Revision	Approved by	Number of Pages
000	郝…	39
Approval Date	10/11-17	

General Nuclear System Ltd.

UK HPR1000 GDA Project

Document Reference:	HPR/GDA/PSR/0005

Preliminary Safety Report
Chapter 5
Reactor Core

DISTRIBUTION LIST

Recipients	Cross Box
GNS Executive	☐
GNS all staff	☐
GNS and BRB all staff	☒
CGN	☒
EDF	☒
Regulators	☒
Public	☒

Table of Contents

5.1 List of Abbreviations and Acronyms

Ag-In-Cd	Silver-Indium-Cadmium
BOC	Beginning Of Cycle
CHF	Critical Heat Flux
DBC	Design Basis Condition
DEC	Design Extension Condition
DNB	Departure from Nucleate Boiling
DNBR	Departure from Nucleate Boiling Ratio
EOC	End Of Cycle
GDA	Generic Design Assessment
GRCA	Grey Rod Cluster Assemblies
HPR1000 (FCG3)	Hua-long Pressurized Reactor under Construction at Fangchenggang nuclear power plant unit 3
ICIA	In-Core Instrumentation Assembly
KSS	Online Monitoring System[OMS]
NNSA	National Nuclear Safety Administration
NPSH	Net Positive Suction Head
PCI	Pellet - Cladding Interaction
RCCA	Rod Cluster Control Assemblies
SPND	Self-Powered Neutron Detector
TCLS	Thermoelectric Converter of Level Sensor
UK HPR1000	The UK version of the Hua-long Pressurized Reactor

System codes (XXX) and system abbreviations (YYY) are provided for completeness in the format (XXX [YYY]), e.g. Online Monitoring System (KSS [OMS]).

5.2 Introduction

This chapter describes the main mechanical components of the reactor core, as well as the fuel rod and fuel assembly design, the nuclear core design, and the thermal-hydraulic design. The design of these components supports the following fundamental safety objective of the UK version of the Hua-long Pressurized Reactor (UK HPR1000) in the area of nuclear safety – protection of the workers and the public:

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.

For UK HPR1000, the decision has not yet been taken on fuel supplier, as detailed technical and commercial evaluation of the supply chain is proposed before this selection is made. However, maximum enrichment of fuel for the UO_2 assemblies should not exceed 5% and maximum discharge burnup level shall not exceed the burnup safety analysis limit determined according to the fuel assembly and fuel rod performance analysis.

The core design of the UK HPR1000 will be described in detail in the GDA Pre-Construction Safety Report (GDA PCSR) and is expected to support the following two aspects for the UK HPR1000:

UKa) ‘‘The reactor core design (including nuclear core design, thermal hydraulic design, fuel rod and fuel assembly design) of the UK HPR1000 will set safety analysis bounding limits for the reactor core which together ensure that the fundamental safety functions are delivered during normal operation and following all design basis condition events’’;

UKb) ‘‘The design and intended operation of the UK HPR1000 core will ensure the risks to the public following a Design Extension Condition event identified for the UK HPR1000 design are below the targets and have been reduced As Low As is Reasonably Practicable (ALARP)’’.

Confirmation of these objectives will be described in detail in the GDA Pre-Construction Safety Report (GDA PCSR), in chapters 5, 12 and 13.

The reference plant of UK HPR1000 is HPR1000 (FCG3), and it will mainly introduce the core design of HPR1000 (FCG3) in this chapter. The general requirements of the HPR1000 (FCG3) reactor core are as follows.

The reactor core is composed of 177 fuel assemblies, cooled and moderated by light water at a pressure of 15.5 MPa. The reactor operates for periods of 12 to 18 months, known as a Cycle or Fuel Cycle, before the reactor is shut down, and a fraction of new fuel assemblies are loaded and the core is shuffled to optimize efficiency. For Cycle 1, assemblies are loaded with fuel of three different levels of enrichment of U-235. And for equilibrium cycles, new fuel assemblies with enrichment of 4.45% w/o U-235 are loaded.

Each fuel assembly, as shown in F-5.3-1, consists of 264 fuel rods in a 17x17 square array. The centre position in the fuel assembly has a guide thimble that is reserved for in-core power monitoring instrumentation. The remaining 24 positions in the fuel assembly have guide thimbles. The guide thimbles are joined to the top and bottom nozzles of the fuel assembly and provide the supporting structure for the fuel grids. The fuel grids consist of an egg-crate arrangement of interlocked straps that maintain lateral spacing between the rods. The grid straps have spring and dimples that grip and support the fuel rods. The fuel rod consists of pellet stacks, top and bottom end plugs, holding spring and cladding. The cladding provides the first barrier to the release of fission products.

Soluble boron in the primary coolant serves as a neutron absorber. The concentration of boron is varied to control reactivity changes that occur relatively slowly, including the effects of fuel burnup and to maintain the desired negative reactivity coefficients. Boric acid with enriched ^{10}B is used to limit the amount of soluble boron required. Additional neutron absorber (gadolinium, an oxide of the element Gadolinium Gd), is used to balance core reactivity and ensure an even power distribution. The Gadolinium is depleted (through neutron absorption) during the course of each fuel cycle, and is therefore known as a ‘burnable neutron absorber’ or ‘burnable absorber’.

The core reactivity and power distribution are also controlled by movable Rod Cluster Control Assemblies (RCCA), which are neutron absorber rods that enable controlled changes in reactivity and power distribution to be made, and thus to start up or shut the reactor down. Each RCCA consists of a group of individual neutron absorber rods securely fastened at the top end to a common hub or spider assembly as shown in F-5.3-2.

The nuclear core design analyses establish the core locations for control rods and burnable absorbers. Although in practice the RCCA locations remain fixed between fuel cycles, while the location of the burnable absorbers rods may be customized for each individual fuel cycle. The analyses define design parameters, such as fuel enrichments for fresh fuel, and boron concentration, and ensure safety analysis limits are met. The design assessment also confirms that the design ensures inherent stability against radial and axial power oscillations, and for control of axial power oscillation induced by control rod movements.

The thermal-hydraulic design analyses confirm that the fuel is maintained in a coolable geometry, and that adequate heat transfer is provided between the fuel cladding and the reactor coolant, to allow heat to be taken away to generate steam through the secondary coolant system and produce electrical power. The thermal-hydraulic design takes into account local variations in dimensions, power generation, flow distribution and mixing. The mixing vanes incorporated in the fuel assembly spacer grid design and the fuel assembly intermediate flow mixers induce additional flow mixing between the various flow channels within a fuel assembly, as well as between adjacent assemblies. In-core and ex-core instrumentation are provided to monitor the nuclear and thermal-hydraulic performance of the reactor in real time, and demonstrate alignment to the parameters modelled.

The design of the Reactor Core will ensure that the fuel can be cooled and reactivity controlled.

In the HPR1000 (FCG3) design, the Fuel System and the design supports the delivery of the three fundamental safety functions of:

- control of reactivity,
- removal of heat from the reactor from the fuel store and
- confinement of radioactive material, shielding against radiation and control of planned radioactive releases, as well as limitation of accidental radioactive releases.

The principal design parameters of reactor nuclear, thermal-hydraulic and mechanical are presented in T-5.2-1.

T-5.2-1Reactor Design Parameters

Thermal and hydraulic design parameters	**HPR1000 (FCG3)**
1）Reactor thermal power, MW	3150
2）Heat generated in fuel, %	97.4
3）Nominal system pressure, MPa	15.5
4）Minimum steady-state system pressure, MPa	15.3
Coolant flow	
5）Total thermal design flow, m^3/h	72000
6）Core bypass flow rate, %	6.5
7）Core flow area for heat transfer, m^2	4.33
8）Average velocity along fuel rods, m/s	4.32
9）Core average mass velocity, $g/(cm^2\ s)$	305.1
Coolant temperature,℃	
10）Nominal inlet	288.6
11）Average in core	308.2
12）Average in vessel	307.0
Heat transfer	
13）Heat transfer surface area, m^2	5094.7

Thermal and hydraulic design parameters	HPR1000 (FCG3)
14）Average core heat flux, W/cm^2	60.22
15）Average linear power density, W/cm	179.5
16）Peak linear power protection set point, W/cm	590.0
17）Peak linear power limit, W/cm	700
18）Power density in hot conditions, kW/l	102.5
Fuel assemblies	
19）Number of fuel assemblies	177
20）Fuel rods per assembly	264
21）Lattice rod pitch, cm	1.26
22）Overall transverse dimensions, cm	21.4×21.4
23）Number of grids per assembly	11
24）Composition of grids	Zirconium alloy & Inconel
Reactor Core	
29）Equivalent diameter, cm	323
30）Core average active fuel height, cm	365.76
31）Height-to-diameter ratio	1.13

Values are approximate.

