Revision	Approved by	Number of Pages
000		80
Approval Date	12/11 2018	

General Nuclear System Ltd.

UK HPR1000 GDA Project

Document Reference:	HPR/GDA/PCSR/0005

Title:

Pre-Construction Safety Report
Chapter 5
Reactor Core

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MODIFICATION RECORD

Revision	Section	Page	Modification
000	ALL	ALL	First Issue

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5.1 List of Abbreviations and Acronyms

Ag-In-Cd	Silver-Indium-Cadmium
ALARP	As Low As Reasonably Practicable
AO	Axial Offset
ASME	American Society of Mechanical Engineers
BOC	Beginning Of Cycle
BCX	Beginning of Cycle, equilibrium Xenon
CGN	China General Nuclear Power Corporation
CHF	Critical Heat Flux
CRDM	Control Rod Drive Mechanism
DBA	Design Basis Accident
DBC	Design Basis Condition
DNB	Departure from Nucleate Boiling
DNBR	Departure from Nucleate Boiling Ratio
EOC	End Of Cycle
GDA	Generic Design Assessment
EMIT	Examination, Maintenance, Inspection and Testing
HPR1000	Hua-long Pressurised Reactor
HZP	Hot Zero Power
LOCA	Loss Of Coolant Accident
MTC	Moderator Temperature Coefficient
PCI	Pellet-Cladding Interaction
PWR	Pressurised Water Reactor
REN	Nuclear Sampling System [NSS]
RCCA	Rod Cluster Control Assembly
RCV	Chemical and Volume Control System [CVCS]
RGP	Relevant Good Practice
RPV	Reactor Pressure Vessel

SCC	Stress Corrosion Cracking
SFR	Safety Functional Requirement
SSE	Safety Shutdown Earthquake
UK HPR1000	The UK version of the Hua-long Pressurised Reactor

System codes (XXX) and system abbreviations (YYY) are provided for completeness in the format (XXX [YYY]), e.g. Nuclear Sampling System (REN [NSS]).

5.2 Introduction

The purpose of this chapter is to introduce the fuel system design, nuclear design and thermal-hydraulic design. The key design information will be presented in this chapter through all GDA steps.

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 version of the Hua-long Pressurised Reactor (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 1: The safety functional requirements (SFRs) or design bases have been derived for the system:**

   1) Argument 1.1: The system design bases have been derived from the safety analysis in accordance with the general design and safety principles (see Sub-chapter 5.4 - Fuel System Design, Sub-chapter 5.5 - Nuclear Design, Sub-chapter 5.6 - Thermal and Hydraulic Design).

   2) Argument 1.2: The system specific design principles are identified based on relevant good practice (RGP) (see Sub-chapter 5.7 - ALARP);

b) **Sub-Claim 2: The system design satisfies the SFRs or design bases:**

   1) Argument 2.1: Appropriate design methods have been identified for the

system including design codes and standards (see Sub-chapter 5.3 - Codes and Standards);

2) Argument 2.2: The system 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);

3) Argument 2.3: The system analysis recognises interface requirements and effects from/to interfacing systems (see Sub-chapter 5.2.2.3 - 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).

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

1) Argument 3.1: The system meets the requirements of the relevant design principles (generic and system specific) and therefore of relevant good practice (see Sub-chapter 5.7 - ALARP);

2) Argument 3.2: Design improvements have been considered and any reasonably practicable changes implemented (see Sub-chapter 5.7 - ALARP).

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

1) Argument 4.1: 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 4.2: 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).

e) **Sub-Claim 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 5.1: An initial 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).

5.2.2 Chapter Structure

The structure of Chapter 5 is as follows.

a) Sub-chapter 5.1 List of Abbreviations and Acronyms

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

b) Sub-chapter 5.2 Introduction

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

c) 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.

d) Sub-chapter 5.4 Fuel System Design

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

e) Sub-chapter 5.5 Nuclear Design

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

f) Sub-chapter 5.6 Thermal and Hydraulic Design

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

g) Sub-chapter 5.7 ALARP Assessment

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

h) Sub-chapter 5.8 Commissioning and Testing

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

i) Sub-chapter 5.9 Ageing and EMIT

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

j) Sub-chapter 5.10 Source Term

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

k) Sub-chapter 5.11 Concluding Remarks

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

l) Sub-chapter 5.12 References

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

m) Appendix 5A The Computer Codes Description in Chapter 5

This appendix introduces the computer codes used in this chapter.

5.2.3 Interfaces with Other Chapters

The interfaces with other Chapters are listed in the following table.

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

PCSR Chapter	Interface
Chapter 1 Introduction	Chapter 1 provides the Fundamental Objective, Level 1 Claims and Level 2 Claims. Chapter 5 provides chapter claims and arguments to support the high level claims presented in Chapter 1.
Chapter 2 General Plant Description	Chapter 2 provides a brief introduction to the fuel and core.
Chapter 4 General Safety and Design Principles	Provides the definition of DBCs and safety functions related to Chapter 5.
Chapter 6 Reactor Coolant System	Control rod drive mechanism contributes to effective reactivity control using of the Rod Cluster Control Assembly (RCCA).
Chapter 10 Auxiliary Systems	Chapter 10 provides detailed design information of the RCV [CVCS].
Chapter 12 Design Basis Condition	The safety functional requirements are derived under DBC-3 and DBC-4. The core thermal response under DBC-2 is described.
Chapter 17 Structural Integrity	The relevant descriptions of irradiation surveillance requirements for the RPV core shell and its radiation damage mechanism will be discussed in Chapter 17.
Chapter 18 External Hazards	Chapter 18 provides relevant external hazards considered in UK HPR1000 operation. It includes external hazards such as earthquakes, which are taken into account during in fuel system design evaluation.
Chapter 20 MSQA and Safety	The organisational arrangements and quality assurance arrangements set out in Chapter 20 are

PCSR Chapter	Interface
Case Management	implemented in the design process and production of Chapter 5.
Chapter 21 Reactor Chemistry	Chapter 21 provides the chemistry regime for the integrity of fuel cladding.
Chapter 22 Radiological Protection	Chapter 22 describes the definition of radioactive sources for UK HPR1000 and covers the various source terms present during normal operation.
Chapter 28 Fuel Route and Storage	Chapter 28 provides the introduction of fuel storage in the storage rack and spent fuel pool.
Chapter 29 Interim Storage for Spent Fuel	Chapter 29 provides the introduction of spent fuel interim storage, including the spent fuel management strategy, general requirements, optioneering considerations, etc.
Chapter 31 Operational Management	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 Chapter 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 nuclear core design principles are analysed in accordance with the requirements declared from related codes and standards, which provides clarifications of the definitions of technical glossaries, nuclear design bases and the methods, conditions and acceptance criteria for core physic tests.

The codes and standards for reactor core safety analysis (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 analysis of codes and standards for the fuel system design are based on function, structure and material characteristics of fuel components.

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 approved 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 approved reactor types.

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

a) Fuel system design

[1] ASME, Boiler and Pressure Vessel Code, NB/NG/Appendix F, 2007 edition.

[2] AFCEN, RCC-C Design and Construction Rules for Fuel Assemblies of PWR Nuclear Power Plants, 2015 edition.

[3] ANS, American National Standard for Light Water Reactors Fuel Assembly Mechanical Design and Evaluation, ANSI/ANS 57.5-1996 R2006

b) Nuclear design

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

[2] IAEA, Design of Fuel Handling and Storage Systems in Nuclear Power Plants, No.NS-G-1.4, 2003 edition.

c) Thermal and Hydraulic design

[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 Rod Cluster Control Assembly (RCCA) mechanical design covers all DBCs.

A CGN in-house STEP-12 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 ensures the control of core reactivity, the nuclear chain reaction could be stopped, and the reactor could 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. And the justification of these SFRs shall take into account the maximum power changes which the fuel assembly and RCCA experience. For instance, this is of particular relevance to the commissioning and testing requirements for the stages of start-up.

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

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

5.4.2.1 Fuel Assembly

The assembly, which is an orthogonal structure with a 17×17 square array, consists of 264 fuel rods and the skeleton. The skeleton is composed of 24 guide thimbles, 1 instrumentation tube, 1 top nozzle, 1 bottom nozzle, 6 mixing grids, 1 top end grid, 1 bottom end grid and 3 mid span flow mixers.

The fuel assembly is mainly made of stainless steel, nickel-based alloy and zirconium alloy.

