assembly and RCCA performance in all DBCs.

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 and DBC-2. Fuel rod design evaluations demonstrate that the design requirements are fulfilled for the fuel rods in order to support Safety Functions H2 and C1{
}.

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).

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:

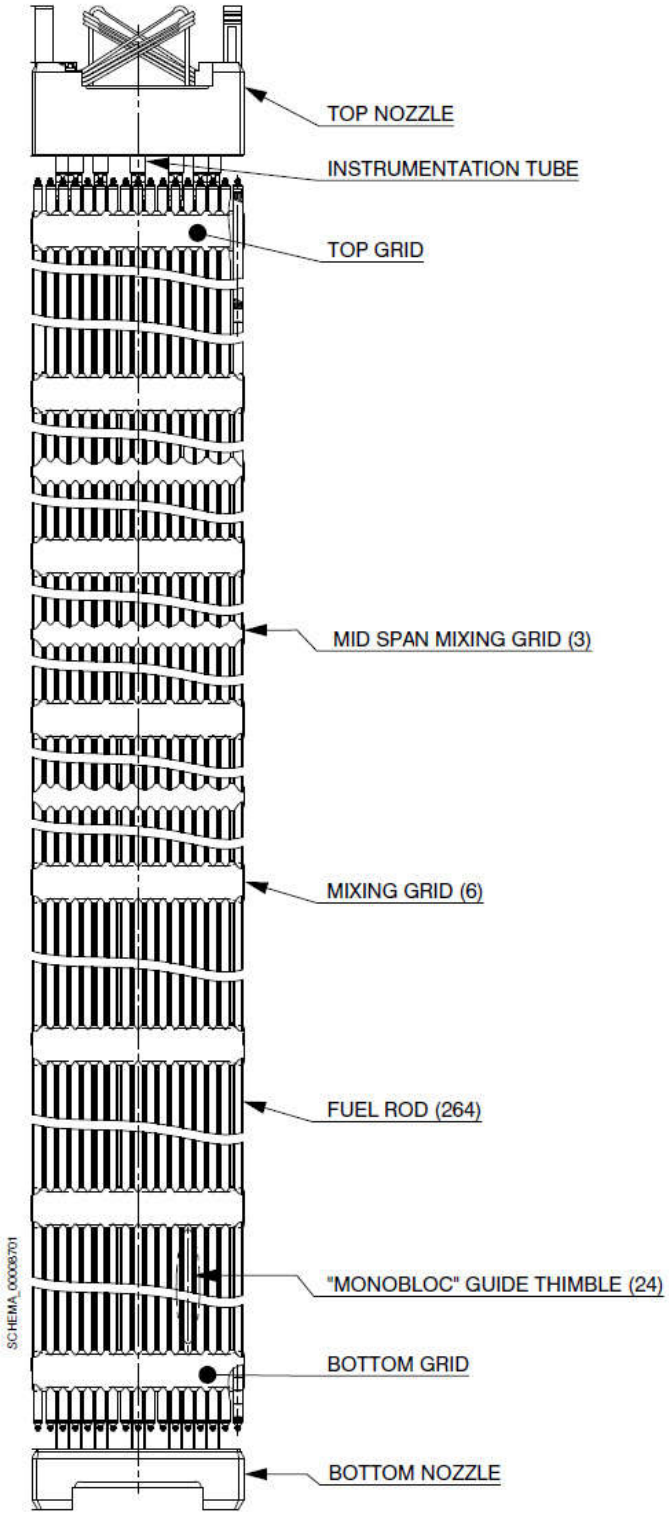
a) Cladding stresses;

b) Thermal stability of absorber materials;

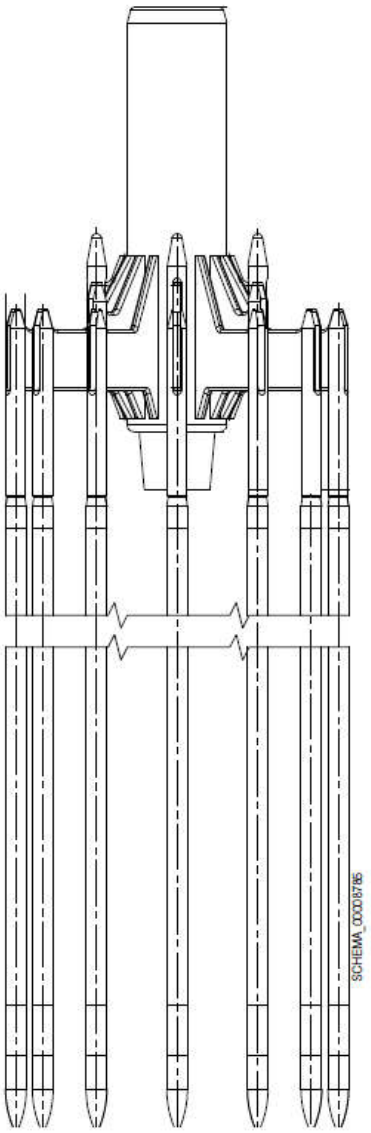
c) Irradiation stability of absorber materials and the cladding; and

