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UK ABWR

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UK ABWR Generic Design Assessment

Generic PCSR Chapter 11 : Reactor Core

Hitachi-GE Nuclear Energy, Ltd.

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Executive Summary

This chapter describes the safety case for the UK ABWR fuel and core components. It combines system descriptions of fuel assemblies, control rods and the core monitoring system with evaluations of these against fuel and core design limits that are defined in the chapter. It lists the high level Safety Functional Claims that are made on the core components, and notes that no Safety Property Claims have been identified for the fuel and core.

The information provided includes: system design; functionality in normal operation and during faults; safety categorisation and classification; nuclear design limits; thermal-hydraulic design limits; evaluation methods; safety case Assumptions, Limits and Conditions for Operation; and compliance with the As Low As Reasonably Practicable (ALARP) principle. The chapter includes the evaluation of the initial core and equilibrium cores loaded with GE14 fuel.

The overall PCSR justification that the UK ABWR is safe and satisfies the ALARP principle is underpinned by hazards assessments, design basis analysis, probabilistic safety analysis, beyond design basis analysis and human factors analysis (described in PCSR Chapters 6, 7 and 24 to 27), which demonstrate that the design of the systems covered by this chapter are fault tolerant. In particular some of the fuel and core design limits specified in this chapter are substantiated by analysis presented in PCSR Chapter 24: Design Basis Analysis.

These analysis chapters specify the high level safety functional claims and apply analysis conditions and assumptions that are based on, and fully consistent with, the design information, design limits, and safety claims for the fuel and core that are presented in this chapter, in order to substantiate those claims.

The designs of the fuel and core components are well advanced for GDA, being largely based on proven technology from operating BWRs. Additional risk reduction measures have been introduced (with reference to the J-ABWR design) in response to safety assessments undertaken in GDA, and these include an enhancement to the countermeasures for core instability as described in PCSR Chapter 28.

This chapter demonstrates that the risks associated with the design and operation of the fuel and core components for the UK ABWR are ALARP, taking account of how ABWR fuel and core component designs have evolved. It is acknowledged that further work will be required post-GDA to develop the design and fully incorporate site specific aspects. In particular, the adoption of GE14 fuel in the UK ABWR design reference does not foreclose on any future use of a fuel design which is different to that assumed in GDA. Fuel developments are not expected to undermine the high level conclusion of this chapter.

11.1 Introduction

This chapter of the PCSR describes the design features, design limits and evaluation of the reactor core for UK ABWR. It lists all of the Safety Functional Claims (SFCs) that are made on fuel and core components and on the core loading patterns to maintain the high level safety functions (HLSFs) during normal operation and design basis fault conditions. It also explains how compliance of the fuel and core with the Nuclear Safety and Environmental Design Principles (NSEDPs – [Ref-11-1]) is demonstrated.

11.1.1 Background

The fuel system consists of the fuel assemblies and control rods.

The design information in this chapter is generally substantiated by evaluations against a set of fuel and core design limits. Those evaluations are presented in supporting documents that are referred to by the chapter itself. However this chapter is also a source of input data used in various other assessments and analyses that substantiate safety of operation of the UK ABWR safety case, which are reported in other PCSR chapters. In particular the design basis analysis, probabilistic safety analysis, beyond design basis analysis and human factors analysis, described in Chapters 24 to 27 of the PCSR, demonstrate that the reactor and support systems are fault tolerant. The assumptions made in those analyses are fully consistent with the design information and safety claims for the fuel and core that are presented in this chapter.

11.1.2 Document Structure

The following sections are included in this chapter:

Section 11.2 Purpose and Scope: this section explains the objectives of the chapter, and lists the main core components and design limits that are within its scope.

Section 11.3 Summary Description: this section provides a brief overview of the designs of the ABWR reactor system, core and fuel. It points to PCSR Chapter 12 for further details of the reactor system design.

Section 11.4 Design Basis: this section describes all of the SFCs that are claimed for fuel assemblies, control rods and core monitoring system. It also describes SFCs claimed for the overall characteristics of the core loading configurations in terms of nuclear design and thermal-hydraulic design limits.

Section 11.5 Design Description: this section provides further design details to supplement Section 11.3. The scope of components described in this section is defined in sub-section 11.2.2 below.

Section 11.6 Design Evaluation: this section provides pointers to the PCSR supporting documents that provide the arguments and evidence to substantiate the SFCs from section 11.4.

Section 11.7 Assumptions, Limits and Conditions for Operation: this section summarises the assumptions, limits and conditions for operation for the fuel and core that are specified in greater detail in the Operating Rules document for the Structures, Systems and Components (SSCs) that are in the scope of this Chapter.

Section 11.8 Summary of ALARP Justification: this section provides a summary of the justification that the risks associated with operation of GE14 fuel in the UK ABWR core are acceptable (in terms of radiation dose consequences) and have been reduced to levels that are As Low as Reasonably Practicable (ALARP).

Section 11.9 Conclusions: this section provides a summary of the main aspects of this chapter, including a review of the results of the evaluations against design limits.

Other relevant information is captured in Appendices as follows:

Appendix A - Safety Functional Claims Table for the Reactor Core

Appendix B - Document map for Level 2 documents that support this chapter

(Unlike the PCSR systems chapters, there are no SPCs (Safety Properties Claims) identified for the fuel and core, so no SPC claims table is required).

This chapter is supported by a set of reference documents including computer code descriptions and reports describing design details, assessment methodologies, and results of fuel and core evaluations. It is also supported by several external technical documents that are publically available. A full list of the Level 2 documents is provided within the document map in Appendix B.

The main links of this chapter with other GDA PCSR chapters are as follows:

- The high level safety functions which are fulfilled by meeting the fuel and core design limits are defined in Chapter 5: General Design Aspects.