5.3 Fuel System Design Basis

5.3.1 Safety Functional and Design Requirements

The design of the fuel system, including the supporting analysis, will ensure that the fundamental safety functions are delivered for all operating modes of the HPR1000 (FCG3) for all fuel cycles.

The fundamental safety design requirements of the fuel system under normal operation and frequent design basis faults (Design Basis Conditions (DBC-1 and DBC-2), see chapter 12) are:

- The reactor core design ensures the control of core reactivity to enable the nuclear chain reaction to be stopped under all circumstances, and the reactor returned to a safe shutdown state using two diverse shutdown systems;
- The reactor core design, thermal hydraulic and fuel assembly design ensure that heat produced in the core can be removed by the reactor coolant under all circumstances;
- The reactor core design and fuel rod performance design ensure that radioactive substances (including fuel, actinides and fission products) can be contained by the fuel cladding.

Generic assessments of the chosen fuel system will be performed to demonstrate that the functional requirements are met.

The fundamental safety design requirements of the fuel system under infrequent design basis faults (DBC-3 and DBC-4 conditions, see chapter 12) are:

- The reactor core design ensures the control of core reactivity to enable the nuclear chain reaction to be stopped under all circumstances, and the reactor returned to a safe shutdown state using two diverse shutdown systems;
- The reactor core design, thermal hydraulic and fuel assembly design ensure that heat produced in the core can be removed by the reactor coolant under all circumstances;
- The reactor core design and fuel rod performance design ensure limitation of release of radioactive substances, in particular fission products from potentially damaged fuel in line with the objective;

Generic assessments of the chosen fuel system will be performed to demonstrate that the functional requirements are met.

5.3.2 Detailed Description of the Fuel System

5.3.2.1 Fuel Assembly

Each fuel assembly is composed of 264 fuel rods arranged in a 17 x 17 square array or fuel bundle, within a supporting structure (or skeleton). The skeleton consists of 1 top nozzle, 1 bottom nozzle, 2 end grids without mixing vanes, 6 structure grids with mixing vanes and 3 mid-span flow mixers, 24 guide thimbles and 1 instrumentation tube.

5.3.2.1.1 Fuel Rods

The fuel pins or fuel rods consist of a stack of sintered uranium dioxide pellets with or without burnable poison (gadolinium) encapsulated in a closed Zirconium alloy tube that is plugged and welded. Within the fuel rod, pressurized helium is utilized as backfill gas. A plenum is provided in the tube for accommodation of fission gas generated during irradiation. A hold-down coil spring on the top end of the pellet stack holds the fuel pellets in position during transport and loading. The fuel pellet characteristics are very tightly controlled during manufacture to provide the desired performance characteristics when in operation. The fuel rods are supported at intervals along their length by the grid assemblies that maintain the lateral spacing between the rods. The fuel rods are loaded into the fuel assembly structure so that there is clearance between the fuel rod ends and the top and bottom nozzles to accommodate fuel rod expansion.

5.3.2.1.2 Top Nozzle and Hold-Down System

The top nozzle is the upper structural element of the fuel assembly and it provides a partial protective housing for core component assemblies. It consists of an adapter plate, welded to a top plate. The adapter plate has an array of holes and slots to allow the coolant to flow upward. Four sets of leaf springs are mounted on the top plate to balance hydraulic up-flow forces during operation and accommodate changes in fuel assembly length relative to the distance between the core plates. The fixed end of the spring set is held in place by a screw which bears directly on the top leaf.

5.3.2.1.3 Bottom Nozzle

The bottom nozzle is the bottom structural member of the fuel assembly. It protects the fuel assembly from debris and channels the coolant through the rod bundle. It consists of a perforated plate with four legs and an anti-debris device. A plenum is formed by the four legs to permit inlet coolant to flow into the fuel assembly. The perforated plate supports the anti-debris device and provides housing for guide thimble attachment screws.

5.3.2.1.4 Grids

Along with the guide thimbles and the top and bottom nozzle, the grids make up the skeleton of the fuel assembly. Each grid is an array of slotted grid straps that are interconnected and welded at the intersections to form an egg-crate structure.

There are two types of grids in each fuel assembly: structural grids and 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 operations 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.

5.3.2.1.5 Guide Thimbles

The thimbles provide channels for insertion of different types of core components depending on the position of the particular fuel assembly in the core. The annular clearance between RCCA rodlet outer diameter and guide thimble inner diameter is designed to enable rapid RCCA insertion during a reactor trip.

5.3.2.1.6 Instrumentation Tube

The instrumentation tube or thimble is located in the centre position of each fuel assembly and provides a channel for insertion of in-core neutron detectors. The instrumentation tube is fixed in position between the top and bottom nozzles. The mechanical elements of the in-core neutron detectors are described further in Section 5.3.2.3.

5.3.2.2 Rod Cluster Control Assemblies

The RCCA provide core reactivity and power control during operations and enable the reactor to be shut down by releasing a magnetic latch, and allowing the absorber rods to lower into the fuel assemblies rapidly.

A RCCA includes several absorber rods with Silver-Indium-Cadmium (Ag-In-Cd) alloy as the neutron absorber material, cladding in stainless steel or stainless steel inert rods. The rods are fastened securely to a common spider assembly at the top. A black RCCA includes 24 absorber rods, and a grey RCCA includes 8 absorber rods and 16 stainless steel rods.

5.3.2.3 In-Core Detectors

The In-Core Detectors are Self-Powered Neutron Detectors (SPND). The active material used for these detectors is Rhodium. SPNDs are placed at seven points axially along the reactor core active region for each instrumented Fuel Assembly. The SPND are used for on-line, in-core monitoring of neutron flux.

There are 46 in-core instrumentation probe lines or fingers located across the core to ensure that the radial neutron flux profile that is representative of the whole core can be monitored, as shown in F-5.3-3. All instrumentation lines are fed into the RPV through the upper closure head in 4 groups I-IV.

The mechanical design of the in-core instrumentation will satisfy the requirements imposed by:

- Operational and reliability requirements, ensuring ergonomic access for maintenance and testing;
- Load conditions;
- Proper selection and use of materials, with appropriate qualification;
- Ensuring doses to operators undertaking maintenance and replacement (approximately every 3 cycles), are ALARP and wastes generated are minimised and do not create novel waste streams;
- Adopting good manufacturing practices.

5.3.3 Design Evaluation

5.3.3.1 Fuel Rod Performance

A set of nuclear safety analysis bounding limits will be set for the fuel rod performance analysis, which will be explained in future stages of GDA. These design limits will be set following a further review before a fuel and fuel assembly manufacturer is selected. The design assessment for the fuel rod will address the following potential physical phenomena：

a) Irradiation densification and swelling;

b) Fuel temperature and margins;

c) Fission gas release;

d) Irradiation creep and growth;

e) Pellet and cladding interaction-Stress corrosion cracking (PCI-SCC);

f) Creep ovalization and the risk of collapse;

g) Strains and stresses;

h) Cycling and fatigue;

i) Oxidation and hydriding;

j) Vibration and fretting wear.

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

The fuel rod performance assessment will continue to demonstrate that the design criteria are satisfied for the fuel cycle ahead before fuel is loaded.

5.3.3.2 Fuel Assembly Performance

Design criteria are yet to be defined for the UK. The ability of fuel assembly will be evaluated to withstand the mechanical stresses as a result of:

a) Fuel handling and loading;

b) Power variations;

c) Temperature gradients;

d) Hydraulic forces, 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;

i) All DBC Conditions.

Assessment will continue to demonstrate that the design criteria are satisfied for the fuel assembly design.