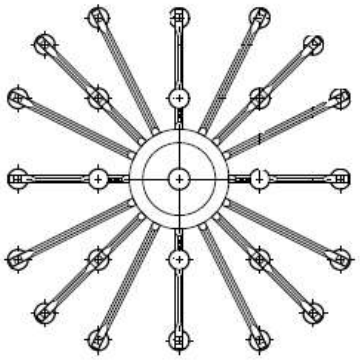
d) Compatibility between RCCA and fuel assembly.

The RCCA evaluations show that the design requirements have been satisfied in order to support Safety Functions R1, R2 and R3{ }.



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, 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 return to DBC-1. Fuel failure does not occur under DBC-1 and DBC-2 (Safety Function C1).

5.5.2 Core Design Description

5.5.2.1 Design Description

5.5.2.1.1 Main Description

The core is composed of 177 fuel assemblies. Under cold conditions, the height of the active core is 365.76 cm and its equivalent diameter is 323 cm giving a height/diameter ratio of 1.13. The main 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. The information of metal reflector is presented in PCSR Chapter 6.

Assemblies with three different levels of ^{235}U enrichment are used in the initial core loading to flatten radial power distribution. Assemblies of the three different ^{235}U enrichments form zones 1, 2 and 3. In the central portion of the core, assemblies of lower enrichment are arranged adjacent to each other to form a chequered pattern. Assemblies with the highest enrichment are arranged at core periphery, encircling the inner channels.

The transition from Cycle 1 to Equilibrium Cycle is expected to take 2 transition cycles and the cycle length is extended from 12 months (Cycle 1) to 18 months (transition cycles and Equilibrium Cycle). During the reloading process, 1/3 to 1/2 of the assemblies will be replaced with fresh assemblies. Figure F-5.5-1 and Figure F-5.5-2 show the loading pattern for Cycle 1 and Equilibrium Cycle. For Equilibrium Cycle, the ^{235}U enrichment of the fresh fuel is 4.45%.

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 the power operation, the burnable absorbers are depleted, thus

positive reactivity is introduced, compensating for the negative reactivity from the fuel depletion and the accumulation of fission products.

In practice, the core reloading pattern, including the number and the placement of fresh fuel assemblies, depends on the target cycle length and power histories of previous cycles.

The fission products are accumulated during fuel depletion, some of which readily absorb neutrons. The depletion of fissile material and the accumulation of fission products are partially offset by the build-up of plutonium, which results from non-fission absorption of neutrons with ^{238}U. Therefore, at BOC, the adequate excess reactivity is available 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 the moderator temperature coefficient becomes less negative with the increase of the soluble boron concentration, the use of burnable absorbers significantly reduces soluble boron concentration to ensure that the moderator temperature coefficient is non-positive, especially at BOC when the soluble boron concentration at the highest level. The depletion rate of the burnable absorber does not cause a problem because the soluble boron is available to compensate for any possible deviation of burnable absorber depletion. Figure F-5.5-3 presents the comparison of core depletion curves with/without burnable absorber rods based on the loading pattern of Cycle 1.

The use of burnable absorber rods provides a favourable radial power distribution. Figure F-5.5-4 shows the layout of the fuel assembly which represents the burnable absorber rod arrangement in a fuel assembly 17×17 array.

5.5.2.1.2 Stability

5.5.2.1.2.1 Introduction of Stability

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

5.5.2.1.2.2 Stability Control and Surveillance

The control of the axial power distribution is achieved by inserting or withdrawing the RCCAs to keep the axial power difference (ΔI) within the 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 allowed, then the operator ensures that the reactor core returns back to Region I (see Figure F-5.5-5). The definition of ΔI is presented in Sub-chapter 5.5.2.2.3. If ΔI exceeds the boundary of the normal operating domain, the power level is automatically reduced.

Xenon induced spatial oscillations are monitored by in-core and ex-core detective

systems and displayed to the operators so as to allow them to detect and subsequently correct any such oscillations. The signals from the in-core and ex-core detectors and partially from the protection system are available for the operators to supervise these oscillations. The loop temperature sensors, pressuriser pressure indication and measured axial offset are provided for the overpower ΔT and overtemperature ΔT protection, which ensures 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.