- Chapter 12: Reactor Coolant Systems, Reactivity Control Systems and Associated Systems provides additional information pertaining to the description and associated safety claims for the Standby Liquid Control and Control Rod Drive systems.
- Design and Safety related aspects associated to the handling and storage of the fuel, control rods and other items of in-core equipment are covered in Chapter 19: Fuel Storage and Handling.
- Chapter 23: Reactor Chemistry evaluates the impact of Water Chemistry in the reactor on the safety functions provided by the SSCs that are in the scope of this chapter.
- Demonstration, using transient analysis, of the adequate performance of SSCs within this chapter scope during design basis events and beyond design basis events is covered in Chapters 24: Design Basis Analysis and Chapter 26: Beyond Design Basis and Severe Accident Analysis.
- Probabilistic analysis that demonstrates adequate reliability of SSCs within this chapter scope is in Chapter 25: Probabilistic Safety Assessment.
- Substantiation of Human Based Safety Claims related to human interactions with SSCs within this chapter scope is described in Chapter 27: Human Factors.
- Safety case aspects pertaining to the interim term storage of spent fuel are addressed at a conceptual design level in Chapter 32: Spent Fuel Interim Storage.
- For generic links to GEP, and CSA documentation, please refer to Generic PCSR Chapter 1: Introduction GA91-9101-0101-01000 (XE-GD-0214). For GEP, where specific references are required, e.g. in Radioactive Waste Management, Radiation Protection, Decommissioning, these will be included in the specific sections within the relevant chapter.

11.2 Purpose and Scope

11.2.1 Purpose

The overall objective of this chapter is to provide a complete source of design information for the fuel assemblies, control rods, including relevant design limits. In addition, functional claims made on those components in the UK ABWR safety case are identified and substantiated as far as is possible with the current design maturity at the conclusion of GDA.

Specific objectives of the chapter are:

- Identify and describe the SSCs that comprise the fuel and core,
- Summarize the processes involved in the fuel and core design,
- State the Safety Functional Requirements for the fuel and core,
- Substantiate the safety of the generic core designs presented,
    - Specify all relevant Safety Case Claims, and describe or provide pointers to where the detailed arguments and evidence for each claim can be found in the supporting Topic Reports, external documents, and the detailed Level 3 documents.
- Demonstrate confidence in the substantiation of the future safe operation of the fuel and core,
- Present the characteristics of the initial core and equilibrium core designs,
- Identify the main links to relevant content of other GDA PCSR chapters, to ensure consistency across the whole safety case, and to ensure the overall safety case presented in the GDA PCSR is complete,
- Provide or identify references to relevant evidence that demonstrates that the risks associated with operation of the SSCs within the scope of this chapter are As Low As Reasonably Practicable (ALARP), and
- Identify aspects of the design substantiation of SSCs within the scope of this chapter that require further work beyond completion of GDA.

11.2.2 Scope

The scope of the reactor and core components included in this chapter is as follows:

Detailed design descriptions and design evaluations for:

- Fuel assemblies (fuel bundle and fuel channel),
- Boron carbide and hafnium control rods, and
- Core monitoring system.

Brief overview descriptions and details of design evaluations for:

- Control Rod Drive (CRD) systems, and
- Standby Liquid Control (SLC) system.

The plant conditions considered are:

- All normal operational states,
- Frequent design basis faults, and
- Infrequent design basis faults.

The chapter considers the initial core and equilibrium cores loaded with GE14 fuel.

This Chapter does not cover:

- Details of core support structures design – covered in Chapter 12: Reactor Coolant Systems, Reactivity Control Systems, and Associated Systems), and
- Details of the Control Rod Drive (CRD) systems and Standby Liquid Control (SLC) system design – covered in Chapter 12.

11.3 Summary Description

11.3.1 ABWR Reactor System

A summary description of the Advanced Boiling Water Reactor (ABWR) reactor system including core and fuel is presented in this section.

The ABWR is a forced recirculation direct cycle reactor using light water as a coolant and moderator. The reactor is equipped with an internal steam separator and 10 Reactor Internal Pumps (RIPs) which ensure the recirculation of the coolant. The rated thermal power is 3,926 MW. The major reactor safety systems (Safety Relief Valves (SRVs), Emergency Core Cooling System (ECCS), Primary Containment Vessel (PCV), etc.) are designed for a thermal power of 4,005 MW (about 102 percent of the rated power).

The ABWR reactor system consists of the following main components: Reactor Pressure Vessel (RPV), Reactor Internals, Reactor Core, Control Rods (CRs), Fine-Motion Control Rod Drive (FMCRD). Figure 11.3-1 shows an overview of these components. The RPV contains the core support structure, reactor internals, steam separator, steam dryer and other internals.

The reactor core consists of 872 Fuel Assemblies (FAs) and 205 CRs. A detailed description is provided in the following sections of this chapter. The core is surrounded by the core shroud which is designed to separate the coolant that flows upward through the core from the coolant that flows downward in the annular area between the core shroud and the RPV wall.

The core shroud is a stainless steel cylinder and is supported by a shroud support and a shroud support leg.

With regard to FA placement, the core can be divided into two regions, i.e., central and peripheral areas. One set of four FAs in the core interior area is supported by a centre fuel support piece installed on top of a CR guide tube. The FAs along the outer core peripheral area are supported by peripheral fuel support pieces located on the top of the core plate.

The top of the FA is supported in the lateral direction by a top guide, which is also supported by the core shroud.

Ten RIPs located at the bottom of the reactor pressure vessel penetrate the shroud support plate. The motor of the RIP is housed within a motor casing and drives the pump impeller through a shaft. By using pumps attached directly to the RPV, jet pumps and the external recirculation system that can be found in many existing BWRs (i.e., recirculation pumps, piping, valves, etc.) have been entirely eliminated. Since there is no large external piping attached to the vessel below the top of the core, the ABWR core remains submerged and cooled, following a pipe rupture and initiation of ECCS.

The coolant from the RIPs is distributed to each FA through the core lower plenum and into the fuel support orifice. The coolant is then heated as it moves upward through the assembly, and exiting the core as a two-phase mixture of steam and water.