5.4.2.1.1 Fuel Rod

The fuel rod consists of a stack of sintered uranium dioxide pellets with or without burnable absorbers (gadolinium) encapsulated in a closed CZ alloy tube that is plugged and welded.

The pellet is a right cylinder with chamfers at the edges and dishes at the centres on the upper and lower faces. The pellet is made from a slightly enriched UO_2 powder or UO_2-Gd_2O_3 powder which is compacted through cold pressing and then sintered to reach the required density.

In order to avoid the excessive stress of the cladding and weld, an upper plenum and the pellet-cladding gap are introduced in fuel rod. They are designed to contain fission gases released and to accommodate the thermal expansions of pellet during irradiation.

A helical stainless steel plenum spring is loaded between the upper face of the last pellet and the lower part of the top end plug. It is designed to prevent any slippage of the stack of pellets during handling and shipping before in-reactor loading.

The fuel rod is helium-pressurised to reduce the compressive stress and creep caused by coolant pressure during operation. With good thermal conductivity, helium gas contributes to transfer the heat produced from pellet to outside of the fuel rod.

The fuel rods are supported at intervals along the length by the grid assemblies that maintain the lateral spacing between the rods. The fuel rods are loaded into the fuel assembly structure and there is clearance between the fuel rod ends and top/bottom nozzles to accommodate fuel rod expansion.

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 core components. It is also used during fuel assembly handling.

The top nozzle assembly consists of a welded square structure and four sets of spring packs, the weld square structure comprises an adaptor plate and a top plate, which is made of type 304L stainless steel. The spring packs are held in place by attachment screws, which are constructed of alloy 718.

The adapter plate is provided with an array of holes and slots to allow the coolant to flow upward, which 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 plate has a large square opening in the centre to permit access for the core components, in-core instrumentation and tools for handling the assembly in the shop or on site.

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 spring packs. The pads protect the spring leaf ends and attachment screws during handling operations.

The hold-down spring arrangements ensure that, in the very unlikely case of spring failure, the failed spring 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 upward hydraulic loads. In normal flow conditions, the assembly is kept in contact with the lower core plate.

5.4.2.1.3 Bottom Nozzle

The bottom nozzle is the bottom structural member of the fuel assembly, which consists of a ribbed structure fitted with 4 legs and topped with a thick anti-debris device. It protects the fuel assembly from debris and channels the coolant through the rod bundle.

The ribbed structure is a type 304L stainless steel component designed to accommodate mechanical loads. The highest peripheral rib forms a chamfered "skirt" intended to stiffen the nozzle. The ribbed structure acts as housing for the guide thimble attachment screws. It supports the anti-debris device and provides an outer enclosure compatible with handling requirements.

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

The guide thimbles are firmly attached to the ribbed plate by socket head screws. The instrumentation tube of each assembly is constrained in a machined recess.

5.4.2.1.4 Grid

Along with the guide thimbles, the top and bottom nozzle, the grids make up the skeleton of the fuel assembly. The grids ensure that the fuel rods are equally spaced relative to each other throughout the fuel assembly lifetime. Each grid is an array of slotted grid straps that are interconnected and welded at the intersections, the straps are made of CZ alloy. The grid assembly of various inner straps and four outer straps forms a square array of 289 cells, of which 24 are used for the guide thimbles and 1 for the instrumentation tube. The remaining 264 grid cells are loaded with fuel rods.

There are two types of grids in each fuel assembly:

- Structural grids;

- Mid span flow mixers.

The structural grids consist of end grids and mixing grids. The support system of structural grids provides both lateral and vertical support for the fuel rods and accommodates thermal expansion, irradiation growth and creep of the fuel rods. The grid dimples and springs position each fuel rod in its grid cell. The grid cell restraining force is high enough to prevent damage during shipping and handling and to preclude grid-to-rod fretting wear during operation. The mixing grids and mid span flow mixers have mixing vanes on the internal grid straps to promote mixing of the coolant, so heat is taken away efficiently by the reactor coolant.

5.4.2.1.5 Guide Thimble

24 guide thimbles, made of CZ alloy, provide channels for the control rods insertion, as well as the thimble plug rods and neutron source rods. The type of core component depends on the position of the particular fuel assembly in the core.

The upper part of guide thimble forms an annular clearance between the RCCA rodlet outer diameter and guide thimble inner diameter, which is designed to enable rapid RCCA insertion during a reactor trip (scram).

The guide thimbles accommodate coolant flow. When the control rod comes to the entrance of the lower part of guide thimble (or the dashpot), the motion of the control rod is slowed down to its travel limit.

Flow holes are located above the dashpot to avoid stagnation of coolant flow during 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 lower thimble part and drilled with a hole threaded for the connection to the bottom nozzle. The plug is also used to centre the guide thimble in the penetration of the nozzle plate.

5.4.2.1.6 Instrumentation Tube

The instrumentation tube is located in the centre and provides a channel for insertion of an in-core neutron detector, which is made of CZ alloy and used to monitor the fission process during normal operation. The instrumentation tube is fixed in position between the top and bottom nozzle, which is attached to the grids by spot weld; however, it is only constrained at the centre of top and bottom nozzles. This tube exhibits a constant thickness and inner diameter throughout its length, which are equal to those of the upper section of the guide thimble.

5.4.2.2 Rod Cluster Control Assembly

The RCCA includes black and grey RCCAs based on the number of absorber rods, there are 24 absorber rods for a black RCCA, while only 8 absorber rods and 16 stainless steel rods for a grey RCCA. As illustrated in F-5.4-2, each RCCA comprises of a group of individual neutron absorber rods fastened at the top end to a cylindrical spider assembly. The absorber material used in control rods is silver-indium-cadmium (Ag-In-Cd) alloy, which is in the form of solid bars encapsulated in stainless steel tubes. A helical spring is used to constrain the motion of absorber bars axially during the shipping and stepping modes. A bullet-like bottom end plug is designed to reduce the hydraulic load during reactor trip and to guide the absorber rods smoothly into the dashpot section of fuel assembly guide thimbles.

The RCCA is used for reactor shutdown and reactivity control by insertion and withdrawal of the absorber rods, this is realised by compensating for core reactivity changes during normal operation varying with changes in power, coolant temperature.

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. Requirements of the fuel system are met through robust design, in conjunction with analytical models and experimental data collected from either test programmes or commercial power plants, as indicated in the support documents *Fuel Rod Design* (see Reference [1]), *PCI Thermal-Mechanical Analysis* (see Reference [2]) and *Fuel Assembly Mechanical Design Report* (see Reference [3]).

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;

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. It should be noted that the vibration and fretting wear is analysed in Sub-chapter 5.4.3.2.3 while the irradiation growth of fuel rods is analysed in Sub-chapter 5.4.3.2.5. All following points in fuel rod design evaluation are taken from Reference [1] and [2], and emphasised as demonstrating that the design requirements are fulfilled for the fuel rods so as to support Safety Functions H2 and C1 listed in Sub-chapter 5.4.1.

5.4.3.1.1 Cladding Free Standing (Safety Function C1)

At the beginning of life, the maximum coolant pressure shall neither lead to the collapse nor to the plastic strain of the cladding. Geometrical and mechanical characteristics of the cladding tube and helium backfill contribute to allowing the fuel rod design to meet this criterion.

5.4.3.1.2 Cladding Collapse (Safety Function C1)

At the level of axial gaps, which could be formed within the fuel column, the combined effects of the differential pressure across the cladding wall and of the cladding creep, could lead to cladding collapse (the increase of ovality results in the cladding circumferential buckling). The fuel rod design, in particular the fuel rod pressurisation and by using of stable fuel during irradiation, avoids any risk of cladding collapse. The calculation is also performed to confirm that there is no risk of creep collapse.

5.4.3.1.3 Cladding Strain (Safety Function C1)

Following the closure of the pellet to cladding diametric gap, fuel swelling and thermal expansion and the interaction between pellet and cladding induce cladding strains along circumferential direction. Criteria on the uniform circumferential strain are defined in order to avoid strain type fuel failure. The analysis shows that the maximum cladding permanent strain is estimated to be { } and verified below the limit in DBC-1, the maximum total strain induced by DBC-2 with uncertainties is { } and remains below the limit in DBC-2.