5.3.3.3 Rod Cluster Control Assemblies Performance

The assessment of the rod cluster control assemblies will consider the following issues:

a) Internal pressure and cladding stresses during normal, transient and accident conditions;

b) Thermal stability of absorber materials;

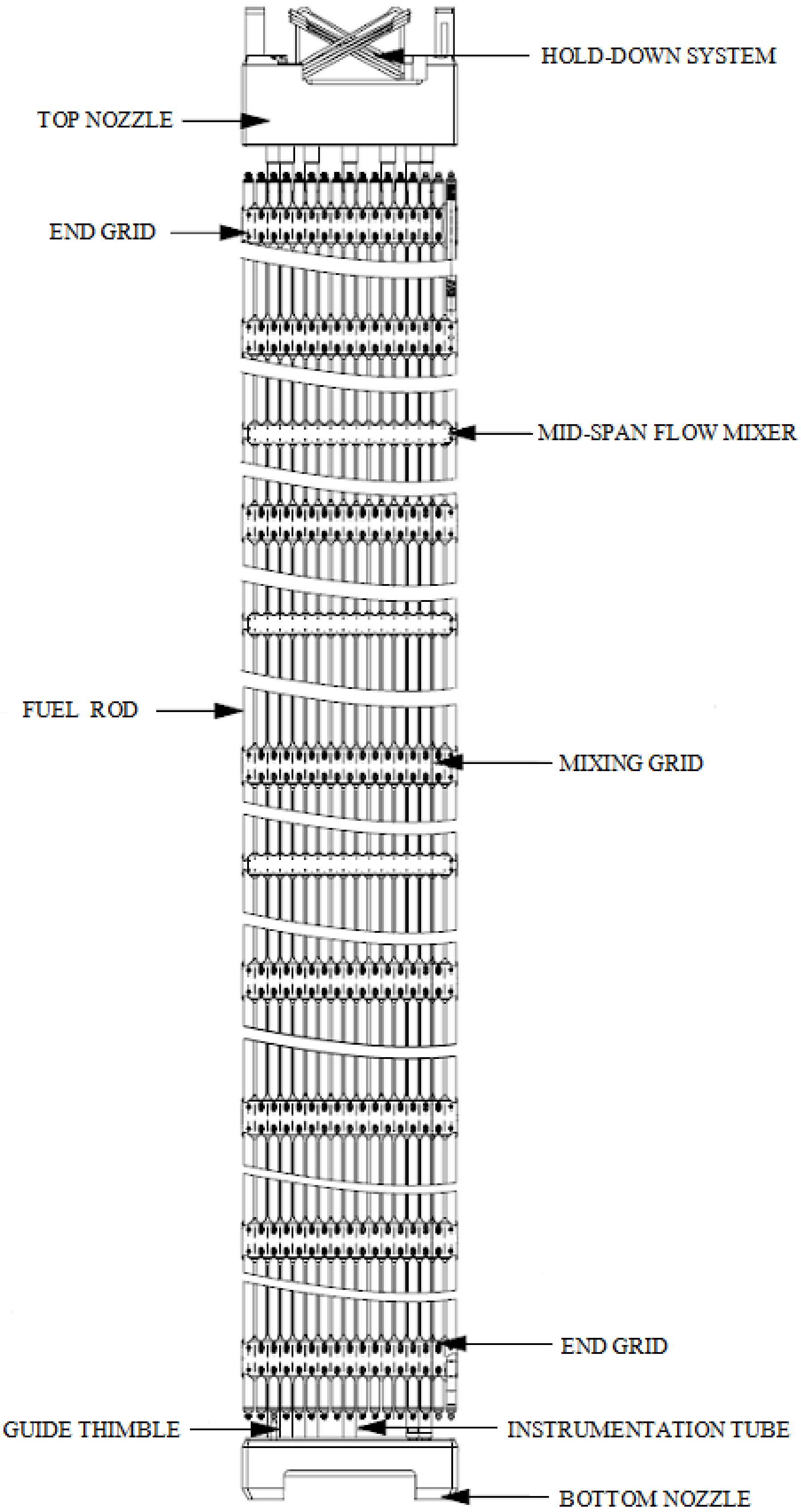
c) Irradiation stability of absorber materials and the cladding;

d) Performance under insertion;

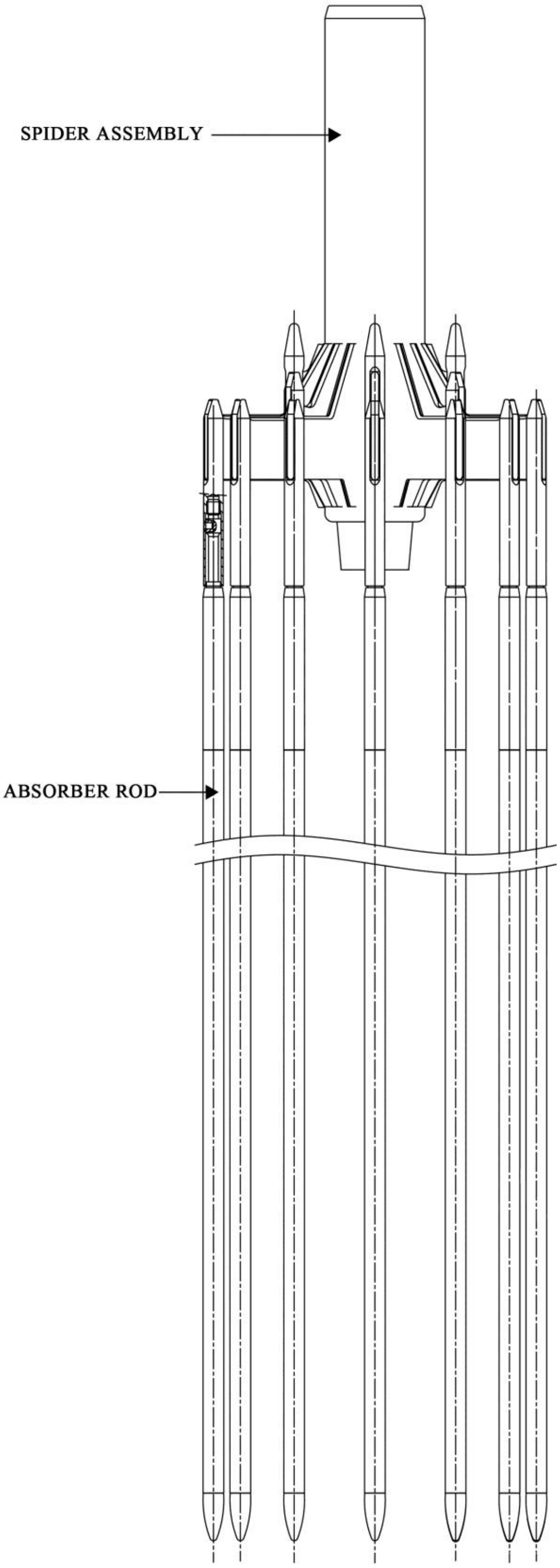
The control rod evaluations will demonstrate that the design criteria are satisfied.

5.3.4 Testing and Inspection Plan

A formal fuel system procurement exercise will be undertaken to select an international manufacturer of high-quality fuel systems using relevant good practice, targeting zero fuel pin failures under DBC-1 and DBC-2 conditions in all cycles. A quality assurance and quality control program will be developed to document and monitor the design, analysis, production and witness/inspection activities which might affect the quality of fuel system, identifying all steps which have the potential to have an impact on delivery of safety functions. Chapter 21 will describe how reactor chemistry is monitored to ensure fuel pin integrity is assured. Chapter 23 will describe the irradiated fuel examination in nuclear power plant to prevent the damaged fuel from returning to the core.