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.

5.5.2.1.3 Means of Control

5.5.2.1.3.1 Reactivity Control

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

a) Chemical Poison

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

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

2) The transient xenon and samarium poisoning, following power changes or changes in rod cluster control assembly position;

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

4) The 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 RCCAs, including SA, SB, SC, SD.

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

The arrangement of RCCA banks is shown in F-5.5-6. 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.

All shutdown RCCAs are withdrawn before the withdrawal of the control RCCAs. During the withdrawal process from zero to full power, the control RCCAs are withdrawn sequentially. The movement of RCCAs is achieved 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 along with other means of reactivity control and to prevent the moderator temperature coefficient from being positive at power operation. The use of burnable absorber rods reduces the required concentration of soluble poison in the coolant at BOC as described previously. The gadolinium in the burnable absorber rods is depleted at a sufficiently slow rate so that the critical concentration of soluble boron is maintained to ensure the moderator temperature coefficient is non-positive throughout the cycle life as discussed in Sub-chapter 5.5.3.2.

5.5.2.1.3.2 Control of Power Distribution

a) DBC-1

Two grey RCCA banks are inserted or withdrawn along with two black RCCA banks in a fixed overlap to minimise the power distribution perturbations and compensate for the reactivity variation resulting from power change. The positions of the banks are changed only with power level, and the insertion or withdrawal of these banks

result in a power change.

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

The refined reactivity control is achieved by the R bank. The R bank has significant negative reactivity worth which can make a temporary reactivity adjustment during reactivity transients. The movement of the R bank is handled within operation band on the top of core (higher than the insertion limit) to minimise xenon transient effects on axial power shape.

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 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.

5.5.2.2 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 fuel rod to the average linear power density of fuel rod.

Without regard to densification effect and uncertainty,

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

Allowing for the uncertainty,

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

Actually, according to synthetic method, F_Q is calculated as follows:

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

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

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 $F_{\Delta H}^N$ 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

The maximum discharge burnup of the fuel assembly and fuel rod are within the range proven in the fuel assembly and fuel rod performance analyses respectively (Safety Function C1). Meanwhile, the fuel loaded into the core shall provide sufficient excess reactivity throughout the entire cycle length until the target discharge burnup is met.

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 throughout the entire cycle length 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).

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 the *Fuel Management Report* (see Reference [8]).

5.5.3.2 Reactivity Feedback

There are two main effects which provide the feedback to a rapid introduction of positive reactivity: Doppler effect and flux spectrum effect. Doppler effect relates to the resonance absorption effect induced by fuel temperature variation, and flux spectrum effect is caused by the variation of moderator density. These reactivity effects are usually characterised by reactivity coefficients. The use of low enrichment fuel ensures Doppler coefficient remains negative in order to provide a rapid negative reactivity feedback. The negative moderator temperature coefficient provides a slow feedback on the coolant temperature or void fraction variations. Negative moderator temperature coefficient is required at power operation. The use of burnable absorber rods in the core reduces the concentration of soluble boron to prevent moderator temperature coefficient from becoming positive.

Hence, the fuel temperature coefficient is negative. When the core is critical and the coolant is in normal operation temperature, the fuel cycle design ensures that the moderator temperature coefficient is non-positive during the whole power level cycle throughout the entire fuel cycle. These design limits ensure that the core provides negative reactivity feedbacks when the temperature rises. (Safety Functions R1 and R2)

Since the reactivity coefficients change along the fuel cycle, the range of these reactivity coefficients are limited with prescribing the upper and lower design limits, which are used as interface data in safety analysis. The 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 the *Nuclear Design Basis* (see Reference [9]).

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 produces negative reactivity.

Otherwise, the effect of effective fuel temperature variation as a function of core

power is evaluated by the Doppler power coefficient. The integral of the Doppler power coefficient with core power is the Doppler power defect, defined as the Doppler effect contribution to integral reactivity change due to the power rise.

5.5.3.2.2 Moderator Coefficient

The moderator coefficient provides a means for quantifying the reactivity variation due to the change in specific 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 unit 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 which introduces positive reactivity.

Thus, if the concentration of soluble poison 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 to be negative at operating temperature.

The moderator coefficient becomes more negative with the increase of core burnup, resulting from the reduction of concentration of soluble boron.