5.4.3.1.4 Cladding Stress (Safety Function C1)

During irradiation in the core, the fuel rod cladding is subjected to high stresses due to the differential expansion of the pellet and the cladding. Criteria on the cladding stress are made to maintain the integrity of cladding. For DBC-1, the analysis which considers the irradiation, temperature and corrosion effects shows that the volume average effective stress of cladding remains below the yield strength of CZ alloy. For DBC-2, the PCI criterion as shown in Sub-chapter 5.4.3.1.9 is applied to preclude the fuel failure induced by excessive cladding stress.

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

Oxidation and hydriding are directly related to fuel performance during operation. Oxidation degrades the cladding thermal conductivity while the hydriding leads to a decrease in the cladding ductility and impact toughness. According to the results of upper-bound analysis, the maximum cladding corrosion thickness is observed at the end of life with the upper-bound value of { } and its corresponding hydrogen content is { }. It is confirmed to be lower than the required limit.

5.4.3.1.6 Fuel Temperature (Safety Functions H2 and C1)

The maximum pellet temperature shall remain lower than the fuel melting point. The aim of this criterion is to prevent fuel melt conditions, which could cause volume variation due to phase change (and dispersion of fuel particles), resulting in severe duty on the cladding. Taking into account the variation of fuel melting point with burn-up, the analysis in DBC-2 shows that a margin of { } exists between the maximum fuel temperature obtained and the melting point calculated.

5.4.3.1.7 Fuel Rod Internal Pressure (Safety Function C1)

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

5.4.3.1.8 Cladding Fatigue (Safety Function C1)

The reactor operating conditions lead to alternate loadings, which eventually impose cladding stress cycling and fatigue. For the base-load mode of operation, it is not necessary to verify the fatigue criterion since the number of power and temperature cycles is very limited. For "12-3-6-3" daily load follow, the maximum cumulated fatigue damage factor remains below 1. Hence, there is no risk of fuel failure induced by cladding fatigue during irradiation.

5.4.3.1.9 Pellet-Cladding Interaction and Stress Corrosion Cracking (Safety Function C1)

During an overpower transient, the combined effects of fuel pellet expansion and the presence of corrosive fission product in the gap, such as iodine, could lead to the PCI-SCC of cladding. The risk of fuel failure induced by PCI-SCC is estimated for the DBC-2 event and the results show that the cladding remains at an adequate PCI margin based on the technical limit of CZ alloy.

5.4.3.1.10 Fuel Rod Design Summary

As shown above, all the fuel rod design criteria are met with margins taking into account the operating conditions. The design of the fuel rod therefore maintains its structural integrity and its capability to transfer heat into the coolant (Safety Functions H2 and C1) in DBC-1 and DBC-2. The evaluation for fuel rod behaviour in DBC-3 and DBC-4 is performed in Chapter 12.

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

5.4.3.2 Fuel Assembly

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

a) Fuel handling and loading;

b) Power variations;

c) Temperature gradients;

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

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

f) Vibration and fretting induced by coolant flow;

g) Creep deformation;

h) External events such as earthquakes;

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

Considering all the mechanical stress caused by the phenomenon shown above, each component was evaluated using specified design loads, defined as the most conservative load in each DBC; the design criteria are provided to preclude fuel assembly damage during all design basis conditions. The following points, taken from Reference [3], are emphasised as demonstrating that the design requirements are fulfilled for the fuel assemblies so as to support Safety Functions R1, R2, R3, C1 and H2.

5.4.3.2.1 Fuel Assembly Lift-off (Safety Functions R1 and R3)

The hold-down assembly is designed to prevent fuel assembly lift-off in DBC-1 and DBC-2.

It is shown that the hold-down force produced by leaf-springs could withstand the hydraulic lift force; the design requirements are fulfilled for the hold-down system so as to ensure the fuel assemblies remain seated on the lower core plate.

The evaluation estimates that the minimum contact force between the fuel assembly bottom nozzle and lower core plate is approximately { } in normal operation. Therefore, the hold-down spring force of a fuel assembly could withstand the hydraulic force, even when considering situational uncertainties.

5.4.3.2.2 Fuel Assembly Structure Components Integrity

5.4.3.2.2.1 Top Nozzle (Safety Functions R1 and H2)

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

The mechanical design of the top nozzle is evaluated by finite element analysis, the results show that the membrane stresses and membrane plus bending stresses of adaptor plate ligaments is { } much lower than design limits { }

5.4.3.2.2.2 Connection (Safety Functions R1 and H2)

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

Mechanical stresses and loads are calculated considering the design loads in DBC-1 and DBC-2. Mechanical strength is evaluated to show the design loads on this connection stay within the allowable values determined experimentally.

The evaluation shows that connection within the fuel assembly will maintain mechanical integrity during operation.

For bottom thread screw connection, the maximum calculated stress is approximately { } while the stress design limit is { }, employing a { } margin during operation.

For the top QD connection, the maximum load is about { } in the handing and shipping condition, employing a larger than { } margin.

5.4.3.2.2.3 Bottom Nozzle (Safety Functions R1 and H2)

{

}

The mechanical design justification of the bottom nozzle is also evaluated through finite element analysis, showing that the membrane stresses and membrane plus bending stresses of adaptor plate ligaments is { } and { } respectively, which is below design limit of { } and { }.

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

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

The evaluated membrane stress, the membrane plus bending stress, is within the design criteria { } The stress design limit defined by ASME code section III { }, and so there is a { } margin.

5.4.3.2.2.5 Grid (Safety Functions R1 and H2)

The main functions of grids are to support the fuel rods and to promote mixing of coolant. The mechanical design justification of the grid aims to ensure that the grids retain mechanical integrity and do not experience any significant deformation at the design loads, and that the fuel rods are restrained into the grid to satisfy the design criteria. The restraint provided by the grids contributes to the vibrational responses of the fuel rods. The grids' behaviour is investigated under DBA event in order to ensure that it complies with the safety criteria.

The results show that the fuel rods can be supported axially. The requirements of grids integrity are also satisfied during shipping, LOCA and SSE events.

5.4.3.2.3 Grid to Rod Fretting Wear (Safety Function C1)

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

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

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

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

5.4.3.2.4 Coolability and Insertion of RCCAs in DBC-4 (Safety Functions R2 and H2)

The fuel assemblies are designed to withstand loads from LOCA and SSE events without loss of capability to perform its safety functions. That means the overall behaviour of the fuel assemblies shall be evaluated, with the aim of ensuring that it is possible to shut down the reactor, through the insertion of a sufficient number of RCCAs.

The calculation is performed to justify that the deformation of fuel assemblies will not affect the insertion of RCCAs and that the fuel assembly retains its rod-bundle geometry with adequate coolant channels to permit removal of residual heat. The maximum impact force on mixing grids is { }, which is much lower than the structural grid buckling strength, leaving larger than { } margin. The fuel assembly can, therefore, meet the criterion in DBC-4.

5.4.3.2.5 Rod Growth (Safety Function C1)

The gaps between fuel rods and nozzles must not be closed in order to prevent interference, which could result in fuel failure. The design evaluation addresses irradiation creep and growth of fuel rods and fuel assemblies.

The criterion is met, as proven by the evidence demonstrating that maximum axial elongation is less than clearances between rods and nozzles, taking into account manufacturing uncertainties, maximum rod growth and minimum fuel assembly growth induced by irradiation.

5.4.3.2.6 Fuel Assembly Mechanical Design Summary

From all above analyses, it is concluded that, fuel assembly performance satisfies all the mechanical strength criteria with margins to support Safety Functions R1, R2, R3, C1 and H2.

During fuel assembly mechanical design, DBC-3 is not evaluated explicitly since the loads in DBC-3 are less conservative than the loads in DBC-4. LOCA and SSE combination is regarded as the most bounding and conservative condition and it is proven that the fuel assembly could maintain integrity in DBC-4.

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;

d) Compatibility between RCCA and fuel assembly.

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

5.4.3.3.1 Cladding Stresses (Safety Functions R1, R2 and R3)

{

}

The results show that in DBCs, the cladding stresses are lower than the stress limit with { }

5.4.3.3.2 Thermal Stability of Absorber Materials (Safety Functions R1, R2 and R3)

The maximum absorber temperature shall be lower than the melting point. The aim of this criterion is to prevent the absorber melting, which would result in failure.

Maximum absorber temperatures in DBC-1 and DBC-2 are calculated and the analysis shows that the maximum absorber temperature is much lower than the melting point. A large margin of more than { } exists between the maximum temperature observed and the melting point.