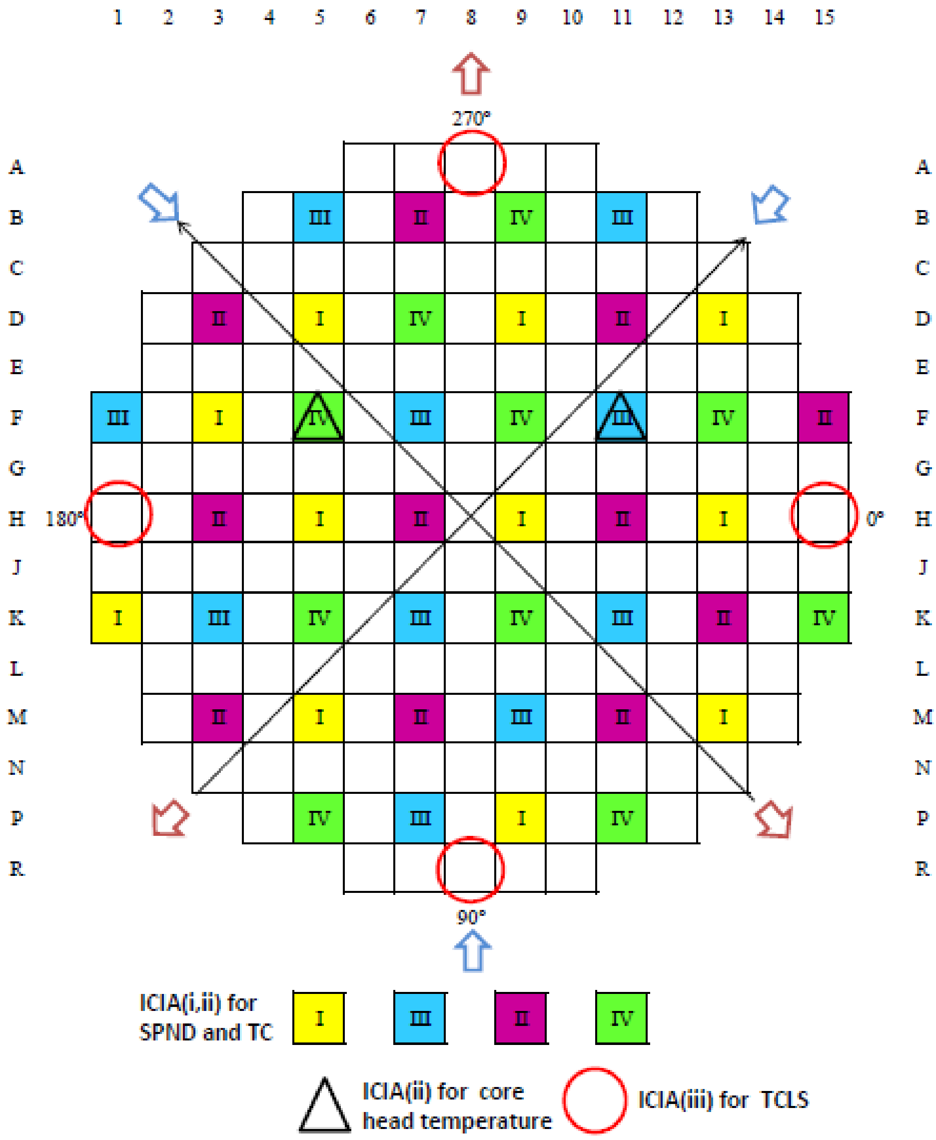


F-5.3-1 Fuel Assembly



F-5.3-2 Rod Cluster Control Assembly

Black and grey rod cluster control assemblies have the same design.



F-5.3-3 In-Core Detectors and Measurement Devices

5.4 Nuclear Core Design

5.4.1 Design Basis for Nuclear Core Design

This section describes the design basis and functional requirements used in the nuclear design of the fuel and reactivity control system.

According to reference [1], reactor core design is the cycle specific calculation of all interactions of the neutrons with the materials in the core relevant for reactor operation. The intent of the calculations is to provide a balance of reloaded and fresh fuel assemblies that will meet the energy requirements for the cycle, and allow the core to operate within safety analysis bounding limits that will be set for the UK HPR1000.

5.4.1.1 Functional Requirement

The nuclear design of the reactor core must ensure reactivity and power can be controlled for DBC-1 and DBC-2 events.

5.4.1.2 Design Bases

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

- Fuel pin power is maintained in an optimised manner across the core to prevent departure from nucleate boiling in DBC-1 or DBC-2 events;
- There is sufficient shutdown margin to bring the core to a safe state under all operating modes and conditions;
- A negative moderator temperature coefficient is obtained for power operating conditions.

To satisfy these requirements, the following design bases have been established for the nuclear design of the reactor core.

5.4.1.3 Fuel Burnup

Fuel burnup is a measurement of the total energy output from the fuel per amount of fuel used and is therefore a useful means to quantify fuel efficiency.

The nuclear design basis is that the maximum discharge burnup level shall not exceed the burnup safety analysis limit determined according to the fuel assembly and fuel rod performance analysis. Batch average discharge burnup has a target level; therefore the core fuel loading must provide sufficient excess reactivity to allow the plant to run for the full cycle. Even though it is not a design basis, the fuel must contain enough initial reactivity excess to maintain the core criticality under full power operation conditions during the whole cycle life with samarium, equilibrium xenon, and other fission products. The end of design cycle life, known as End of Cycle is defined to occur when the boron (chemical shim) concentration is essentially zero (approximately 10ppm) with control rods essentially withdrawn at nominal power.

5.4.1.4 Reactivity Coefficients

For the initial fuel cycle, the fuel temperature coefficient will be negative, and the moderator temperature coefficient of reactivity will be negative for power operating conditions, thereby providing negative reactivity feedback characteristics.

There are two major effects should be taken into account as the compensation for a rapid increase in reactivity: the resonance absorption effects (Doppler) resulting from fuel temperature change and the reactivity effects (variations in spectrum and boron absorption) resulting from moderator density change. These two reactivity coefficients are usually used to represent the basic physics characteristics. The Doppler coefficient of reactivity is negative as a result of the use of slightly enriched uranium and it provides the most rapid reactivity compensation.

The moderator temperature coefficient of reactivity is negative in the core from hot zero power to nominal power, which provides another slower negative feedback effect related to coolant temperature or void content. The fixed burnable absorber and/or control rods limit the soluble boron concentration to achieve this negative feedback.

Only when the burnable absorber content (quantity and distribution) relates to achievement of a non-positive moderator temperature coefficient, it can be considered as a design basis.

5.4.1.5 Control of Power Distribution

The power distribution is controlled so that the operational thermal limits (Maximum peak linear power density and departure from nucleate boiling ratio (DNBR)) are not exceeded either under DBC-1 and DBC-2 conditions.

Calculations of extreme power shapes which affect fuel design limits are performed with proven methods. The conditions under which limiting power shapes are assumed to occur are chosen conservatively with regard to any permissible operating state. In spite of the close consistency between calculations and measurements in peak power, a nuclear uncertainty is implemented in calculation of power distribution. This uncertainty provides margins for the analysis for normal operating states and for anticipated transient states.

5.4.1.6 Maximum Controlled Reactivity Insertion Rate

The maximum reactivity insertion rate due to withdrawal of RCCA or Grey Rod Cluster Assemblies (GRCA) at power, or by boron dilution, is limited. During normal operation at power, the maximum controlled reactivity insertion rate is limited. The maximum reactivity change rate for accidental withdrawal is set such that the peak linear power density and the departure from nucleate boiling ratio limitations are not challenged.

5.4.1.7 Shutdown Margin

An adequate shutdown margin or the potential to provide appropriate negative core reactivity in operating power conditions and shutdown conditions, respectively, is a safety analysis bounding limit.

Two independent reactivity control systems are provided, namely control rods and soluble boron in the coolant.

The control rods are designed to provide sufficient negative reactivity to account for the power effect from full power to zero power and to provide the required shutdown margin. The reactivity addition resulting from power reduction consists of contributions from Doppler temperature defect, variable average moderator temperature, flux redistribution and reduction in void content. In addition, the control rod system provides the minimum shutdown margin under DBC-1 occurrences and is capable of making the core subcritical rapidly enough to prevent fuel damage limits being exceeded, assuming that all RCCA are successfully inserted except the highest reactivity worth RCCA which is calculated with other RCCA inserted.

The Chemical Volume Control System (RCV[CVCS]) provides compensation for the density reactivity and xenon depletion by changes in soluble boron concentration, which enable the achievement and maintenance of cold shutdown. Therefore, shutdown is achieved by both mechanical and chemical poison control systems.

5.4.1.8 Stability

The core has an inherent stability for power oscillations at fundamental mode. Spatial power oscillations within the core is readily detected and suppressed on a constant level of core power output.

5.4.2 Description

5.4.2.1 Nuclear Design Description

The reactor core consists of a specified number of fuel rods which are held in bundles by spacer grids and top and bottom fittings. The fuel rods consist of uranium pellets stacked in a cladding tube plugged and seal welded to encapsulate the fuel. The bundles, known as fuel assemblies, are arranged in a pattern which approximates a right circular cylinder.

The core is composed of 177 fuel assemblies. Each fuel assembly contains a 17 x 17 rod array composed nominally of 264 fuel rods, 24 guide thimbles, and an in-core instrumentation tube. In cold conditions, the active height of the core is 365.76 cm and its equivalent diameter is 323cm giving an H/D ratio of 1.13.

A summary of design parameters is shown in T-5.4-1.

Assemblies with different levels of uranium enrichment are used in the initial core loading to create a balanced power radial distribution. F-5.4-1 shows the uranium fuel loading pattern which is likely to be used in the first core. In the central portion of the core, two zones featuring the assemblies with the two lowest enrichment levels intersect one another to form a checkerboard pattern. The third zone is arranged around this region, at the periphery of the core, and contains the assemblies with the highest enrichment level.