The steam-water mixture from the core enters the upper core plenum where additional mixing occurs, and subsequently enters an array of stand pipes that form the steam separator.

A centrifugal effect within the steam separator removes the water from the mixture. The steam enters the steam dryer where the remaining moisture is extracted. This dry steam exits the reactor pressure vessel through the four main steam lines. The water extracted by the steam separator and the steam dryer is discharged into the annular region, mixed with the feedwater, and then is driven back into the lower core plenum by the RIPs.

The ABWR reactor system described above, including RPV, reactor internals and Control Rod Drive (CRD), is described in Generic PCSR Chapter 12: Reactor Coolant Systems, Reactivity Control Systems and Associated Systems. The reactor is equipped with a Neutron Monitoring System (NMS) that monitors reactor power over the full range of operation, from start up to rated power, and is described in [Ref-11-2].

Table 11.3-1 shows the major specifications of the reactor and the core. Figure 11.3-1 shows the overview of the ABWR reactor.

Table 11.3-1 : Major Specifications of Reactor and Core

Reactor thermal power	3,926 MW
Reactor coolant recirculation pump	10 units
Core flow rate*	52.2×10^3 t/h
Core inlet sub-cooling*	54.0 kJ/kg
Average steam quality at core outlet*	14.6 wt%
Reactor pressure (Pressure vessel at dome)*	7.17 MPa [abs]
Core Effective height*	3.81 m
Equivalent diameter*	5.16 m
Steam flow rate*	7.64×10^3 t/h
Steam pressure*	7.17 MPa [abs]
Steam temperature*	287 °C

*Values are approximate.

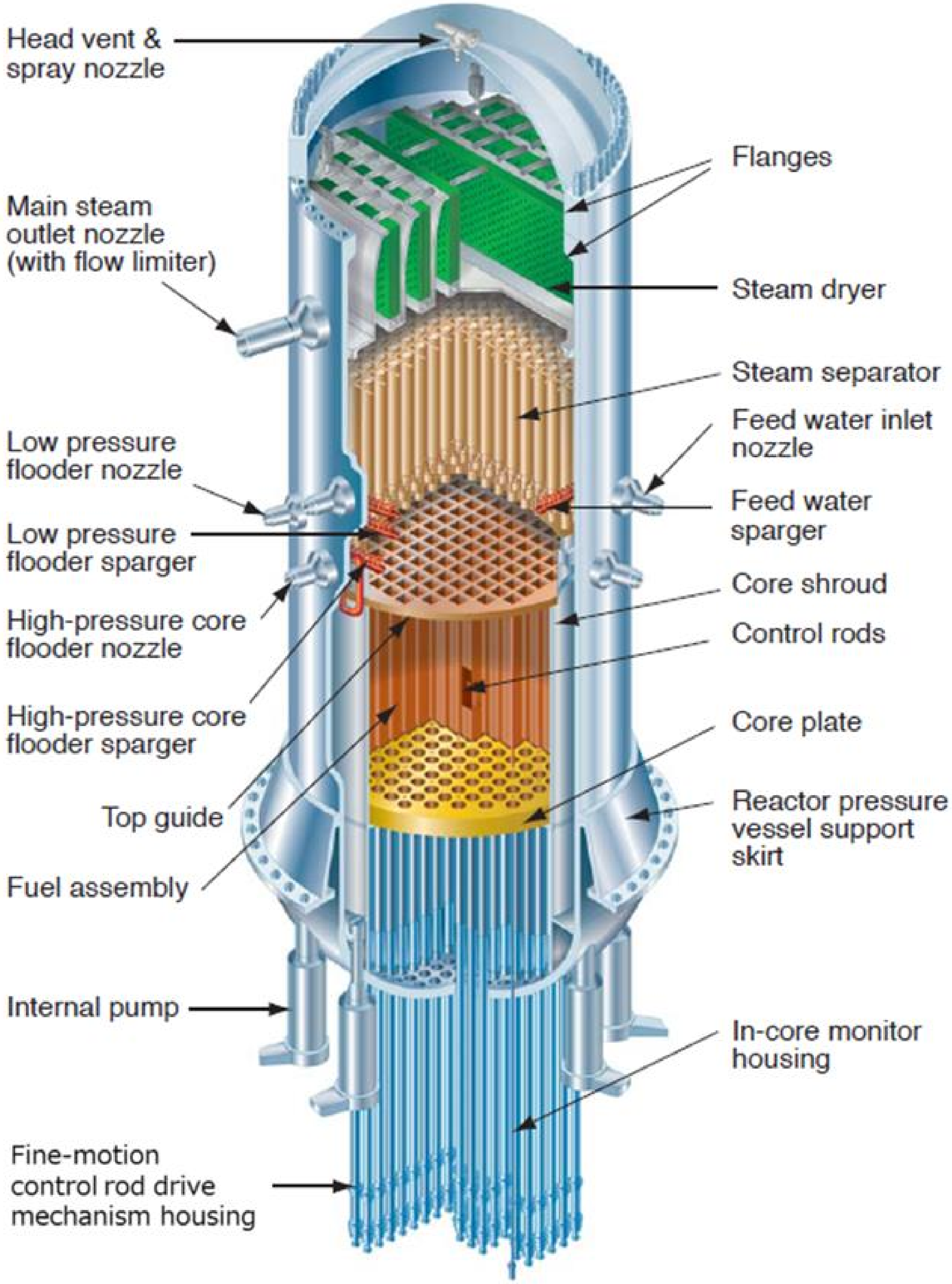


Figure 11.3-1 : Overview of ABWR Reactor

11.3.2 Core and Fuel

The ABWR core has a right cylindrical shape that is approximately 3.8 m in height with an equivalent diameter of about 5.2 m. It consists of 872 FAs and 205 CRs. Figure 11.3-2 shows the core configuration.