5.4.3.3.3 Irradiation Stability of Absorber Materials and the Cladding (Safety Functions R1, R2 and R3)

The absorber shall be held upright to enable it to control reactivity and to maintain its integrity within the cladding. The gap between absorber and cladding gradually decreases during the lifetime. The creep and swelling of an absorber resulting from irradiation may result in cladding failure and RCCA seizing within the guide thimble or the control rod guide assembly during a reactor trip.

The operational experience shows that no cladding failure or RCCA seizing was detected during the whole lifetime, proving the irradiation stability of absorber materials and the cladding.

5.4.3.3.4 Compatibility between RCCA and Fuel Assembly (Safety Functions R1, R2 and R3)

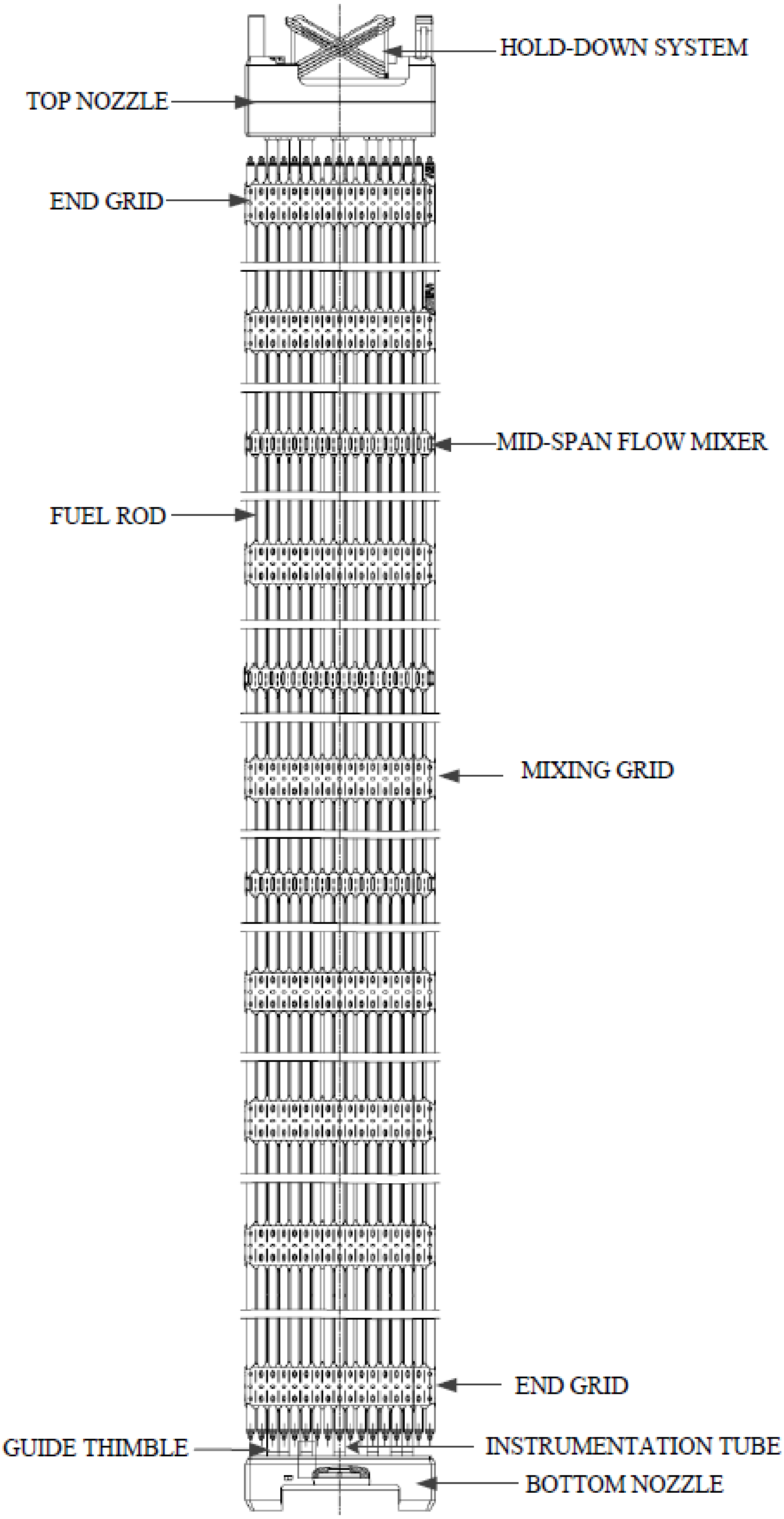
The RCCA shall be compatible with the fuel assembly when inserted. The compatibility evaluation ensures the top of the fissile column is covered by an absorber, and the bottom end of RCCA shall not interfere with fuel assembly guide thimble shoulder screw.

Calculation results indicate that the top of the fissile column is covered during whole assembly lifetime by an absorber pellet in DBC-1 and DBC-2, and a gap { } remains present between the bottom end of the RCCA and guide thimble shoulder screw shoulder. In order to maintain integrity, the stress experienced by the spider assembly under insertion shall be less than the required stress limits. During a drop, the maximum impact force between spider and top nozzle is less than the spider spring design limits { }

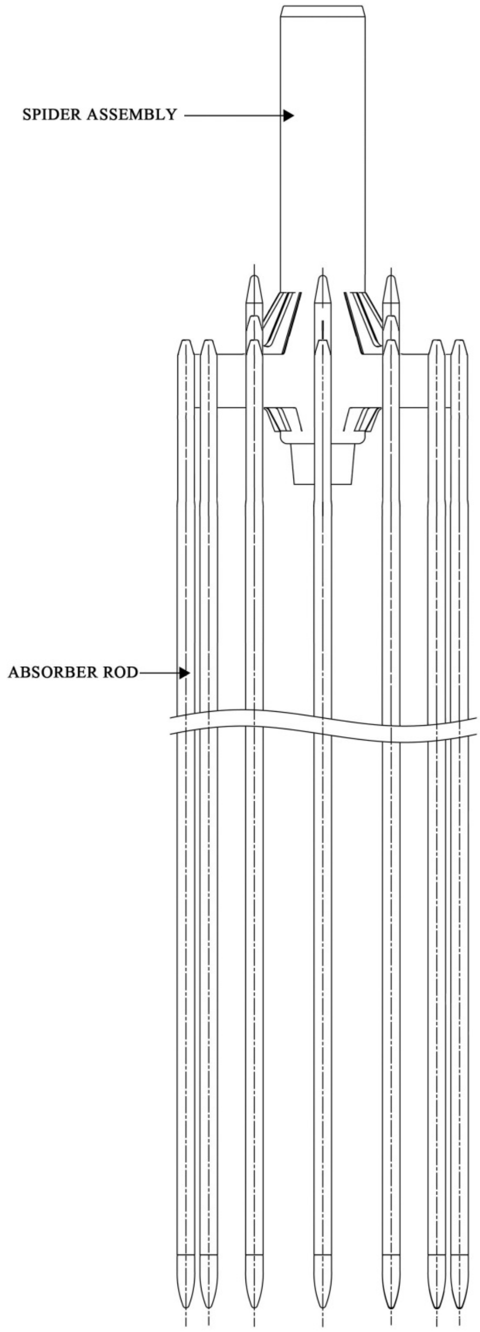
Considering the uncertainties, even under design basis conditions, the RCCA is compatible with fuel assemblies during insertion.

5.4.3.3.5 Rod Cluster Control Assemblies Design Summary

As shown above, the design criteria of RCCAs are met with available margins. Therefore, it can be justified that the RCCA is geometrically compatible with the proposed fuel assembly under DBC-1 and DBC-2, and structural integrity is maintained in DBCs.



F-5.4-1 Fuel Assembly



F-5.4-2 Rod Cluster Control Assembly

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 Chapter 4 are identified.

In general, under DBC-1, margins are always guaranteed between plant operation parameters and the set-point 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.

Assemblies with three different levels of enrichment of ^{235}U are used in the initial core loading to create a flat radial power distribution. Assemblies of the three different enrichments of ^{235}U 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 checkerboard-like 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 to 18 months (the transition cycles and Equilibrium Cycle, respectively). During the reloading process, almost 1/3 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 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 required 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 stored 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 keeping the axial power difference (ΔI) within the operating domain. The normal operating domain is divided into two regions, Region I or 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 allowable, 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. The signals from the 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 system, 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 that caused by the 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 position;

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

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),

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 reactivity changes);

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

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 withdrawn 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 Chapter 6.

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 under DBC-1. 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 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, the insertion or withdrawal of these banks result in a power change.