The enrichments by mass of uranium 235 for cycle 1 will be defined later for the regions 1, 2 and 3 respectively of the core. See F-5.4-1.

Fuel assembly parameters are listed in T-5.4-1.

Burnable poison material (gadolinium - Gd_2O_3) is blended within the fuel pellet material itself. This is used to balance power peaking, avoid high soluble boron concentration levels at beginning of cycle (BOC), and ensure a negative moderator temperature coefficient, particularly at BOC. During operation, the poison content in these rods is depleted, thus adding positive reactivity to offset some of the negative reactivity from fuel depletion and fission product build-up.

Besides, the burnable poison is located appropriately to lower the power peaking factor, providing a smoother radial power distribution. Several patterns of burnable poison distributions within a fuel assembly in a 17x17 array are shown in F-5.4-2.

5.4.2.2 Power Distributions

Relative power distributions are quantified in terms of hot channel factors. These factors are a measure of the peak pellet power within the reactor core relative to the average pellet (F_Q) and the total energy produced in a coolant channel relative to the core average channel ($F_{\Delta H}$), and are expressed in terms of parameters related to the nuclear or thermal-hydraulic design.

5.4.2.2.1 Radial Power Distribution

At a given height, the core radial power shape is a function of:

- The loading pattern of fuel assemblies;
- The location of poisoned rods;
- The insertion of rod cluster control assemblies;
- The core burnup;
- The power level and the moderator density;
- The concentration and the distribution of xenon and samarium.

The effect of non-uniform flow distribution is negligible.

As the hot channel position varies during the operation, a single-reference radial design power distribution is selected for departure from nucleate boiling calculations, with the maximum rod integrated power artificially raised to the design value of $F_{\Delta H}$. The reference power distribution is chosen conservatively to concentrate power in one area of the core, and the benefits of flow redistribution are minimized. Assembly powers are normalized to core average power.

5.4.2.2.2 Axial Power Distribution

The axial power profile depends mainly on:

- Core power level
- Core height
- Coolant temperature and flow
- Coolant temperature program as a function of reactor power
- Fuel cycle lifetimes
- Rod bank worth
- Rod bank overlaps

Some signals are detected by in-core and ex-core flux instrumentation. These signals are used for core monitoring during normal operation to measure the average axial power distribution of the core. The axial power distribution is characterized by the Axial Offset (*AO*) or the *ΔI* defined as follow:

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

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

Where Φ_t and Φ_b are the power fraction in the top and bottom halves of the core.

P_r is the ratio of actual to nominal power level.

5.4.2.2.3 Limiting Power Distribution

Fluctuations limiting in the radial power distribution during normal operation have been considered in the fuel management design and the control rod location design.

The control rod worth and insertion are able to adjust the axial power distribution fluctuation.

The axial power distribution is controlled by maintaining the axial offset within a target operating band to limit the axial power oscillation due to xenon. Maintaining the axial offset within a band can minimize xenon transient effects on the axial power shape since the xenon distribution is kept in phase with the power distribution.

5.4.2.2.4 Online Surveillance System

The in-core and ex-core instrumentation provide the monitoring of 3D power distributions required for the Online Monitoring System (KSS[OMS]). KSS[OMS] is for monitoring the parameters of reactor, and not used to protect reactor. The system is useful for the plant operating, and the interface between system and operators is friendly. KSS[OMS] system is not considered in the safety demonstration.

5.4.2.2.5 Testing

An extensive series of physics tests are planned to be performed on the first core. Since not all limiting situations can be created at beginning of life, the main purpose of the tests is to provide a check on the calculation methods used in the predictions for the conditions of the test. Tests performed at the beginning of each reload cycle are undertaken to verify the selected safety-related parameters of the reload design – safety analysis bounding limits.

5.4.2.3 Reactivity Coefficients

The response of the reactor core to changing plant conditions or to operator adjustments made during normal operation, as well as the core response during abnormal or accidental transients depends on the kinetic characteristics. These kinetic characteristics are quantified in reactivity coefficients, and provide a natural damping of the reactor power changes through negative feedback mechanisms. The reactivity coefficients reflect the changes in the neutron multiplication due to varying plant conditions, such as thermal power, moderator and fuel temperatures, coolant pressure. Since reactivity coefficients change during the life of each core/cycle, conservative values of coefficients are employed in transient analysis to determine the response of the plant throughout the life. A set of reactivity coefficients will be calculated across each fuel cycle and compared with safety analysis bounding limits.

5.4.2.3.1 Fuel Temperature (Doppler) Coefficient

The fuel temperature (Doppler) coefficient definition is the reactivity change per effective fuel temperature degree Celsius change, and it is a measure of the Doppler broadening of U-238 and Pu-240 resonance absorption peaks – natural feedback to reduce the reactivity of the fuel if the power increases. Doppler broadening of other isotopes is also considered, but their contribution to the Doppler effect is small. An increase in fuel temperature increases the effective resonance absorption cross-sections of the fuel and produces a corresponding reduction in reactivity.

At a certain point, the effective fuel temperature is no longer equal to the moderator temperature but varies as a function of core power. This effect is taken into account in the Doppler power coefficient. The integral of the Doppler power coefficient as a function of relative power is the Doppler or fuel contribution to the overall power defect.

5.4.2.3.2 Moderator Coefficient

The moderator temperature (density) coefficient definition is the reactivity change per moderator temperature degree change. Both effects of the changes in moderator density and effects of the changes in the temperature are considered. This provides natural feedback to make a reactivity decrease (power decrease as well) of the core if the power increases.

When the coolant temperature rises, the moderator density would decrease. If the moderator coefficient is negative, less neutron are moderated. That the Coolant temperature increase (keeping the density constant) makes a neutron spectrum hardened and then the resonance absorption in U-238, Pu-240 and other isotopes increases. The hardened spectrum leads a decrease in the fission to capture ratio in U-235 and Pu-239 as well. These two effects make the moderator coefficient more negative. In addition, when temperature increases, water density changes more rapidly with temperature. As a result, an increasing temperature leads to a more negative moderator temperature coefficient.

5.4.2.3.3 Power Coefficient

The power coefficient is a combination of moderator temperature effect and fuel temperature effect as the core power level changes. The power coefficient is defined as the reactivity change per percent power change.

As combination of moderator temperature coefficients and fuel temperature coefficients becomes more and more negative with burnup, the power coefficient becomes more negative with burnup..

5.4.2.4 Core Control

5.4.2.4.1 Control Requirements

To establish the required shutdown margin under conditions where a cooldown to ambient temperature is required, according to reference [2], concentrated soluble boron is added to the reactor coolant through operation of the RCV[CVCS], as described in chapter 10.3.

For core conditions including refuelling, the maximum boron concentration is calculated to ensure it is well below the solubility limit. The RCCA are employed to bring the reactor to the shutdown condition.

The ability to accomplish the shutdown for hot conditions is demonstrated by the RCCA reactivity available with an allowance for the worst stuck rod with that required for control and protection purposes. The shutdown margin includes an allowance for analytic uncertainties which assumes the use of Ag-In-Cd RCCA. The largest reactivity control requirement generally occurs at end of cycle (EOC) when the moderator temperature coefficient typically reaches its peak negative value as reflected in the larger power defect.

RCCA provide sufficient reactivity to account for the power defect from full power to zero power and to provide the required shutdown margin. The reactivity addition resulting from power reduction consists of contributions from Doppler effect, moderator temperature, flux redistribution, and reduction in void content.

5.4.2.4.2 Means of Control

Core reactivity is controlled by means of a chemical poison dissolved in the coolant, RCCA, and fuel rods containing burnable poison as described below.

5.4.2.4.2.1 Chemical Poison

Boric acid is used to control the reactivity changes at a relatively low rate associated with:

- The moderator temperature defect from cold shutdown at ambient temperature to the hot operating temperature at zero power
- The transient xenon and samarium poisoning, such as load following or control rods' position changes.
- The excess reactivity required for the compensation of build-up of long-life fission products and fissile inventory depletion effects.
- Burnable poison depletion.

5.4.2.4.2.2 Burnable Poison

Burnable poison provides a solution to partly control the excess reactivity present during the fuel cycle. The moderator temperature coefficient can meet the criterion in this way. As described previously, burnable poison works by reducing the requirement for soluble poison in the moderator at the beginning of the first cycle. The poison in the rods is depleted with burnup at a sufficiently low rate, limiting the concentration of soluble boron, which ensures that the moderator temperature coefficient always meets the criterion.