The ABWR core is light-water moderated and fuelled with slightly enriched uranium-dioxide. The use of water as a moderator produces a neutron energy spectrum in which fissions are induced mainly by thermal neutrons (kinetic energy $\leq$ 0.025 eV). During normal operation the moderator boils producing a spatially variable distribution of steam voids in the core. In the ABWR the fission rate is reduced by an increase in the steam void content in the moderator. Therefore any system input that increases the reactor power, either in a local or global sense, produces additional steam voids that reduces the reactivity and therefore the power. This void feedback effect is one of the inherent safety features of the ABWR system.

The fuel system consists of FAs and CRs. The FA consists of a fuel bundle and a fuel channel that surrounds the fuel bundle. The fuel bundle consists of fuel rods, water rods, expansion springs, spacers, and upper and lower tie plates. The selected fuel design utilises a 10×10 fuel rod array that includes 78 full length fuel rods, 14 part length fuel rods and 2 large central water rods. The cast stainless steel lower tie plate includes a conical section that seats in the fuel support and a grid that establishes the proper fuel rod spacing at the bottom of the bundle. The lower tie plate also houses a debris filter to prevent debris from entering the assembly and potentially leading to fretting failure of the fuel rod cladding. The cast stainless steel upper tie plate maintains the fuel rod spacing at the top of the bundle and provides the handle that is used to lift the bundle for transferring the fuel bundle from one location to another. Finger springs located between the lower tie plate and fuel channel are utilised to maintain a nearly constant leakage flow into the region outside the FAs, named the bypass region. Another contributing source to the bypass flow is provided by two lower tie plate holes in each FA that, in total, provides sufficient cooling for plant instrumentation located outside the fuel channels. The entire fuel bundle is held together by 8 threaded tie rods located around the periphery of the bundle. Another key component of the bundle are the 8 spacer grids, which have the function to maintain proper spacing between the fuel rods along the axial length of the bundle as well as influencing critical power performance.

The main materials used for the FA are zircaloy, zirconium, stainless steel, inconel, ceramic UO_2 and Gd_2O_3. Material characteristics such as mechanical properties and corrosion resistance, shall conform to the conditions of the boiling water reactor and satisfy the design intent of the reactor during operation.

The CR is inserted into and withdrawn from the core through a CR guide tube located within the lower plenum of the reactor. Each CR is connected to a fine-motion CRD by a mechanical coupling.

Local Power Range Monitors (LPRMs) are provided for monitoring core neutron flux at power operation conditions. They are in-core fission chambers that are assembled and fixed inside the enclosing tubes located in the core. These instrument assemblies provide signals for continuous local power range neutron flux monitoring. Start-up Range Neutron Monitors (SRNMs) are provided for monitoring core neutron flux at low power conditions. Neutron Sources are required for SRNM at the initial core. The locations of LPRM, SRNM and Neutron Source are shown in Figure 11.3-2. Further details are given in [Ref-11-2].

New fuel bundles are transferred into the Reactor Building (R/B). They are raised to the operations deck, assembled into FAs, stored in the Spent Fuel Storage Pool (SFP) and then loaded into the reactor. The irradiated FAs are relocated (also known as shuffling) within the reactor core at an outage for the subsequent cycle, and after the multiple cycle irradiation, they are transferred for storage in the SFP racks. After a sufficient period of cooling, spent FAs are loaded into casks and transferred for storage in a Spent Fuel Interim Storage (SFIS) facility. Records are kept for all the processes described. Further details of the fuel route and SFIS are given in Generic PCSR Chapter 19 (Fuel Storage and Handling) and 32 (Spent Fuel Interim Storage) respectively.