Boron is used to compensate for reactivity changes due to xenon poisoning during load following and 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 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 and overtemperature protection.

5.5.2.2 Important Parameter Description

5.5.2.2.1 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.

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

The calculation of nuclear enthalpy rise hot channel factor takes account of the uncertainties and penalties as follows:

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.

5.5.2.2.2 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:

$$F_Q = \max_{on\,z} Q\ (z)\ \ \text{(without uncertainty)}$$

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

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)

$S(z)$, densification factor (under DBC-2)

5.5.2.2.3 Axial Offset

The axial offset (AO) 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 top and bottom parts of the core and P_r is relative power.

5.5.3 Design Evaluation

5.5.3.1 Fuel Burn-up

The maximum discharge burn-up 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 burn-up is met.

Fuel burn-up 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 (natural boron enrichment) approximates to 10 ppm.

Based on the fuel management, the results on discharge burn-up of all the cycles are within the burn-up design limit, the calculation results are shown in *Fuel Management Report* (see Reference [4]).

5.5.3.2 Reactivity Feedback

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 cycle length. These design limits ensure that the core provides negative reactivity feedbacks when the temperature rises. (Safety Functions R1 and R2)

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 constant negative of Doppler coefficient, which is capable of providing 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 strictly required under DBC-1. The use of burnable absorber rods in the core reduces the concentration of soluble boron to prevent moderator temperature coefficient from becoming positive.

Since the reactivity coefficients change throughout the entire cycle length, the upper and lower limits of these reactivity coefficients are set for using as input data in safety analyses. Table T-5.5-3 and Figure F-5.5-7 show the design limits for transient analysis and calculated values for Cycle 1 and Equilibrium Cycle.

The calculated results, including Doppler coefficient, moderator coefficients and power coefficients are shown in *Nuclear Design Basis* (see Reference [5]).

5.5.3.2.1 Fuel Temperature (Doppler) Coefficient

The fuel temperature (Doppler) coefficient is defined as the reactivity change per degree change in fuel temperature and is primarily a measure of the Doppler broadening of ^{238}U 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 and ^{240}Pu. The effective resonance absorption cross sections of fuel increase with the rise of fuel temperature. This produces a corresponding negative reactivity.

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 Density and Temperature Coefficients

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

Since water density changes more rapidly with temperature, the moderator temperature coefficient becomes more negative with increasing temperature.

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 a positive reactivity.

Thus, if the concentration of soluble poison is too high, the value of the moderator temperature coefficient will be positive. The use of burnable absorbers reduces the initial concentration of soluble boron to maintain moderator temperature coefficient to be negative at operating temperatures.

The moderator coefficient becomes more negative with the increase of fuel burn-up, resulting from the reduction of concentration of soluble boron.

5.5.3.2.2.2 Moderator Void Coefficient

The moderator void coefficient measures the effect of voids in the moderator on core reactivity. This coefficient is not very significant because of the low void content in reactor coolant.

5.5.3.3 Control of Power Distribution

Under DBC-1 and DBC-2, to prevent the Departure from Nucleate Boiling (DNB) and to ensure the fuel rod integrity, the design limits of power distribution are met as follows, with at least a 95% confidence level:

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

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;

d) The fuel management design ensures that the power in fuel rod and burn-up are consistent with assumption applied in fuel rod mechanical integrity analysis.

For DBC-1, Figure F-5.5-8 shows F_Q^{LOCA} envelopes $Q_{(z)}^T$ for each cycle. All these curves obtained by calculations do not exceed the design limit, which confirms that F_Q^T stays below the design limit. All the transients in the normal operating domain complying with the operation limit for operating regions do not lead to exceeding of the assumptions used for LOCA analyses. The calculated results of nuclear enthalpy rise hot channel factor $F_{\Delta H}^N$ are shown in *Fuel Management Report* (see Reference [4]).

For DBC-2, Figure F-5.5-9 shows that the fuel melting limit envelopes $Q_{(z)}^T$ for each cycle. All the transients which do not trigger the overpower protection do not result in fuel melting. The overpower protection domain is shown in Figure F-5.5-10.

For the accidents in which the axial power distribution is only slightly perturbed, the reference axial power distributions are applied in the calculation of DNBR. The shape of the reference axial power distributions are shown in F-5.5-11. The reference axial power distributions are proven to be the most conservative axial power distribution in terms of Departure from Nucleate Boiling Ratio (DNBR) under DBC-1. Figure F-5.5-12 and Figure F-5.5-13 present the results of reference axial power distribution confirmation in Cycle 1 and Equilibrium Cycle. Under DBC-2, all the transients which do not trigger the overtemperature protection satisfy the DNB design limit, as shown in Figure F-5.5-14 and Figure F-5.5-15.

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 worth of RCCA banks. 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.

5.5.3.5 Shutdown Margin

Adequate shutdown margin or sub-criticality is maintained at power operation state or shutdown states respectively.

In all analyses involving reactor trip, the rod cluster control assembly with the highest reactivity worth is stuck out of core (stuck rod criterion). (Safety Functions R2 and R3)

The RCCAs provide sufficient negative reactivity to achieve shutdown state taking into account the power defect effect due to temperature variation from full power to zero power. The positive reactivity addition resulting from power reduction consists of contributions from Doppler effect, moderator effect, flux redistribution, void effect and specific uncertainties and allowances. To compensate for the positive reactivity addition due to the temperature drop from hot shutdown state to ambient temperature and to maintain sub-criticality, concentrated soluble boron is added to the reactor coolant. For all operating states, the concentration of soluble boron which is required for achieving sub-criticality is below the solubility limit. The hot shutdown state can be achieved by insertion of RCCAs. The minimum limit of shutdown margin is given in Table T-5.5-4. All the evaluation results are presented in *Nuclear Design Basis* (see Reference [5]).

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 relative to core criticality during refuelling are as follows:

a) $K_{eff} < 0.99$ when all RCCAs out of the core;

b) $K_{eff} < 0.95$ when all RCCAs inserted.

The calculation of criticality during refuelling state is giving in Reference [5].

5.5.3.6.2 Criticality for Fuel Storage

The criteria have been met for new fuel assembly storage in rack and fuel assembly storage in spent fuel pool in the UK HPR1000.

a) $K_{eff} < 0.95$ for new fuel assemblies storage in rack during normal condition,

b) $K_{eff} < 0.98$ for new fuel assemblies storage in rack during the most unfavourable conditions,

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

The consideration 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 absorption added in structural materials is considered,

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

e) The value of water temperature is taken to generate maximum reactivity in case of flooded conditions,

f) The most unfavourable conditions are adopted, and a sensitivity analysis is conducted to obtain appropriate uncertainty so as to consider mechanical uncertainty,

g) The fuel storage systems are designed to remain subcritical when two unlikely independent events occur.

Fuel storages in the new fuel storage rack and in spent fuel pool are introduced in PCSR Chapter 28, and the interim storage for spent fuel is introduced in PCSR Chapter 29.

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 reactor pressure vessel which is critical for 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 neutrons originating from the core.

The distribution of the neutron fluxes in various structural components varies considerably from core to reactor vessel. The neutron flux at internal surface of vessel can reach 1.3×10^{10} $n\cdot cm^{-2}\cdot s^{-1}$ based on core parameters and power distribution in Equilibrium Cycle, which can be used for long term radiation damage estimates. 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 (Dimensions at cold conditions)	
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 absorbing rods in a black RCCA	24
Grey RCCA	
Number of grey RCCAs	12
Number of absorbing rods in a grey RCCA	8
Number of inert stainless steel rods in a grey RCCA	16
Excess reactivity	
Max assembly K_{inf} (cold, clean core, zero boron)	
• Cycle 1	1.40236
• Equilibrium Cycle	1.38657
Max core K_{eff} (cold, zero power, BOC, zero boron)	
• Cycle 1	1.21216
• Equilibrium Cycle	1.23173

T-5.5-2 Nuclear Design Objectives and Limitations

Average discharge burn-up (Equilibrium Cycle), MWd/tU	47000
Average linear power density at nominal power, W/cm	179.5
Maximum linear power $Q^T(z)$ (under DBC-1)	Figure F-5.5-8
Total heat flux hot channel factor, F_Q^T (under DBC-2)	Figure F-5.5-9
Nuclear enthalpy rise hot channel factor (at hot full power), $F_{\Delta H}^N$	1.65