5.4.2.4.2.3 Rod Cluster Control Assemblies

The number of RCCA is shown in T-5.4-1. The RCCA are used for shutdown and control purposes to offset fast reactivity changes associated with:

- Shutting the reactor down with a largest worth rod stuck in the fully withdrawn position
- The reactivity compensation as a result of an increase in power above hot zero power (power defect, including Doppler and moderator reactivity changes)
- Variations in boron concentration, coolant temperature, or xenon concentration (with rods not exceeding the allowable rod insertion limits)
- Controlling power changes in load following mode

The insertion of RCCA is limited at full power to maintain the required shutdown margin. As the power level is reduced, the shutdown reactivity requirements are also reduced and RCCA insertion limits are reduced. In addition, the RCCA withdrawal pattern determined from these analyses is used in determining the maximum worth of an inserted rod cluster control assembly ejection accident.

5.4.2.5 Control Rod Patterns and Reactivity Worth

RCCA are designated by function as the Control Groups and the Shutdown Groups. The RCCA patterns are displayed in F-5.4-3. The Control Groups are labelled G1, G2, N1, N2 and R. The Shutdown Groups are labelled SA, SB, SC, and SD. RCCA groups labelled G1, G2, N1, and N2 are used for power control. The R sub-group of RCCA are used for temperature control.

Two criteria have been employed for selection of the control groups. First, the total reactivity worth must be sufficient to meet shutdown margin requirements. Second, the limits for power peaking factors (F_Q and $F_{\Delta H}$) must be met for variations in Control Group positions in DBC-1 and DBC-2 conditions as these rods may be partially inserted during power operation.

Control rod velocity and differential reactivity worth are both needed to calculate the control rod reactivity worth versus time following reactor trip. For nuclear design purposes, to be conservative, assume that highest reactivity worth rod is stuck out of the core and the flux is skewed to the bottom of the core when calculating the reactivity worth versus rod position.

5.4.2.6 Criticality of the Reactor during Refuelling

When fuel assemblies are in the pressure vessel and the vessel head is not in place, keff will be maintained at or below 0.95 with control rods and soluble boron. Further, the core will be maintained sufficiently subcritical so that removal of all rod cluster control assemblies will not result in criticality.

5.4.2.7 Stability

The core is designed so that the diametrical and azimuthal oscillations due to spatial xenon effects are self-damping and no operator action or control action is required to suppress them. The stability to diametrical oscillations is so great that it is highly improbable. Convergent azimuthal oscillations due to prohibited motion of individual control rods would be readily observable and alarmed.

Any axial xenon spatial power oscillations that occur in core life are controlled. Assurance that fuel design limits are not exceeded is provided by the Reactor Protection System.

5.4.3 Analysis Methods

The actual analysis methods will depend on the fuel vendor chosen for the UK project. A general description is provided in the following section. The current approach for HPR1000 (FCG3) is described. Similar types of codes to those used at HPR1000

(FCG3) are expected to be used for the UK HPR1000 analysis.

5.4.3.1 Typical Lattice Calculation Code

The lattice calculation code uses a modern multi-group cross section library as input data. This code is able to treat the full assembly, 1/8 and 1/4 assembly geometries (depending on rotational symmetry) and calculate the necessary 2-group cross sections under different operation conditions.

The lattice calculation code provides few-group parameters at different depletion burnups on a unit assembly-wide basis using a neutron transport equation solver.

5.4.3.2 Typical Core Calculation Code

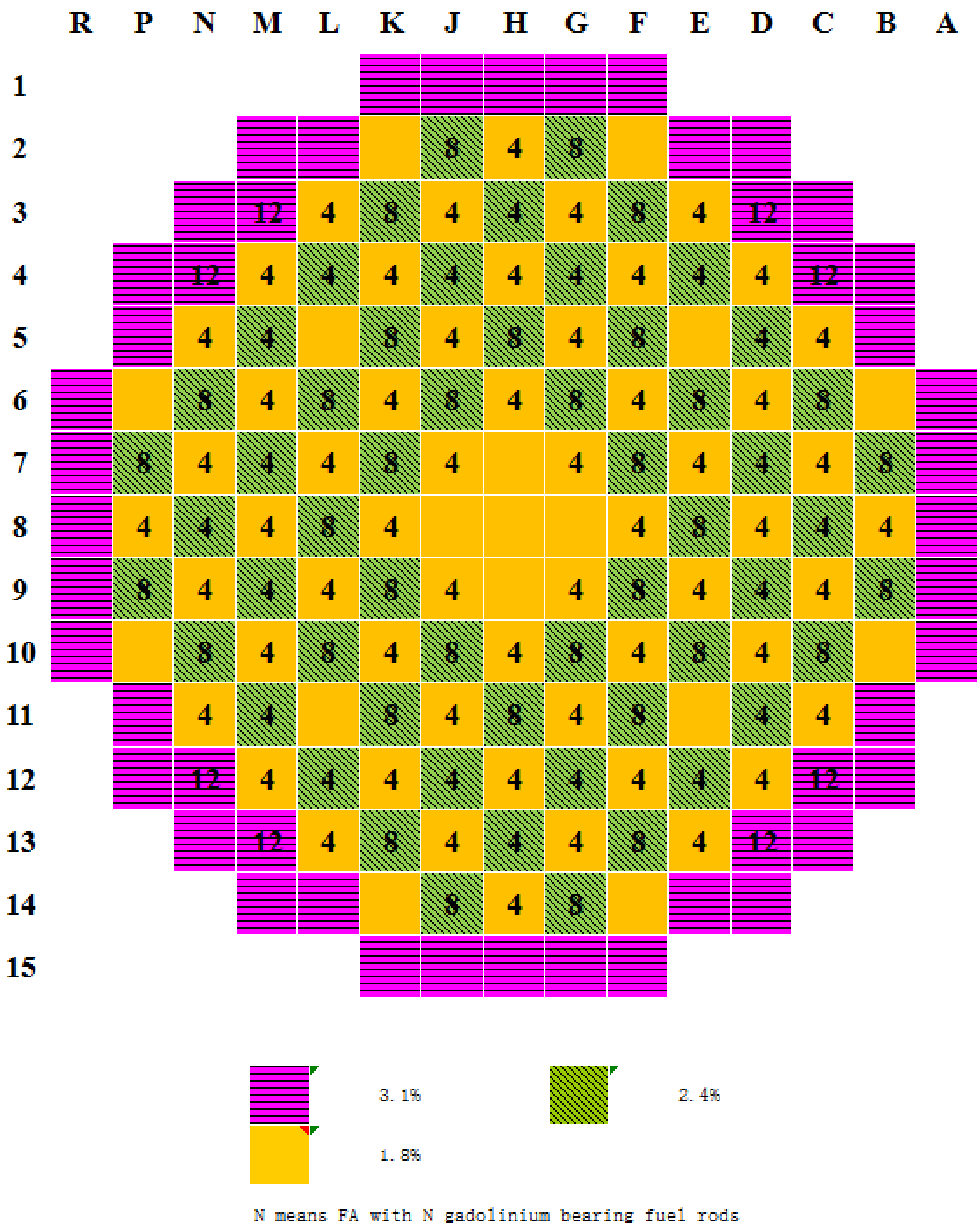
The core calculation is a 3D diffusion based core calculation code which solves 2-group problems. It makes use of the 2-group homogenized cross sections produced by lattice calculations described in 5.4.3.1. The results obtained can then be used in burnup calculations which are able to predict the node burnup and the isotopes densities.