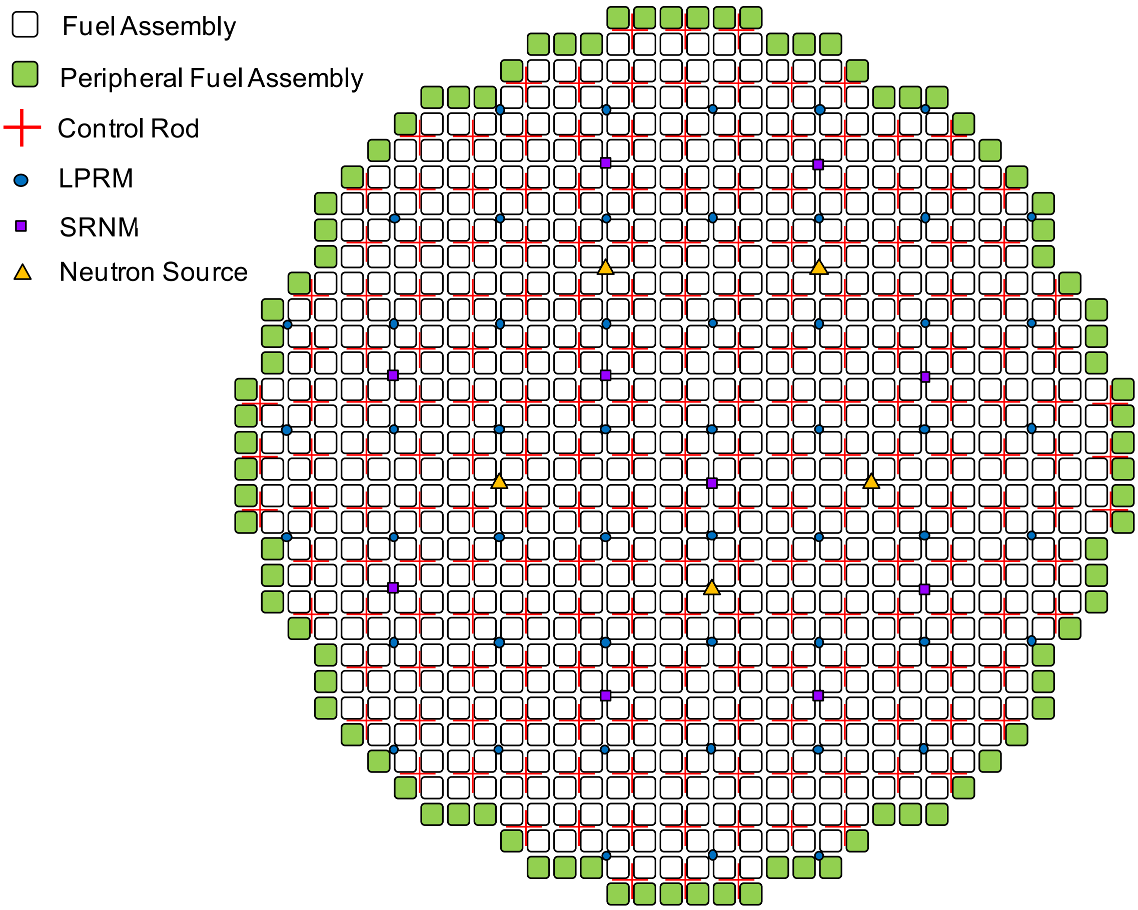


Figure 11.3-2 : ABWR Core Configuration

11.4 Design Basis

11.4.1 Claims and Link to High Level Safety Functions

Claims in the Reactor Core are listed in Appendix A with both a Safety Claim ID (e.g. NF SC 1) and a corresponding Safety Functional Claim (SFC) ID (e.g. FA SFC 4-10.1). The Safety Claim ID is implemented to enhance the traceability of linkage between Claims-Arguments-Evidence [Ref-11-1]. Also the Safety Functional Claim ID indicates the linkage to High Level Safety Functions (HLSFs) as shown in Appendix A. When a single claim maintains multiple HLSFs, multiple SFC IDs are given for this single claim. A short description on the application of High Level Safety Functions in the development of the claims, arguments and evidence is provided in Generic PCSR Chapter 1: Introduction.

11.4.2 Fuel System

The plant conditions for the fuel system design are divided into three categories.

- Normal operation,
- Frequent design basis faults, and
- Infrequent design basis faults.

The fuel system (comprised of the fuel bundle, fuel channel and CR) has the following functional requirement in normal operation and frequent design basis faults:

- Containment of radioactive materials, in particular fission products, within a barrier.

The fuel system has the following functional requirements under normal operation, frequent design basis faults and infrequent design basis faults:

- Reactivity control and safe core shutdown,
- Preservation of geometry for heat removal, and
- Limitation of radioactive material release resulting from potential fuel damage at less than prescribed criteria.

11.4.2.1.Fuel Rod

Mechanical Failure - Creep Deformation (Cladding Lift-Off) - for Containment

Cladding creep out rate due to fuel rod internal pressure is less than or equal to the fuel pellet irradiation swelling rate in normal operation and all frequent design basis faults so as to preclude creep deformation [FA SFC 4-10.1].

Background:
Fuel rod internal pressure increases due to the power rise, the corresponding gas temperature rise and the reduction in the fuel rod void volume caused by fuel pellet expansion. If the rate of the cladding outward deformation caused by the internal pressure exceeds the rate at which the fuel pellet expands due to irradiation swelling, the creep deformation could occur with a potentially adverse effect such as increase in pellet-cladding gap, decrease in pellet-cladding thermal conductance and increase in release of gaseous fission products.

Thermal Failure – Fuel Melting - for Containment

The fuel centerline temperature is less than the melting temperature in normal operation and all frequent design basis faults so as to preclude subsequent potential cladding damage [FA SFC 4-10.2].

Background:
Fuel melting is prevented so that molten fuel will not contact the cladding nor produce local hot spots that could result in potential cladding damage.
Evaluation to confirm that the above requirements are met is given in Generic PCSR Chapter 24 Section 24.3

Mechanical Failure – Cladding Strain - for Containment

The cladding circumferential strain due to pellet-clad mechanical interaction is less than the design limit at normal operation and all frequent design basis faults so as to preclude mechanical failure due to cladding strain [FA SFC 4-10.3].

Background:
After the initial rise in power and the establishment of steady-state operating conditions, the pellet-cladding gap will eventually close due to the combined effects of cladding creep-down, fuel pellet irradiation swelling and fuel pellet fragment outward relocation. Once hard pellet-cladding contact

has occurred, cladding outward diametral deformation could occur. The consequences of this cladding deformation are dependent on the deformation rate (i.e. strain rate).
Evaluation to confirm that the above requirements are met is given in Generic PCSR Chapter 24 Section 24.3

Mechanical Failure - Fatigue - for Containment

The cumulative duty from cladding strain due to cyclic loadings in normal operation and all frequent design basis faults is less than the cladding fatigue capability [FA SFC 4-10.4].

Background:
Cyclic loadings are applied to the fuel rod cladding as a result of power changes anticipated during normal reactor operation including frequent design basis faults, planned surveillance testing, normal control blade manoeuvers, shutdowns and special operating modes such as daily load following.

Mechanical Failure - Creep Collapse - for Containment

The fuel cladding damage from cladding collapse into a fuel column axial gap is precluded in normal operation and all frequent design basis faults [FA SFC 4-10.5].

Background:
The condition of an external coolant pressure greater than the fuel rod internal pressure provides the potential for elastic buckling or possibly even plastic deformation if the stresses exceed the material yield strength.

Mechanical Failure – Fuel Rod Stress - for Containment

The cladding stresses or strains are less than material failure limits at normal operation and all frequent design basis faults so as to preclude mechanical failure due to cladding stresses or strains [FA SFC 4-10.6].

Background:
Stresses or strains imposed by the effects such as the difference between external coolant pressure and fuel rod internal gas pressure, cladding ovality, radial thermal gradients, spacer contact, thermal

bow and circumferential thermal gradients, could occur in the cladding tube. The combined stresses or strains from these effects could potentially cause cladding damage.