5.5.3.2.2.2 Moderator Void Coefficient

The moderator void coefficient is defined as the change in reactivity with the change of 1% 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 should 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 burnup in fuel rod are consistent with assumption 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 F_Q^T is enveloped by the LOCA limit along the active core height. The LOCA limit is shown in Figure F-5.5-7 and the evaluated results are given in the *Nuclear Design Basis* (see Reference [9]). The result show that all the transients in the normal operating domain complying with the operation limit for operating regions do not lead to overstepping 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 overpower protection do not lead to fuel melting. The penalty functions of overpower ΔT protection channel is shown in Table T-5.5-4. The overpower ΔT protection channel ensures the linear power density do 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 on all transients under DBC-2,

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

The evaluation results are given in *Nuclear Design Basis* (see Reference [9]).

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 the *Nuclear Design Basis* (see Reference [9]). The 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 the transients which do not trigger the overtemperature protection satisfy the DNBR design limit. The evaluation results are given in the *Nuclear Design Basis* (see Reference [9]).

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

5.5.3.4 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 that the linear power density does not exceed the maximum allowable value 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 reactivity worth of RCCA bank. It is ensured to be lower than the design limit. 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.

5.5.3.5 Shutdown Margin

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 positive reactivity addition 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 should be satisfied throughout the cycle length from BOC to end of cycle (EOC). The design limit of shutdown margin respectively for BOC and EOC are given in Table T-5.5-4. The evaluation results are presented in the *Nuclear Design Basis* (see Reference [9]).

5.5.3.6 Sub-Criticality

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

5.5.3.6.1 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 [9].

5.5.3.6.2 Criticality for Fuel Storage

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

a) $k_{\text{eff}} < 0.95$ for fresh fuel assemblies in storage rack in normal condition;

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

c) $k_{\text{eff}} < 0.95$ for fuel assemblies storage in 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) Fuel assembly array is transversely infinite and is encompassed by selected conservative reflector;

c) The neutron absorber added in structural materials is considered;

d) The soluble boron for neutron absorption in the water is not considered;

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

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

g) The most unfavourable conditions are adopted by sensitivity analysis; and

h) The fuel storage system designs should incorporate sufficient factors of safety to require at least two unlikely, independent, and concurrent changes in process conditions before a criticality accident is possible.

Fuel storages in the new fuel storage rack and in spent fuel pool are introduced in PCSR Chapter 28.6.3, and the interim storage for spent fuel is introduced in PCSR 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 [10]).

5.5.3.7 Vessel Irradiation

Neutrons generated in the reactor core can leak from the active region. When these neutrons with high energy irradiate the 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.

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, 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	
{ }	{ }
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)	1.212
• Cycle 1	1.232
• Equilibrium Cycle	

T-5.5-2 Nuclear Design Objectives and Limits

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
{ }	{ }
Nuclear enthalpy rise hot channel factor (at hot full power), $F^{N}_{\Delta H}$	1.65
Maximal Reactivity insertion rate, pcm/s	55

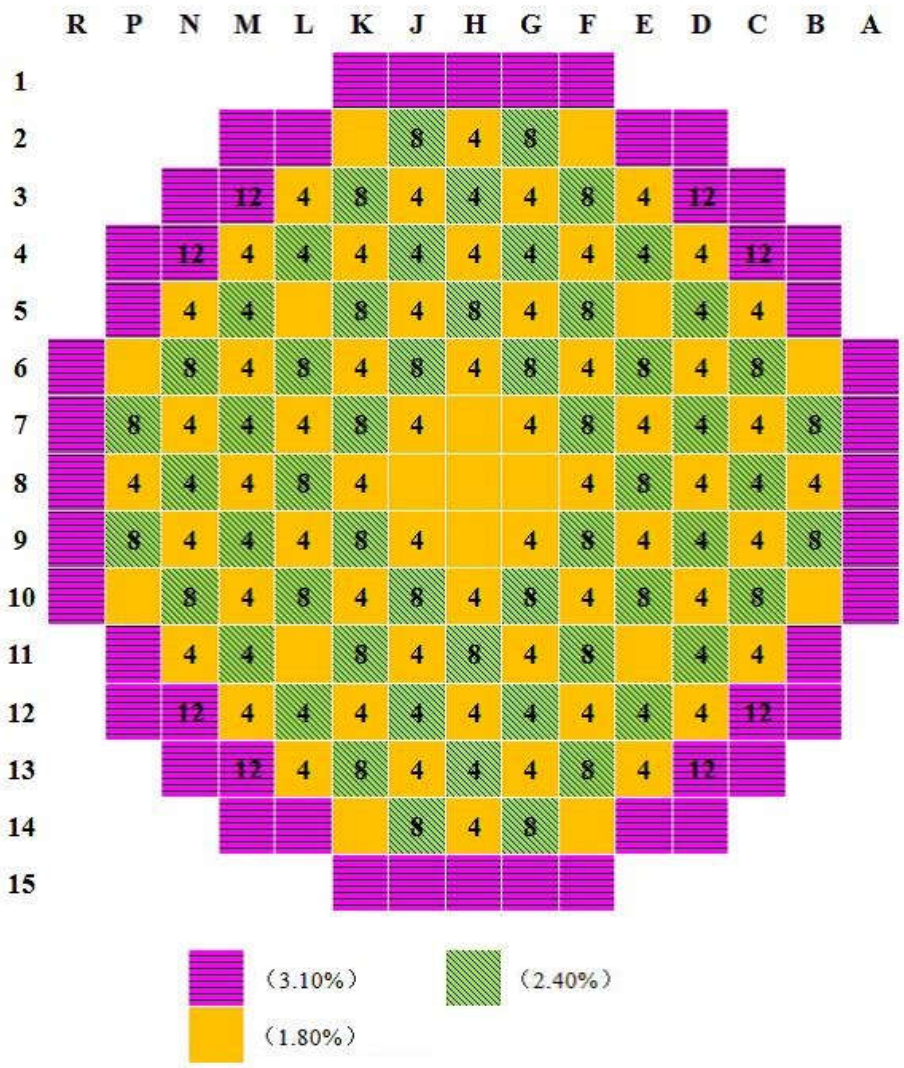
T-5.5-3 Design Limits of Nuclear Design Parameters