T-5.4-1 Description of the HPR1000 (FCG3) Reactor Core

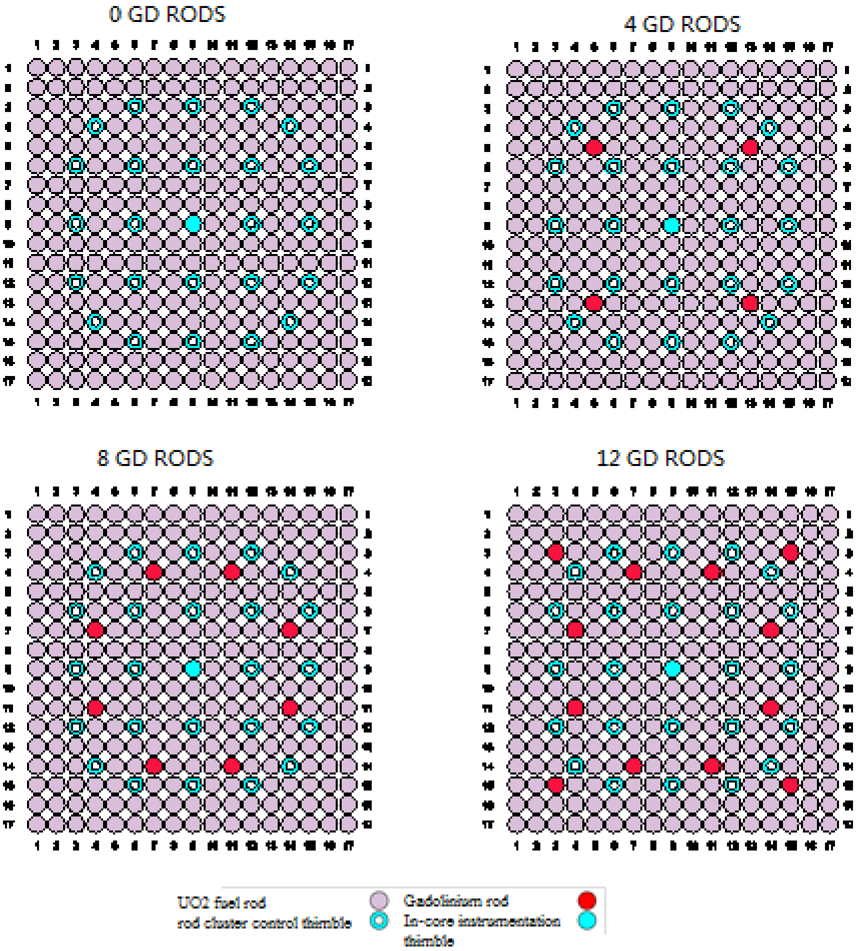
— **Core**	
• Equivalent diameter, cm	323
• Average active height of the core fuel, cm	365.76
• Height/diameter ratio	1.13
• Total surface area, cm^2	81849
— **Fuel assemblies (cold dimensions):**	
• Number	177
• Rod array	17×17
• Number of rods per assembly	264
• Lattice pitch, cm	1.26
• Overall dimensions of assembly, cm	21.4×21.4
• Composition of grids	Zirconium alloy & Inconel
• Number of guide thimbles per assembly	24
• Composition of the guide thimbles	Zirconium alloy
— **Fuel rods (cold dimensions):**	
• Number	46728
• Outside diameter, mm	9.5
• Diametrical gap, mm	0.17
• Thickness of the cladding, mm	0.57
• Cladding material	Zirconium alloy

— **Fuel pellet:**	
• Material	Sintered UO_2
• Density of the UO_2 (% of theoretical density)	95
• Enrichment of fuel for the UO_2 assemblies (% by weight U-235, first cycle)	
Zone 1 of cycle 1	Approximately 1.8%
Zone 2 of cycle 1	Approximately 2.4%
Zone 3 of cycle 1	Approximately 3.1%
— **Neutron Absorber:**	
• Neutron Absorber Rod Composition	Ag-In-Cd
• Composition (% by weight)	80% Ag, 15% In and 5% Cd
• Cladding material	316 type stainless steel
— BLACK RCCA	
• Number	56
• Number of absorber rods per black RCCA	24
— GREY RCCA	
• Number	12
• Number of absorber rods per grey RCCA	8

Values are approximate.



F-5.4-1 Generic Core Pattern and Distribution of Burnable Poison Rods in the First Cycle for HPR1000 (FCG3)



F-5.4-2 Distribution of Burnable Poison Rods in a Fuel Assembly in the First Cycle for HPR1000 (FCG3)

	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															

F-5.4-3 Rod Cluster Control Assembly Pattern

5.5 Thermal Hydraulic Design

5.5.1 Functional Requirement

The thermal and hydraulic design of the reactor core must ensure the removal of heat produced in the fuel via the coolant fluid for all design bases conditions and ensure containment of radioactive substances for DBC-1 and DBC-2 events.

5.5.2 Design Bases

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

−Fuel damage (defined as penetration of the fission product barrier; that is, the fuel rod cladding) is not expected during normal operation and operational transients (DBC-1) or any anticipated operational occurrences (DBC-2). It is possible, however, that a very small number of fuel rod failures could occur. These are within the capability of the plant cleanup system and are consistent with the plant design bases.

−The reactor can be brought to a safe state following a DBC-3 event with only a small fraction of fuel rods damaged (as defined in the above definition), although sufficient fuel damage might occur to preclude resumption of operation without considerable outage time.

−The reactor can be brought to a safe state and the core can be kept subcritical with acceptable heat transfer geometry following transients arising from DBC-4 events.

To satisfy these requirements, the following design bases have been established for the thermal and hydraulic design of the reactor core.

5.5.2.1 Departure from Nucleate Boiling Design Basis

There is at least a 95% probability that departure from nucleate boiling (DNB) will not occur on the limiting fuel rods during normal operation and operational transients and any transient conditions arising from faults of moderate frequency (DBC-1 and DBC-2 events), at a 95% confidence level.

By preventing DNB, adequate heat transfer from the fuel cladding to the reactor coolant can be ensured, thereby cladding damage because of inadequate cooling can be prevented. 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 nuclear control and protection systems are such that this design basis is met for transients associated with DBC-1 and-2 events, including overpower transients (See chapter 12).

Margin with regard to departure from nucleate boiling is known as the Departure from Nucleate Boiling Ratio (DNBR) and is calculated as follow:

$$DNBR = \frac{\varphi_{critical}}{\varphi_{loc}}$$

Where:

ϕ_{loc} is the actual local heat flux;

$\phi_{critical}$ is the critical heat flux predicted by the Critical Heat Flux (CHF) correlation.

Using sub-channel code and the CHF correlation applicable to the fuel type to be chosen, DNBR of different transients are calculated. The minimum calculated DNBR must be greater than the DNBR design criteria to ensure fuel integrity.

For most accidents, DNBR criteria are defined using the correlation applicable to the chosen fuel and a statistical design method in which several uncertainties including the correlation uncertainty and a number of plant thermal hydraulic parameter uncertainties are combined statistically. These criteria are valid for a given range of thermal hydraulic conditions. For accidents where limiting thermal hydraulic conditions are outside the validity domain of the method, the minimum calculated DNBR is compared to the deterministic correlation criteria.

5.5.2.2 Fuel Temperature Design Basis

During DBC-1 and DBC-2 events, there is at least a 95% probability at a 95% confidence level that the peak linear power (W/cm) in the fuel rods will remain below the level that could raise fuel to tis melting temperature.

5.5.2.3 Core Flow Design Basis

A minimum value of thermal hydraulic design flowrate pass through the fuel rod region of the core is assumed to 93.5%, and this is effective for fuel rod cooling.

Core cooling evaluations are based on the thermal hydraulic flowrate (minimum flow) entering the reactor vessel. 6.5% of the flowrate is taken as the maximum bypass flowrate. This includes rod cluster control guide thimble and instrumentation tube cooling flow, leakage between the core barrel and the core shroud, core peripheral assemblies bypass, head cooling flow, and leakage to the vessel outlet nozzles.

5.5.2.4 Hydrodynamic Instability

Modes of operation associated with DBC-1 and DBC-2 events do not lead to hydrodynamic instability

5.5.3 Description of the Thermal and Hydraulic Design of the Reactor Coolant System

5.5.3.1 Plant Configuration Data

Plant configuration data for the thermal-hydraulic and fluid systems external to the core are provided as appropriate in chapters 6, 7, and 10.

5.5.3.2 Operating Restrictions on Pumps

The minimum net positive suction head (NPSH) and minimum seal injection flowrate must be established before operating the reactor coolant pumps. With the minimum seal injection flowrate established, the operator has to verify that the system pressure satisfies NPSH requirements.

5.5.3.3 Temperature-Power Operating Map

The relationship between reactor coolant system temperature and power is a linear relationship between zero and 100% power.

5.5.3.4 Load-Following Characteristics

The reactor coolant system is designed on the basis of steady-state operation at full power heat load. This power level is, for that design, the most conservative of all possible level allowed by the load follow operation.

5.5.3.5 Thermal and Hydraulic Characteristics Summary Table

The thermal-hydraulic parameters of the HPR1000 (FCG3) are given in T-5.5-1.