T-5.5-3 (1/2) Nuclear Design Parameters

	Design limits	Results of Cycle 1	
		BOC	EOC
Reactivity coefficients			
Doppler temperature coefficients, pcm/℃	-4.65 ~ -1.80	-3.16	-3.53
Moderator temperature coefficient, pcm/℃	≤ 0	-7.13	-52.37
Moderator density coefficient (G1G2N1 inserted), pcm/g.cm^{-3}	< 0.580×10^{5}	0.328×10^{5}	0.375×10^{5}
Maximum boron differential worth, pcm/ppm	-19.00	-11.57	-10.51
Effective delayed neutron fraction β_{eff} (value factor is 0.97*)	0.00750 ~ 0.00440	0.00697	0.00504
Neutron lifetime L, μs	31.0	27.8	29.5
Maximum differential worth of bank R, pcm/step	15.0(BCX), 21.0(EOC)	10.7(BCX)	14.5(EOC)

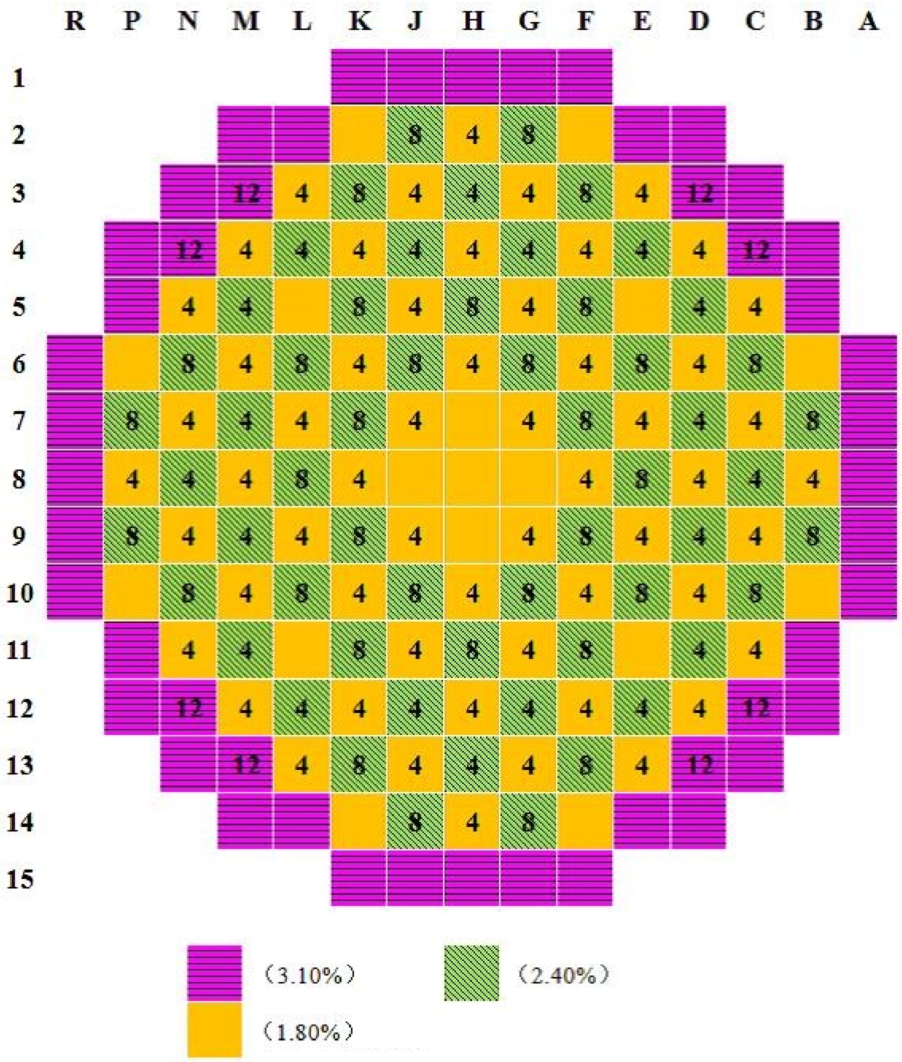
* Value factor comes from considering the six groups of delayed neutron

T-5.5-3 (2/2) Nuclear Design Parameters

	Design limits	Results of Equilibrium Cycle	
		BOC	EOC
Reactivity coefficients			
Doppler temperature coefficients, pcm/℃	-4.65 ~ -1.80	-3.36	-3.70
Moderator temperature coefficient, pcm/℃	≤ 0	-16.36	-70.60
Moderator density coefficient (G1G2N1 inserted), pcm/g.cm^{-3}	< 0.580×10^5	0.421×10^5	0.488×10^5
Maximum boron differential worth, pcm/ppm	-19.0	-7.4	-7.8
Effective delayed neutron fraction β_{eff} (value factor is 0.97*)	0.00750 ~ 0.00440	0.00610	0.00508
Neutron lifetime L, μs	31.0	15.8	20.5
Maximum differential worth of bank R, pcm/step	15.0(BCX) 21.0(EOC)	8.0(BCX)	14.5(EOC)

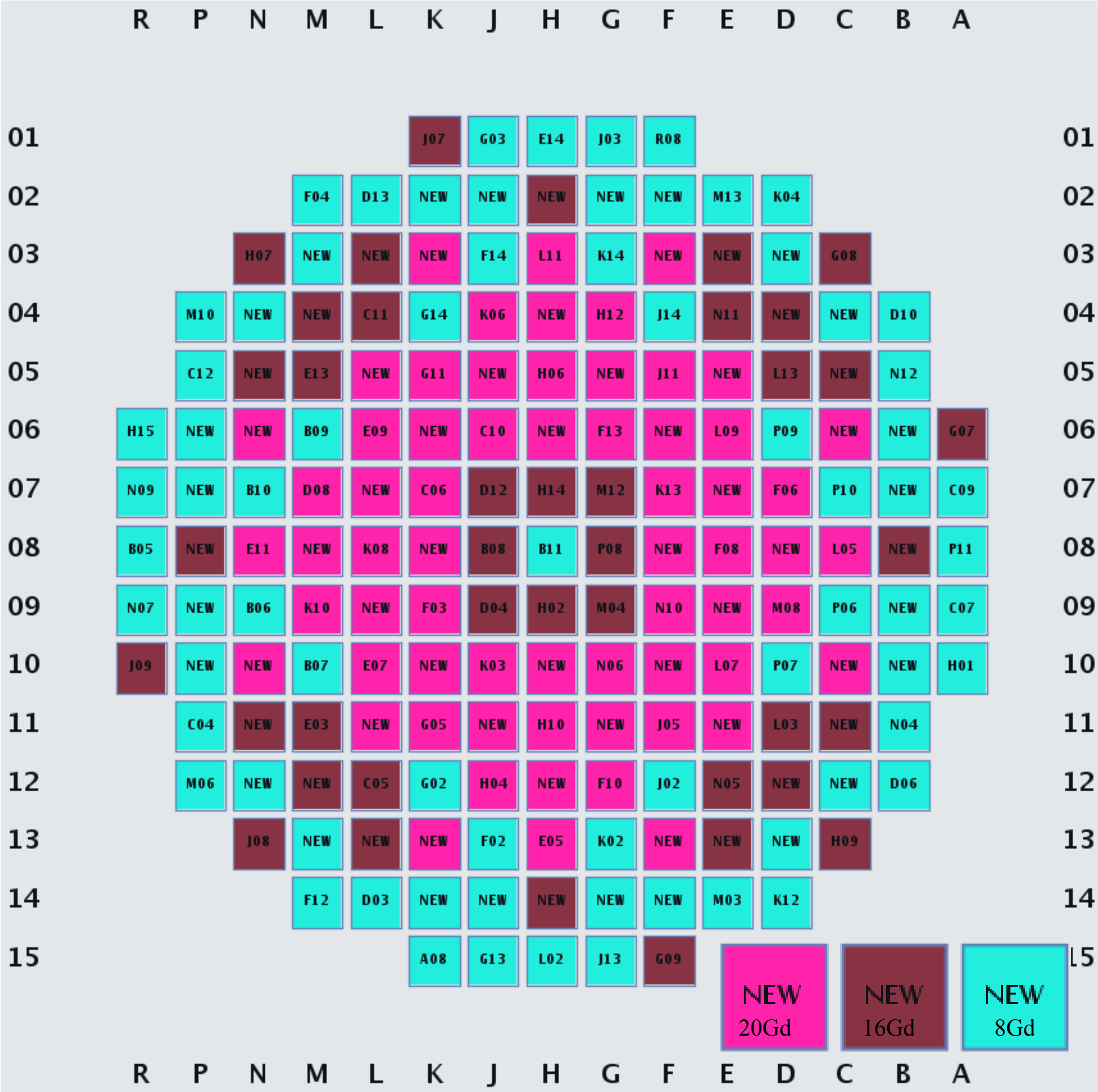
* Value factor comes from considering the six groups of delayed neutron