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-5.5-8
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)

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

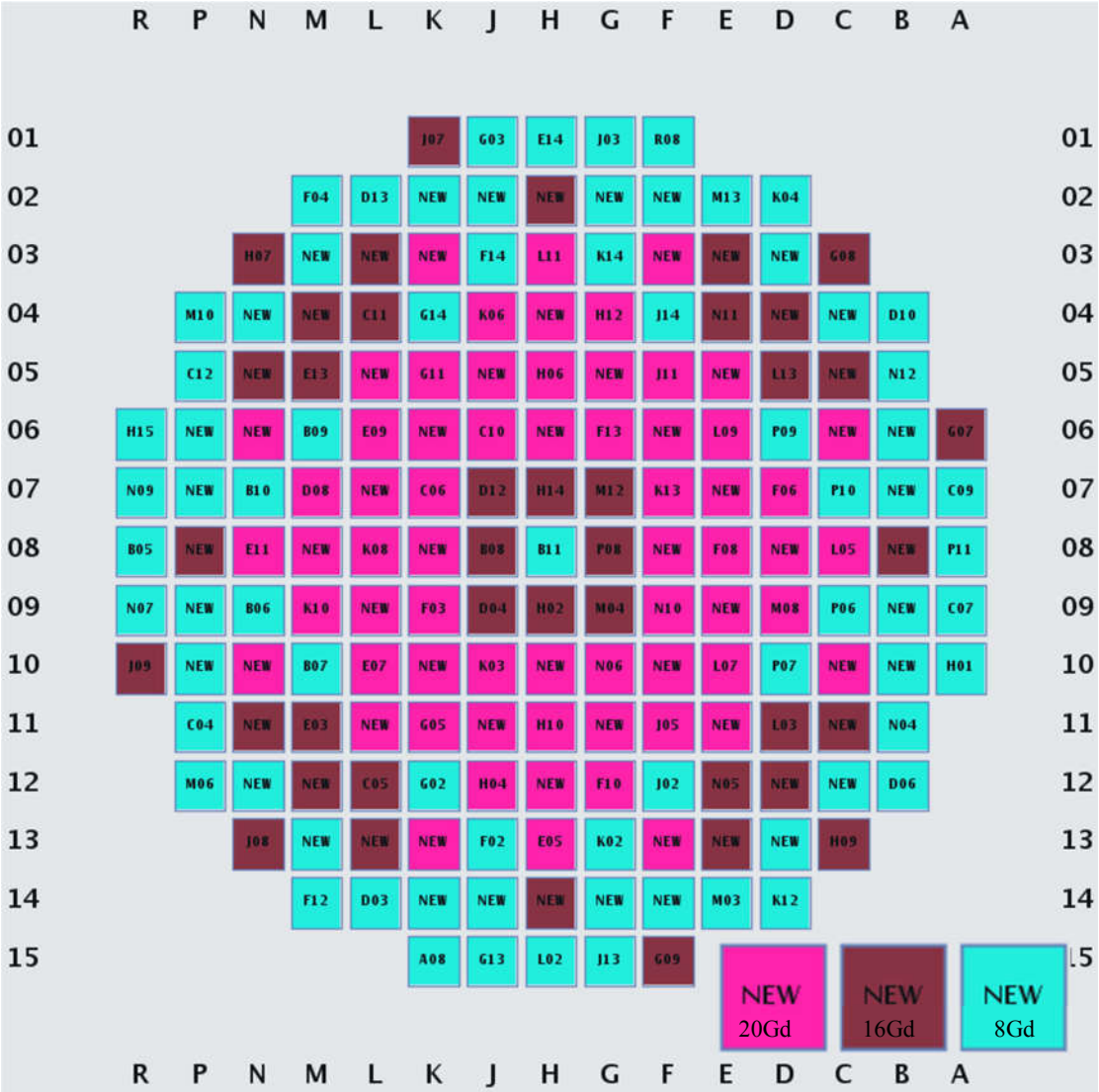
T-5.5-5 Shutdown Margin

	Condition	Limit
Shutdown margin (pcm)	BOC	2000
	EOC	3300



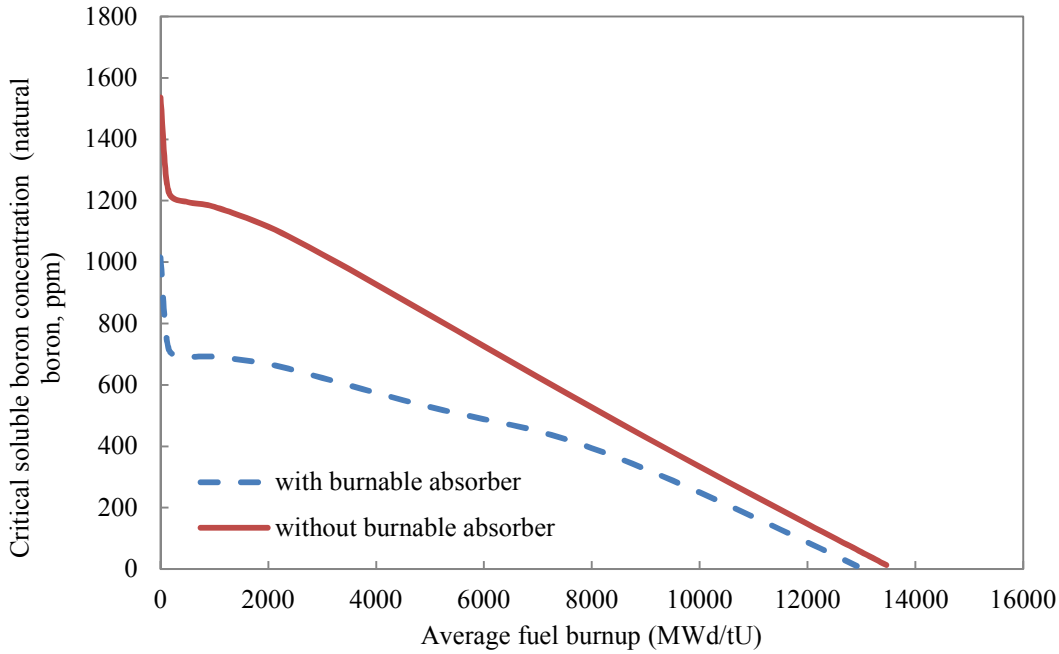
Note: The numbers on the assemblies indicate the number of burnable absorber rods.

F-5.5-1 Loading Pattern of Cycle 1



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

F-5.5-2 Loading Pattern of Equilibrium Cycle



F-5.5-3 Critical Soluble Boron Concentration of Cycle 1 with and without Burnable Absorber

No burnable absorber rods

4 burnable absorber rods

8 burnable absorber rods

12 burnable absorber rods

16 burnable absorber rods

20 burnable absorber rods

UO_2 Fuel Rod

Instrumentation Tube

Guide Thimble

Burnable Absorber Rod

F-5.5-4 Burnable Absorber Rod Layout in Fuel Assemblies

F-5.5-5 Normal Operating Domains (for Safety Analysis)