Chemical Failure - Hydride Embrittlement Cracking - for Containment

The fuel cladding damage from the localised reduction in cladding ductility due to hydriding is precluded in normal operation and all frequent design basis faults [FA SFC 4-10.7].

Background:
There are two considerations relevant to fuel rod hydrogen absorption. The first consideration involves the potential for hydrogenous impurity evolution, historically from the fuel pellets, resulting in primary hydriding and fuel rod failure. The second consideration is the partial absorption by the fuel rod cladding of hydrogen liberated by the cladding waterside corrosion reaction. The hydrogen absorption produces localised reduction in cladding ductility that could reduce the cladding failure strain below specified cladding strain limits.

Chemical Failure - Significant Oxidation - for Containment

The fuel cladding damage from cladding oxidation is precluded in normal operation and all frequent design basis faults [FA SFC 4-10.8].

Background:
Cladding undergoes oxidation at slow rates during normal reactor operation. Cladding oxidation causes thinning of the cladding tube wall and introduces a resistance to the fuel rod-to-coolant heat transfer. Therefore, cladding oxidation affects the overall strength of the cladding through loss of structural material and reduced elevated temperature material strength.

Mechanical Failure - Fretting Wear - for Containment

The fuel cladding damage from fretting wear is precluded in normal operation [FA SFC 4-10.9].

Background:
Spacers provide lateral support and spacing of the fuel rods. Flow induced vibration response could cause significant vibration and consequent fretting wear, particularly at spacer–fuel rod contact points.

Thermal Failure - Cladding Overheating- for Containment

See Subsection 11.4.6.1.

Combined Failure - Pellet Cladding Interaction (PCI) and Stress Corrosion Cracking (SCC) - for Containment

The fuel cladding damage from PCI/SCC is precluded in normal operation and any frequent design basis faults [FA SFC 4-10.10].

Background:
A PCI failure could occur due to a combination of tensile stress from Pellet Cladding Mechanical Interaction (mechanical component) and corrosion from fission product accumulation, such as iodine (chemical component).

Thermal Failure - Embrittlement from Oxidation - for Preservation of Geometry

In all infrequent design basis faults,

(1) Peak cladding temperature is less than 1,200 °C [FA SFC 2-1.1].

(2) Peak cladding oxidation is less than 15 percent equivalent cladding reacted (ECR) [FA SFC 2-1.2].

Background:
The above requirements are established to preserve adequate postquench ductility and thus preserve the fuel geometry and coolability.
Evaluation to confirm that the above requirements are met is given in Generic PCSR Chapter 24 Section 24.3.

Sudden Rupture of Fuel Cladding - Reactivity Insertion Accident - for Preservation of Geometry

Fuel enthalpy meets the prescribed design limit for all frequent and infrequent design basis faults [FA SFC 2-1.3].

Background:
The enthalpy limit is established to maintain the fuel integrity or coolable geometry. For coolable geometry, the limit is established to preclude the impairment of CR insertability and coolability in the event of a reactivity insertion accident. This event could cause potential fuel melting, fragmentation and dispersal of fuel, resulting in damage to the reactor coolant pressure boundary and damage to the core, core support structure and pressure vessel internal structure that may impair cooling of the fuel.

Evaluation to confirm that the above requirements are met is given in Generic PCSR Chapter 24 Section 24.3

11.4.2.2.Fuel Assembly and Components - for Preservation of Geometry

(1) The FA structural components (fuel rods, upper tie plate, lower tie plate, spacers, water rods and fuel channel) do not fail due to stresses less than or equal to the FA component mechanical capability [FA SFC 1-2.1][FA SFC 2-1.4][FA SFC 5-6.1].
(2) The cumulative duty on FA structural components from cyclic loadings is less than the material fatigue capability [FA SFC 1-2.2] [FA SFC 2-1.5].
(3) The fuel cladding damage from fretting wear is precluded by the FA structural components [FA SFC 1-2.3] [FA SFC 2-1.6].

Background:
Requirements on FA structural components are established for preservation of geometry by considering:
(1) The thermal, pressure and mechanical interaction loadings predicted to occur during startup testing, normal operation, frequent design basis faults and infrequent design basis faults.
(2) The loading predicted to occur during transportation and handling.

11.4.2.3.Fuel Assembly and its Components - for Limitation of Radioactive Material Release

Fuel assembly and its components are designed so that the effective dose received by any person is less than the prescribed limit, and within the functions of relevant plant components [FA SFC 4-10.11]. Further details are discussed in Generic PCSR Chapter 24 Section 24.3 and [Ref-11-4].

11.4.2.4.Spent Fuel Storage - for Containment

(1) When fuel assemblies are stored in the spent fuel storage pool which is surrounded by water, fuel cladding damage from corrosion is precluded [FA SFC 4-10.12].
The fuel rack is designed to be subcritical up to the maximum storage capacity.
(2) When fuel is transferred to dry storage under an inert environment, the performance will depend on the temperature. Under normal operation and any frequent design basis faults the inert environment and maximum temperature are controlled and kept less than the prescribed limit so as to preclude cladding failure [FA SFC 4-10.13].
The fuel container is designed to be subcritical up to the maximum storage capacity.

Fuel storage in spent fuel storage pool is discussed in Generic PCSR Chapter19 Section 19.3 (Spent Fuel Storage). Fuel integrity at spent fuel storage pool is discussed in [Ref-11-22].
Spent Fuel Interim Storage is discussed in Generic PCSR Chapter 32: Spent Fuel Interim Storage. Fuel integrity at Spent Fuel Interim Storage is discussed in [Ref-11-22].

11.4.2.5.Control Rod (CR) - for Reactivity Control and Safe Core Shutdown

Requirements on functional design are shown in Subsection 11.4.5.
Requirements on structural design are as follows:
(1) The CR stresses are less than the allowable limit [CR SFC 1-2.1].
(2) The material of the CR shall be shown to be compatible with the reactor environment [CR SFC 1-2.2].