T-5.5-4 Shutdown Margin



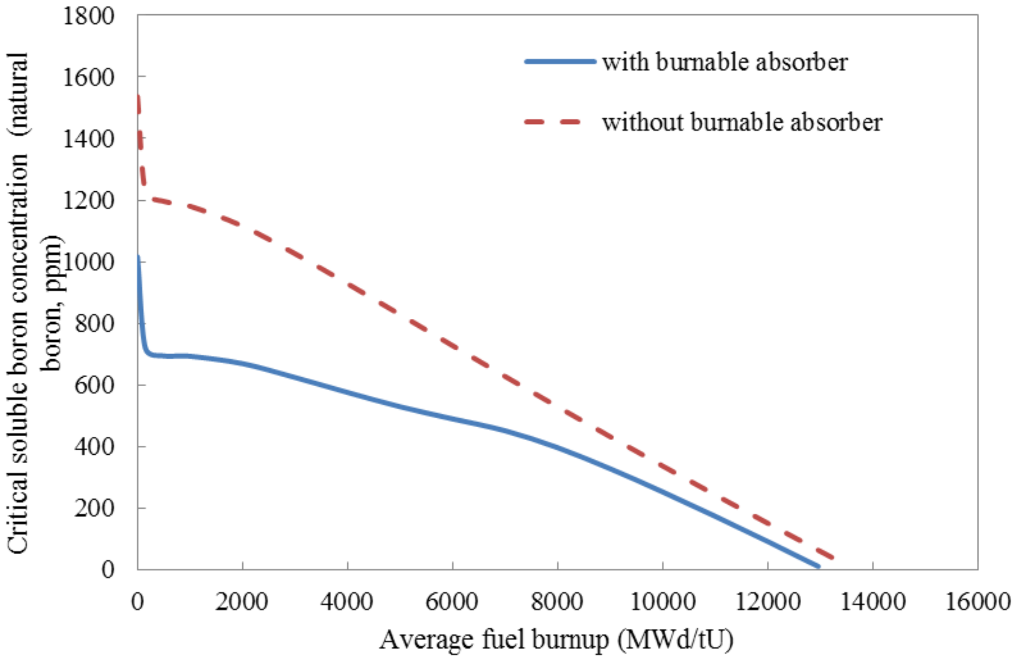
F-5.5-1 Loading Pattern of Cycle 1

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



F-5.5-2 Loading Pattern of Equilibrium Cycle

* The enrichment of new fuel assemblies is 4.45% in Equilibrium Cycle.



F-5.5-3 Core Depletion of Cycle 1 with and without Burnable Absorbers

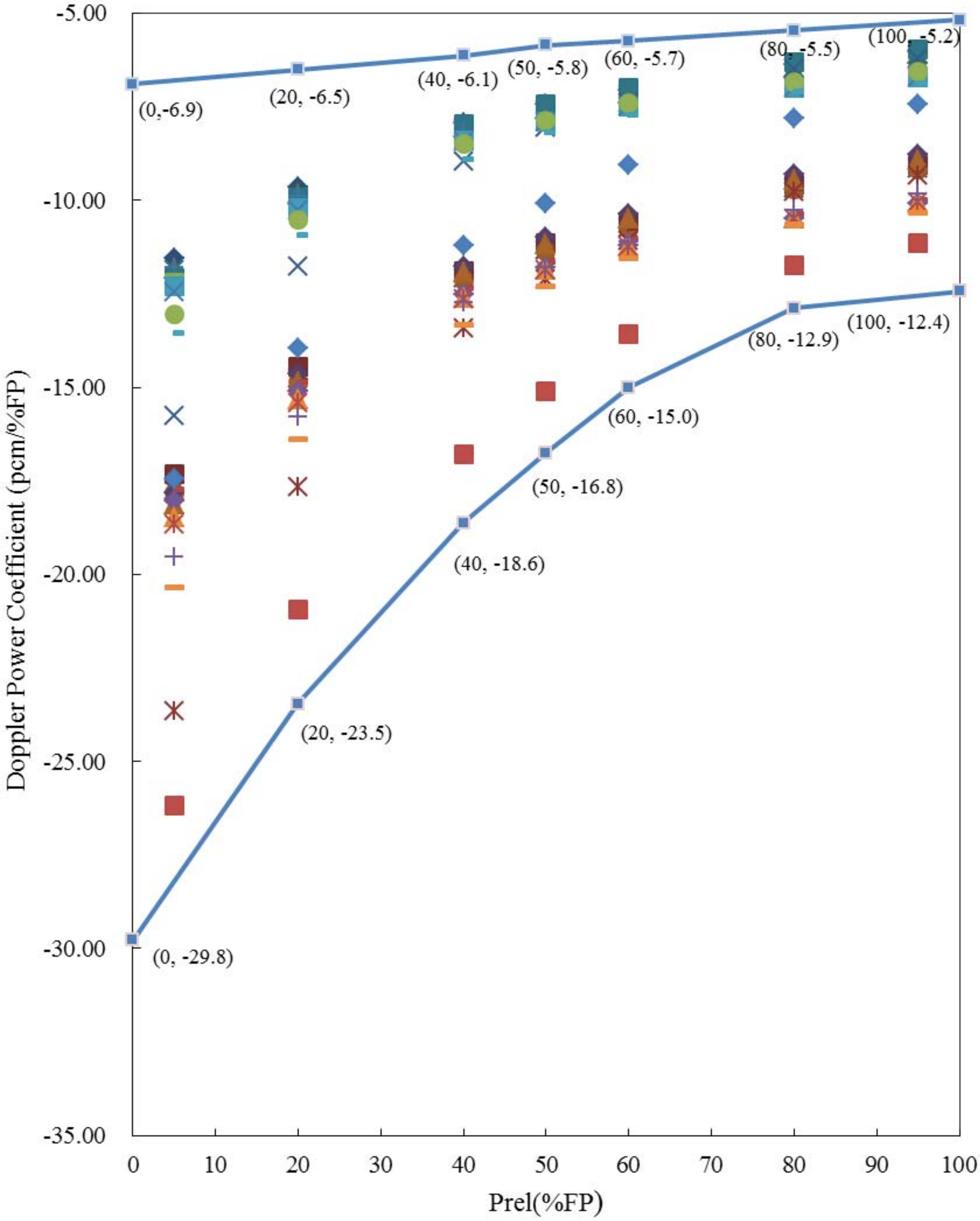
F-5.5-4 Burnable Absorber Rod Layout in Fuel Assemblies

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

	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

F-5.5-6 Arrangement of RCCA Banks



F-5.5-7 Doppler Power Coefficient for Cycle 1 and Equilibrium Cycle

* The blue curves present the upper and lower limits (Maximum absolute value) of Doppler power coefficient.

F-5.5-8 Maximum Linear Power Density (DBC-1)

F-5.5-9 Maximum Axial Power Distribution (DBC-2)

F-5.5-10 Overpower Protection Domains for Cycle 1 and for Subsequent Cycles

F-5.5-11 Reference Axial Power Distributions

F-5.5-12 Reference Axial Power Distribution Confirmation (Cycle 1)

F-5.5-13 Reference Axial Power Distribution Confirmation (Equilibrium Cycle)

F-5.5-14 DNBR Margin under DBC-2 (Cycle 1)

F-5.5-15 DNBR Margin under DBC-2 (Equilibrium Cycle)

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 both earlier in this chapter and 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);

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;

b) Fraction of fuel failure is limited under DBC-3 and DBC-4 to ensure the reactor to be 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 (see Reference [6]) 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:

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

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

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

q'_{CGN} : Uniform DNB (or critical) heat flux predicted by the CHF correlation

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

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

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 accidents, the DNBR design limit is determined by using the CGNC-1 Critical Heat Flux (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 limit design takes into account the effect of the rod bow penalty. Rod bow in relation to DNBR is described in Sub-chapter 5.6.3.1.3.4.

The statistical DNBR design limit is 1.35 (see Reference [7]).

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 deterministic DNBR design limits are as follows:

- 1.17 for loss of flow conditions;
- 1.16 for full flow conditions.