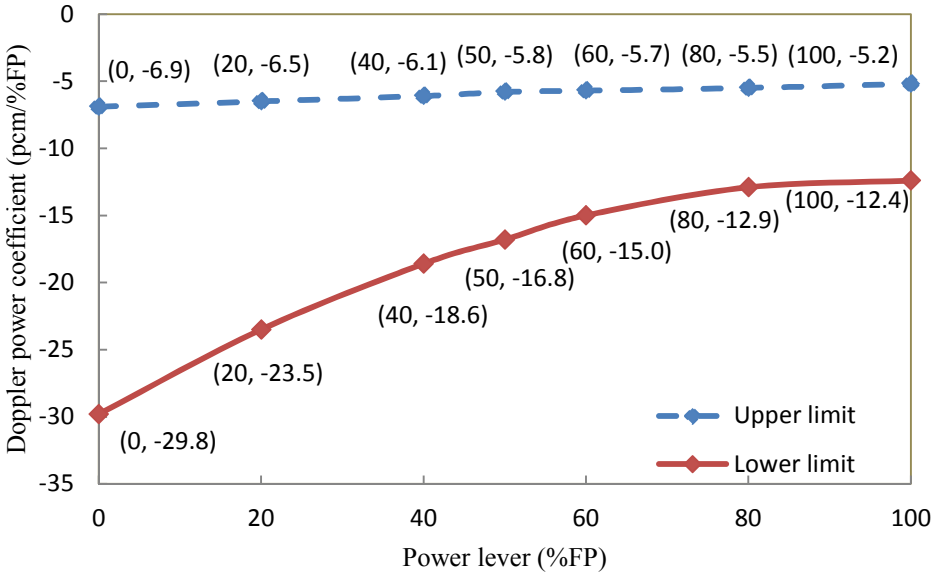
	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A
01															
02						N2		R		N2					
03					SA		SC		SC		SA				
04				N1		G2		G1		G2		N1			
05			SA		SD		SB		SB		SD		SA		
06		N2		G2		R		N1		R		G2		N2	
07			SC		SB		SD		SD		SB		SC		
08		R		G1		N1				N1		G1		R	
09			SC		SB		SD		SD		SB		SC		
10		N2		G2		R		N1		R		G2		N2	
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-5.5-6 Arrangement of RCCA Banks

F-5.5-7 LOCA Limit (DBC-1)



F-5.5-8 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 ensure 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 and DBC-2 (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 or DBC-2; and

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

5.6.2 Design Description

Values of parameters related to fuel temperature and linear power density are presented in Table T-5.6-1 (*NSSS Operating Parameters,* Reference [11]) 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).

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.

5.6.2.1 Departure from Nucleate Boiling Design Basis

There is at least a 95% probability that DNB will 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 Description,* Reference [12]). W3 CHF correlation is used upstream of the first mixing grid of fuel assembly (*UK HPR1000 - W3 CHF Correlation,* Reference [13]).

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

5.6.2.1.1 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 described in *Thermal Hydraulic Design* (see Reference [14]).

5.6.2.1.2 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 the *Thermal Hydraulic Design* (see Reference [14]).

The design limits of the W3 CHF correlation and the deterministic DNBR design limits with W3 CHF correlation are also described in the *Thermal Hydraulic Design* (see Reference [14]).

5.6.2.2 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℃. 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. The melting temperature of uranium dioxide used in design is 2590℃.

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

5.6.2.3 Core Flow Design Basis

The minimum value of thermal design flowrate that will 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.

5.6.2.4 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 Design Evaluation

5.6.3.1 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.

5.6.3.1.1 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 Description,* Reference [12]).

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 [13]). Its application has been expanded to flows in tube bundles with any axial power profile thanks to the addition of the following factors:

a) A cold wall factor, to take into account the presence of non-heated surfaces like guide tubes;

b) A non-uniform flux factor; and

c) A grid performance factor. The factor considered for this analysis corresponds to the one associated to non-mixing grids.

The validity domain of FC2000 CHF correlation is described in Reference [12] *FC2000 CHF Correlation Description*, and the validity domain of W3 CHF correlation is described in Reference [13] *UK HPR1000 - W3 CHF Correlation*.

5.6.3.1.2 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. {

}

5.6.3.1.3 Engineering Hot Channel Factor

5.6.3.1.3.1 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.

5.6.3.1.3.2 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.

5.6.3.1.3.3 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.

5.6.3.1.4 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.

5.6.3.1.4.1 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.

5.6.3.1.4.2 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.

5.6.3.1.4.3 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.

5.6.3.1.4.4 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. 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.

5.6.3.2 Linear Power Density

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

5.6.3.3 Core Hydraulic

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

5.6.3.3.1 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.

5.6.3.3.2 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.

5.6.3.3.3 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.

5.6.3.3.4 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).

5.6.3.4 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.

5.6.3.4.1 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.

5.6.3.4.2 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.

5.6.3.5 Uncertainties

5.6.3.5.1 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.

5.6.3.5.2 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.

5.6.3.5.3 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.

5.6.3.5.4 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.

5.6.3.5.5 Uncertainties in Flowrates

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

5.6.3.5.6 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.