Background:
Requirements on structural design of the CR are established to meet the following conditions considering any mechanical, pressure, temperature and irradiation effects during normal operation and frequent/infrequent design basis faults:
(1) Sufficient mechanical strength to prevent displacement of reactivity control material.
(2) Sufficient strength to prevent deformation that could inhibit motion.

11.4.3 Control Rod Drive (CRD) System

Requirements on functional design are shown in Subsection 11.4.5. Requirements on design are shown in Generic PCSR Chapter 12 Subsection 12.4.3.1: Control Rod Drive System.

11.4.4 Standby Liquid Control System (SLC)

Requirements on functional design are shown in Subsection 11.4.5. Requirements on other design aspects are shown in Generic PCSR Chapter 12 Subsection 12.4.3.2: Standby Liquid Control System.

11.4.5 Nuclear Design

The functional requirement met by the nuclear design at normal operation, frequent design basis faults and infrequent design basis faults is:

- Reactivity control to enable the chain reaction to be stopped under all circumstances and to return the reactor to a safe shutdown state

The nuclear design must ensure this safety function so as not to exceed the allowable fuel design limits for frequent design basis faults, and not to exceed allowable fuel design limits (see Subsection 11.4.2.1) related to the fuel enthalpy for reactivity insertion transients (e.g., unplanned withdrawal of CR during reactor start-up).

11.4.5.1.Control of Power Distribution

The power distribution is controlled so that the operational thermal limits (Maximum Linear Heat Generation Rate and Minimum Critical Power Ratio) (see Subsection 11.5.5.1 for their definition) are not exceeded at operation and all frequent design basis faults [ND SFC 1-8.1].

11.4.5.2.Nuclear Power Restriction Characteristics of Reactor

The reactivity coefficients, such as the moderator void coefficient and the Doppler coefficient, are always negative and always satisfy the following conditions:

(1) The overall moderator void coefficient is negative within a range that pressurization transients do not threaten fuel integrity [ND SFC 1-8.2].
(2) The Doppler coefficient is negative and has sufficient reactivity feedback characteristics to terminate infrequent design basis faults (i.e., reactivity insertion accident) [ND SFC 1-8.3].
(3) The moderator temperature coefficient is not a value that may pose a safety concern in the current design range and it has no specific requirement [ND SFC 1-8.4].
(4) The power reactivity coefficient is negative and its feedback is large enough to reduce the spatial oscillation of xenon [ND SFC 1-8.5].

The validity of the reactivity coefficient is confirmed through core characteristics evaluation, rather than imposing restrictions on the value itself.

11.4.5.3.Independence of Reactor Shutdown System

The reactor shutdown system is able to bring the core to a subcritical state, with adequate margin that bounds inherent biases and uncertainties associated with calculational methods, from a hot stand-by or power operation condition [ND SFC 1-4.1].

The reactor shutdown system consists of two independent systems: CR with the FMCRD system that provides a hydraulic fast scram capability and a redundant electric run-in insertion method of CRs (elements that contribute to the excellent safety characteristics of the ABWR), and the Standby Liquid Control System (SLC) which can maintain a subcritical state at hot stand-by assuming a zero xenon condition (i.e., no shutdown credit is taken for xenon) [ND SFC 1-4.2] [ND SFC 1-5.1].

11.4.5.4.Shutdown Margin with Control Rod

The CR and FMCRD system of the reactor shutdown system (with hydraulic and electric run-in capability of CRs) is able to bring the core to a subcritical state at hot or cold conditions when one or more CRs with the largest reactivity worth (specifically, one rod or a pair of rods belonging to the same hydraulic control unit) are completely withdrawn from the core and cannot be inserted [ND SFC 1-4.3].

11.4.5.5.Hot Shutdown Capability of Reactor Shutdown System

For frequent design basis faults, the CR and FMCRD system is able to bring the core to a subcritical state at hot conditions and maintain this state at hot conditions, preventing the allowable fuel design limit from being exceeded [ND SFC 1-3.1] [ND SFC 1-4.4].

The requirements on FMCRD are discussed in Generic PCSR Chapter 12 Subsection 12.4.3.1: Control Rod Drive System.

11.4.5.6.Cold Shutdown Capability of Reactor Shutdown System

At least one of the independent reactor shutdown systems is able to bring the core to a subcritical state at cold conditions and maintain this state at cold conditions [ND SFC 1-3.2] [ND SFC 1-4.5].

11.4.5.7.Maximum Reactivity Worth and Reactivity Insertion Rate of Control Rod

The maximum reactivity worth and the reactivity insertion rate of the CR, which are less than the design limits used in the safety analyses of anticipated reactivity insertion events, do not damage the reactor coolant pressure boundary and do not cause damage to the core, core support structure and pressure vessel internal structure that may impair cooling of the fuel [ND SFC 1-1.1].

11.4.5.8.Power Oscillation Control Characteristics of Reactor

See Subsection 11.4.6.2.

11.4.6 Thermal Hydraulic Design

The functional requirements met by the thermal hydraulic design at normal operation, frequent design basis faults and infrequent design basis faults are:

- Removal of heat produced in the fuel via the coolant fluid, and
- Reactivity control by moderator density control to return the reactor to a safe shutdown state.

The thermal hydraulic design ensures this safety function so as to maintain fuel integrity and prevent the allowable fuel design limits at frequent design basis faults from being exceeded.

11.4.6.1.Thermal Failure - Cladding Overheating - for Removal of Heat

At normal operation and frequent design basis faults;

(1) Minimum Critical Power Ratio (MCPR) is equal to or has margin to the safety limit (see Subsection 11.6.6.3) so as to preclude thermal failure with overheating [THD SFC 2-1.1].

(2) When MCPR becomes less than the safety limit, cladding temperature is less than 800 °C such that ballooning rupture failure is precluded (ballooning rupture can occur above 800 °C) [THD SFC 2-1.2].