5.5.4 Design Evaluation

5.5.4.1 Critical Heat Flux

The critical heat flux correlation used in the core thermal analysis is explained in sub-chapter 5.5.2.1.

For evaluating the DNBR margin either a deterministic method or a statistical method is considered in conjunction with the fuel assembly specific correlation. Several uncertainties are then statistically treated to obtain a DNBR uncertainty factor. Applying this factor leads to a statistical design DNBR limit.

5.5.4.2 Core Hydraulics

5.5.4.2.1 Core and vessel pressure drops

Viscous drag (friction) and/or geometry changes (form losses) in the fluid flow path can cause the occurrence of unrecoverable pressure losses. The flow field is assumed to be incompressible, turbulent, single-phase water. These assumptions apply to calculate the core and vessel pressure drop , the purpose is to establish the Reactor Coolant System loop flow rate. Two-phase considerations are neglected in the vessel pressure drop evaluation because the core average void is negligible.

Two phase flow considerations are however considered in the core thermal sub-channel analyses. Pressure losses in the core and vessel are calculated by the following equations:

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

Where:

ΔP_L : Unrecoverable pressure drop, MPa

ρ : Fluid density, kg/m^3

L : Length, m

D_e: Equivalent diameter, m

V : Fluid velocity, m/s

K : Form loss coefficient, dimensionless

f : Friction loss coefficient, dimensionless

The core pressure drop characteristics were determined by hydraulic tests of 17 x 17 fuel assemblies' behaviour. These tests were carried out in a thermal hydraulic test loop under a large range of Reynolds numbers, which cover the range of HPR1000 (FCG3) core.

The local loss component of the vessel pressure drops were obtained from scale model hydraulic test data on a number of vessels, and from loss relationships.

5.5.4.2.2 Bypass Flow

The maximum or minimum design value of the core bypass flow is used in the core thermal hydraulic design in a conservative method, which is described in sub-chapter 5.5.2.3.

5.5.4.2.3 Inlet Flow Distributions

The inlet flow distribution is generally non-uniform. Investigations with sub-channel codes 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 so that the impact of inlet flow misdistribution on the hot channel DNBR is negligible. This flow redistribution is due to the redistribution of fluid velocities.

5.5.4.3 Influence of Power Distribution

The core power distribution, which is largely determined by fuel enrichments, burnable poison loading, loading pattern, and core power level at beginning of cycle, is also a function of variables such as control rod worth and position, and fuel depletion through lifetime. Radial power distributions in various planes of the core are often illustrated for general interest. However, the core radial enthalpy rise distribution, which is determined by the integral of power up each channel, is of greater importance to DNBR analyses.

These power distributions are characterized by $F_{\Delta H}^{N}$ and axial power profiles.

In a core with N fuel rods and height H, with the local power density q'(W/cm) at a point (x, y, z) known, the nuclear enthalpy rise hot channel factor is given by:

$$F_{\Delta H}^{N} = \frac{hot\ rod\ power}{average\ rod\ power} = \frac{max_{over(x,y)} \int_0^H q'(x_0, y_0, z)\,dz}{\frac{1}{N}\sum_{all\ rods}\int_0^H q'(x_0, y_0, z)\,dz}$$

Where:

x_0, y_0 are the position coordinates of the hot rod.

For a single fuel rod, the location of minimum DNBR depends on the axial profile, and the value of DNBR depends on the enthalpy rise to that point. Basically, the maximum value of the rod power integral is used in order to identify the most possible rod for minimum DNBR. An axial power profile is obtained and, this profile normalized to the value of $F_{\Delta H}^{N}$, creates again the axial heat flux along the limiting rod. The surrounding rods are assumed to have the same axial profile with average rod power, which are typical distributions found in hot assemblies. By this way, the worst case of axial profiles can be combined with the worst case of radial distributions for reference DNB calculations.

For operation at a fraction *P* of full power, the design $F_{\Delta H}^{N}$ used in DNB calculations is given by the following relationship which has been used in HPR1000 (FCG3).

$$F_{\Delta H}^{N} = F_{\Delta H\,0} \times (1 + 0.3 \times (1 - P))$$

The value of $F_{\Delta H\,0}$ in HPR1000 (FCG3) is listed in T-5.5-1.

Based on analysis, axial power shape which results in a calculated minimum DNBR is chosen.

5.5.4.4 Core Thermal Response

As stated in Sub-chapter 5.5.2, the design bases of the application are to prevent DNB and to prevent fuel melting for DBC-1 and DBC-2 events. The protection systems described in chapter 8 are designed to meet these bases. The response of the core to DBC-2 transients will be given in chapter 12 of the GDA PCSR.

5.5.4.5 Analytical Methods

5.5.4.5.1 Core Thermal Hydraulic Analysis

The objective of reactor core thermal-hydraulic design is to determine the maximum heat removal capability in all flow sub-channels and shows that the core safety limits are not exceeded with the consideration of hydraulic and nuclear effect. The thermal-hydraulic design considers local variations in dimensions, power generation, flow redistribution and mixing.

A sub-channel code for core modelling will be used with the applicable DNB correlations to determine DNBR distributions along the hot channels of the reactor core under all DBC conditions. The details of this code will be described later in GDA.

5.5.4.5.2 Hydrodynamic and Flow Power Coupled Instability

Two specific types of flow instabilities are considered for reactor operation. These are the Ledinegg (or flow excursion) type of static instability and the density wave type of dynamic instability.

Ledinegg instability involves a sudden change in flowrate from one steady-state to another. The criterion for stability is $(\partial \Delta p / \partial G)_{\text{internal}} \geq (\partial \Delta p / \partial G)_{\text{external}}$. The pump head curve has a negative slope, whereas the reactor coolant system pressure drop-flow curve has a positive slope over the DBC-1 and DBC-2 operational ranges. Thus, the Ledinegg instability will not occur.

A simple method has been developed by Ishii for parallel closed channel systems to evaluate whether a given condition is stable with respect to the density wave type of dynamic instability.

The application of this method to HPR1000 (FCG3) reactors indicates that a large margin to density wave instability exists.

The method of Ishii applied to HPR1000 (FCG3) reactor design is conservative due to the parallel open channel feature of HPR1000 (FCG3) PWR cores. For such cores there is little resistance to lateral flow leaving the flow channels of high power density. There is also energy transfer from channels of high power density to lower power density channels. This coupling with cooler channels has led to the opinion that an open channel configuration is more stable than the above closed channel analysis under the same boundary conditions.

5.5.4.6 Testing and Verification

5.5.4.6.1 Tests Prior to Initial Criticality

Reactor coolant flow tests are performed following fuel loading, at several power levels after first criticality. This data allows determination of the coolant flow rates at reactor operating conditions. This test verifies that proper coolant flow rates have been used in the core thermal and hydraulic analysis. Subsets of these analyses are repeated for each cycle.

5.5.4.6.2 Initial Power and Plant Operation

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

5.5.4.6.3 Component and Fuel Inspections

Inspections performed on the manufactured fuel are described in Sub-chapter 5.3.4. 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.

T-5.5-1 Reactor Thermal and Hydraulic Characteristics of HPR1000 (FCG3)

Design parameters	**HPR1000 (FCG3)**
Reactor power output, MW	3150
Heat generated in fuel, %	97.4
System pressure, nominal, MPa	15.5
System pressure, minimum, MPa	15.3
$F_{\Delta H}^{N}$ (Design Limit)	1.65
Coolant flow	
Total thermal hydraulic design flowrate, m^3/h	72000
Effective flowrate for heat transfer, m^3/h	67320
Effective flow area for heat transfer, m^2	4.33
Average velocity along fuel rods, m/s	4.32
Coolant temperature	
Nominal inlet temperature, ℃	288.6
Average temperature rise in core, ℃	39.1
Heat transfer	
Active heat transfer surface area, m^2	5094.7
Average heat flux, W/cm^2	60.22
Maximum heat flux for normal operation, W/cm^2	147.54
Average linear power, W/cm	179.5
Peak linear power for normal operation, W/cm	439.8
Peak linear power resulting from overpower transients, W/cm	≤ 590
Peak linear power for prevention of centreline melt, W/cm	700
Power density, kW/l（hot dimension）	102.5
Fuel centre temperature	
Peak at peak linear power for prevention of centreline melt,℃(Design Limit)	2590

Values are approximate.

5.6 References

[1] NNSA, Design for Reactor Core Safety in Nuclear Power Plants, HAD 102.07, 1989.

[2] NNSA, Safety regulations for Design of Nuclear Power Plants, HAF 102, 2004.