5.6.3.5.7 Uncertainty in Mixing Coefficient

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

5.6.3.6 Summary of Thermal Effects

The reactor protection system ensures that DNB design basis and fuel temperature design basis are met under DBC-2. The relevant transient analysis is described in Chapter 12.

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

The ALARP assessment of fuel and core design contains the compliance with RGP, Safety Assessment Principles (SAPs) and Technical Assessment Guides (TAGs).

RGP is typically defined in the following non-exhaustive list of sources:

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

Besides RGP, SAPs and TAGs of Office for Nuclear Regulation (ONR) related to fuel and core design are also analysed. The following ones are related to nuclear design and thermal-hydraulic design.

a) Safety Assessment Principles for Nuclear Facilities, Revision 0 (2014), ONR

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

- Safety of Nuclear Fuel in Power Reactors, NS-TAST-GD-075 Revision 1 (2017), ONR
- Guidance on the Demonstration of ALARP, NSTAST-GD-005 Revision 9 (2018), ONR

International operating experience (OPEX) from European Pressurised Reactor (EPR), Advanced Passive pressurised water reactor (AP1000) and Advanced Boiling Water Reactor (ABWR) has also been considered in order to optimise the UK HPR1000 design.

With the compliance analysis with RGP, SAPs and TAGs, no gap or risk have been identified in nuclear design or thermal-hydraulic design at this stage. At present, no potential improvement is identified during the safety analysis process. So there is no risk assessment or specific ALARP assessment in fuel and core design. Optioneering process will be performed if any potential improvement is identified in the future (*ALARP Demonstration Report of PCSR Chapter 05,* Reference [3]).

The ALARP of AFA 3G^{TM} AA design is demonstrated by summarising: the experience of AFA 3G^{TM} AA designers and the assessment process applied to the design, the design codes and standards applied in AFA 3G^{TM} AA design and their relationship to international codes, the use of international operational feedback to optimise the design including the ONR request from the UK version of the European Pressurised Reactor (UK EPR) project, the comparison of the AFA 3G^{TM} AA design against

Health and Safety Executive (HSE) SAPs confirm that the SAPs requirements applicable at the fuel design stage have been addressed including the identification of gaps and proposed improvements (*ALARP Demonstration Report of PCSR Chapter 05,* 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 test including mechanical tests and flow loop tests are run when justified by design changes on 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 recommended in Chapter 21, the fuel rod integrity will be confirmed in-service mainly by REN [NSS]. The design of which has been established for detection,

monitoring and sampling of the primary circuit. The results of the sample analysis will confirm that the radioactivity of primary coolant is maintained below the limit, from which it can be concluded that 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 Source Term

In DBC-1 and DBC-2 there should be no fuel failures due to design basis transients, therefore the contribution to the source term will be 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 and DBC-2, 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 Chapters 22.

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

5.11 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.

5.12 References

[1] CGN, Generic Design Assessment (GDA) for UK HPR1000 - Fuel Change Impact Assessment, GHX00100009DRDG03GN, Rev. A, 2019.

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[4] FRAMATOME, UK HPR1000 - Suitability Analysis of Codes and Standards in Fuel Design, GHX42500015SFSL44GN, Rev. 2.0, 2019.

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[14] CGN, Thermal Hydraulic Design, GHX00100004DRRG03GN, Rev. E, 2019.

[15] CGN, PINE - A Lattice Physics Code Qualification Report, CNPRI-GN-F11-17REC010-005, Rev. A, 2019.

[16] CGN, COCO - A 3-D Nuclear Design Code Qualification Report, CNPRI-GN-F11-17REC010-008, Rev. A, 2019.

[17] CGN, POPLAR - A 1-D Core Calculation Code: Qualification Report, CNPRI-GN-F11-17REC010-010, Rev. A, 2019.

[18] CGN, LINDEN - A Subchannel Analysis Code: Qualification Report, CNPRI-GN-F11-17REC010-012, Rev. A, 2019.

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
PCM	5.5.3

Computer Codes	Sub-chapter
POPLAR	5.5.3
LINDEN	5.6.3

a) 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 f 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 alloy.

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 coefficients, 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.
The detailed information about PINE is given in Reference [15] *PINE - A Lattice*

Physics Code Qualification 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.

The detailed information about COCO is given in Reference [16] *COCO - A 3-D Nuclear Design Code Qualification 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.

The detailed information about COCO is given in Reference [17] *POPLAR - A 1-D Core Calculation Code: Qualification 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 [18] *LINDEN - A Subchannel Analysis Code: Qualification Report*.