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 than 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 CGNC-1 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 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. {

}

5.6.3.1.2 Engineering Hot Channel Factor

5.6.3.1.2.1 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.

5.6.3.1.2.2 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 verification, and the manufacturing data of 95% of the limit fuel rods cannot exceed this design value at 95% of confidence level.

5.6.3.1.2.3 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.

5.6.3.1.3 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.

5.6.3.1.3.1 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.

5.6.3.1.3.2 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.

5.6.3.1.3.3 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.

5.6.3.1.3.4 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 { }. For the burn-ups greater than { }, 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 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 advanced fuel assemblies in 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;

e) Flow in the gaps between the fuel assemblies on the core periphery and the adjacent the 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 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. Then single simple 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).

$Y(\alpha, G, \varnothing)$, the corrective factor for two-phase model has two major advantages: its lack of discontinuity and its coherence with the split model.

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 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 UK HPR1000 indicates that a large margin to density wave instability exists. The method of Ishii applied to UK HPR1000 design is conservative due to the parallel open channel feature of 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 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 5.6.3.1.2. In addition, flow distribution is considered in a penalizing way as discussed in Sub-chapter 5.6.3.1.3.

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	
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

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

a) Safety Assessment Principles and Technical Assessment Guides of Office for Nuclear Regulation related to fuel and core design:

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

   2) 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, Revision 7 (2015), ONR

b) International Atomic Energy Agency safety standards related to fuel and core design:

c) Relevant operational experience from other reactor types.

d) Regulatory Observations and Regulatory Issues related to fuel and core design.

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 (see Reference [8]).

5.8 Commissioning and Testing

5.8.1 Core Physics Test

Nuclear design calculations guarantee that the core physics parameters do not exceed the safety values. Through core physics tests, that the core is operated as designed can be ensured by checking the core physics parameters are consistent with design predictions. After each core refuelling, core physics tests are carried out to ensure that the reactor is safe and operated in accordance with design.

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 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 and 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 analyses is conservative.

The commissioning and testing arrangements will be adapted from those developed for the HPR1000.

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

As recommended in Chapter 21, the fuel rod integrity will be confirmed in-service mainly by the Nuclear Sampling System (REN [NSS]). The design of which has been established for detection, monitoring and sampling of the primary circuit. The results of the sample analyses 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 when the abnormal radioactivity levels within the primary coolant are detected. Visual inspection will be also required to examine items including cladding surface and structural integrity of the grid, for example.

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 the potential releases from the failures are within the capability of the plant clean-up 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 safety functions for UK HPR1000 discussed in Chapter 4. Evidence provided demonstrates that these principles are satisfied by the design of the UK HPR1000.

5.12 References

[1] CGN, Fuel Rod Design, GHX42500004DRRL44GN, Rev. B, 2018

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[3] CGN, Fuel Assembly Mechanical Design Report, GHX42500002DRRL44GN, Rev. B, 2018

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[7] CGN, Thermal Hydraulic Design, GHX00100004DRRG03GN, Rev. B, 2018

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Appendix 5A Chapter 5 Computer Code Description

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

Computer Codes	Sub-chapter
JASMINE	5.4.3.1
PCM	5.5.3
POPLAR	5.5.3
GINGKO	5.6.3
LINDEN	5.6.3
LOCUST	5.6.3
BIRCH	5.6.3

- JASMINE

JASMINE is a fuel rod design code used to accurately simulate of fuel rod behaviour during irradiation. It performs the thermal-mechanical analyses through the use of various calculation models such as thermal model, mechanical model, FGR model, internal pressure model and cladding corrosion model.

JASMINE applies the space discretization and time discretization in the calculation. The space discretization includes axial discretization and radial discretization. The time discretization is the combination with the macro time step and micro time step. For each macro time step, time-related variables (such as linear heat rate, fast neutron flux, and thermal-hydraulic data) are given by the user. For the micro time step, it is automatically divided by the code between the adjacent macro time steps.

JASMINE is applicable to calculate PWR fuel rod behaviour with the fuel of UO_2 and UO_2-Gd_2O_3, and the cladding of zirconium alloy.

- PCM

PCM is a nuclear design code package that contains PINE and COCO. PINE is an advanced Pressurized Water Reactor (PWR) fuel assembly 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.

a) 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 burn-up calculation.

The equivalence principle is applied to carry out resonance calculation. The Method of Characteristics (MOC) is applied to perform two-dimensional heterogeneous transport calculation. B1 approximation is applied to take into account the leakage effect. PINE uses three different advanced burn-up calculation strategies, which are Projected Predictor-Corrector (PPC) method, Linear Rate (LR) method and Log Linear Rate (LLR) method.

b) 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 (RIA).

The solver of COCO is based on Nodal Expansion Method (NEM) which can handle 2-D and 3-D geometries. The NEM solver can provide flux distribution in full core and 1/4 core geometries. Furthermore, the NEM solver is accelerated using CMFD 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 burn-up models are developed. The former focuses on the fission products, minor actinides, etc. The latter handles the intra-node burn-up distribution. In macroscopic burn-up, nodal surface burn-up is calculated to correct cross sections.

The PCM nuclear design code package has been intensively validated against Nuclear Power Plant (NPP) data, experimental data and benchmarks.

■ 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 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 validation of the POPLAR includes NPP data validation.

- GINKGO

GINKGO is a system transient analysis code, which is used to analyse PWR transients under normal operating conditions and accident conditions. For these transients, GINKGO simulates the reactor vessel and core, hot and cold legs, pressurizer, steam generator, reactor coolant pump in PWR plant. The modelling of Nuclear Steam Supply System (NSSS), Engineered Safety System (ESS), Reactor Protection System (RPS), Instrumentation and Control System (I&C) and secondary system components are also taken into account.

To account for the thermal-hydraulics features of the coolant in different transients, the separated phase model at thermal equilibrium is used in the code. Three governing mixture balance equations combined with a drift-flux model are applied. A reactor point kinetic model with six-group delayed neutron and a simplified decay heat model are combined to predict the core transients.

The validation of GINKGO includes separate effect validation and integral effect validation.

- 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, including pressure, mass flow rate, quality and void fraction, etc. Based on the calculated thermal-hydraulic parameters, the Departure from Nucleate Boiling (DNB) of reactor core can be predicted by using a specific Critical Heat Flux (CHF) correlation.

The two-phase flow in LINDEN is the four equations model combined with a drift-flux correlation. It takes into account the slip velocity between liquid and vapor phases and also the thermal non-equilibrium of liquid phase during sub-cooled boiling. The flow model has four conservation equations, including a mixture mass equation, a mixture energy equation, a mixture momentum equation and a liquid energy equation. Among which, the liquid energy equation is used to simulate the thermal non-equilibrium of liquid phase during sub-cooled boiling.

The validation of LINDEN includes experimental data validation and NPP operational data validation.

- LOCUST

LOCUST is a system thermal-hydraulic code which has the capability of performing LOCA analysis. It focuses on the analysis of LBLOCA, IB/SBLOCA, SGTR, etc.

LOCUST is used to simulate two-fluid, non-equilibrium, and heterogeneous hydrodynamic conditions in various NPP transients. A six-equation model is employed in hydrodynamics model, which forms the trunk of LOCUST. Auxiliary models include heat structure model, trip system, control system, and point reactor kinetics model.

The most important features of LOCUST are flexible nodalization, capability to analyse two-fluid, thermal non-equilibrium in all fluid volumes. The code is incorporated with models to simulate special processes such as choked flow, thermal stratification, and counter-current flooding limitations.

The validation of LOCUST includes separate effect validation and integral effect validation.

■ BIRCH

BIRCH is a fuel rod temperature analysis code, mainly used to analyse the integrity of fuel rod in PWR under accident conditions. BIRCH calculates the radial temperature distribution of a fuel rod and the heat flux of cladding surface during transient conditions. In addition, it also calculates energy storage in the fuel pellet, gap heat transfer coefficient and thermal expansion of pellet and cladding, etc.

A 1-D heat conduction differential equation is implemented in heat conduction model together with auxiliary closure models, including cladding-coolant heat transfer model, gap conductance model, water-zircaloy reaction model and fuel pellet melting model, etc.

The physical properties of coolant and data of evolution of nuclear power required in the calculation are obtained from the system transient analysis code GINKGO or 3-D core calculation code COCO.

The validation of BIRCH includes separate effect validation and integral effect validation.