Background:

(1) CPR is the figure of merit utilised for plant operation to avoid boiling transition and thus preclude thermal failure with embrittlement from oxidation. The above safety limit is established so that no boiling transition arises in 99.9 percent or more of the fuel rods in the event of a frequent design basis fault. A core-specific operating limit CPR (OLMCPR) is established to provide adequate assurance that the design limit is not violated. The plant is operated with the MCPR at or above the OLMCPR. The SLMCPR is a lower operating bound on the steady-state MCPR. This operating requirement is obtained by removing the delta CPR effect of frequent design basis faults from the OLMCPR. Evaluation to confirm that the above requirements are met is given in Generic PCSR Chapter 24 Section 24.3.

(2) In addition, phase transformation (α phase$\rightarrow\alpha+\beta$ phase) temperature of Zircaloy is approximately 800 °C. For cases where the Peak Cladding Temperature (PCT) is less than 800 °C, the microstructure of the cladding keeps the α phase after rapid cooling due to rewetting. Therefore, cladding ductility does not significantly decrease. PCT less than 800 °C is set in terms of maintaining fuel integrity during fuel handling after experiencing boiling transition. Oxidation during short periods of Boiling Transition (BT) at the temperatures less than 800 °C is not considered to be significant. Therefore, embrittlement failure due to oxidation does not occur below 800 °C. Furthermore, PCT less than the rupture-temperature curve is set so as to preclude thermal mechanical failure due to ballooning rupture. Evaluation to confirm that the above requirements are met is given in Generic PCSR Chapter 24 Section 24.3.

11.4.6.2.Power Oscillation Control Characteristics of Reactor – For Reactivity Control

The ABWR is stable in the normal operating region in normal operation. The calculated decay ratio (DR) for thermal hydraulic stability does not exceed the limiting criteria in the normal operating region shown in Figure 11.5-11 [THD SFC 1-8.1]. In addition, in case that power oscillation due to thermal hydraulic instability occurs, it is controlled and the reactor is led to a stable state [THD SFC 1-8.2].

11.4.7 Core Monitoring System

Core performance parameters such as power, flow, MLHGR, MCPR and exposure are monitored so that the fuel integrity is maintained during the reactor operation. These parameters are calculated using the measurement values such as neutron flux, pressure and flow [CMS SFC 5-12.1] (See Subsection 4.4.2 of [Ref-11-3]). Instrumentation that performs these measurements is described in [Ref-11-2].

11.5 Design Description

11.5.1 Fuel System

The fuel system consists of the Fuel Assembly (FA) and CR. The FA consists of fuel bundle and fuel channel. The fuel bundle consists of fuel rods, water rods, expansion springs, spacers, debris filter and upper and lower tie plates.

The main materials used for the FA are zircaloy, zirconium, stainless steel, Inconel, ceramic UO_2 and Gd_2O_3 (Gadolinia). Material characteristics such as mechanical properties, corrosion resistance, etc. shall conform to the conditions of the boiling water reactor and satisfy the design intent of the reactor during operation.

The reference fuel design in the UK ABWR GDA, designated as GE14, has been deployed in reload quantities for over 15 years and has been approved for use in the following countries: Germany, Switzerland, Sweden, Finland, Spain, Mexico, Taiwan and the United States.

GE14 is the single largest design (in terms of the number of reloads, bundles, and fuel rods) operated in the history of GNF BWR fuel and is also the largest of any supplier. The experience base as of February 2014 is >175 reloads, >33,500 bundles, and >3 million fuel rods. GE14 has been operated in the above countries in 43 different reactors, in every existing BWR water chemistry environment and in every existing BWR operating regime (annual, 18, and 24-month cycles, low to high power density including Extended Power Uprate (EPU), and various control strategies.)

To demonstrate ABWR system response, a reference core of GE14 fuel is used. This core is described in Subsection 11.5.4.

It is noted that fuel and core designs different than that described herein, which meet Design Basis in Section 11.4, may be introduced post GDA.

11.5.1.1.Fuel Assembly (FA)

The evolution steps of BWR fuel designs are illustrated in Figure 11.5-1. The increase in fuel rod array provides lower linear heat generation and rod power, thereby resulting in higher fission product retention capability. All BWR fuels - 7×7, 8×8, 9×9 and 10×10 - can be used in the same BWR core.

The 10×10 FA is shown in Figure 11.5-2, and the fuel rods, fuel module (cell) and channel fastener assembly are shown in Figure 11.5-3, Figure 11.5-4 and Figure 11.5-5. A FA consists of a fuel bundle and a fuel channel that surrounds the fuel bundle. The GE14 design utilises a 10×10 fuel rod array that includes 78 full length fuel rods, 14 part length fuel rods and 2 large central water rods. The cast stainless steel lower tie plate includes a conical section that seats into the fuel support and a grid that establishes the proper fuel rod spacing at the bottom of the bundle. The lower tie plate also houses a debris filter to prevent debris from entering the assembly and potentially leading to fretting failure of the fuel rod cladding. The cast stainless steel upper tie plate maintains the fuel rod spacing at the top of the bundle and provides the handle that is used to lift the bundle for transferring the said bundle from one location to another. To correctly identify the FA, a unique serial number is engraved on the top of the handle. A boss projects from one side of the handle to ensure proper orientation of the assembly in the core. A visual inspection of both the serial number engraved on the handle and boss projection aid in the verification of proper bundle orientation and location. Finger springs located between the lower tie plate and fuel channel are utilised to maintain a nearly constant leakage flow into the bypass region (i.e., the region outside of the fuel channel). Additional bypass flow enters the bypass region through 2 lower tie plate holes that help provide sufficient cooling for plant instrumentation located outside the fuel channels. The entire fuel bundle is held together by 8 threaded tie rods located around the periphery of the bundle. Another key component of the bundle are the 8 spacer grids which have the function of maintaining proper spacing between the fuel rods along the axial length of the bundle as well as influencing critical power performance.

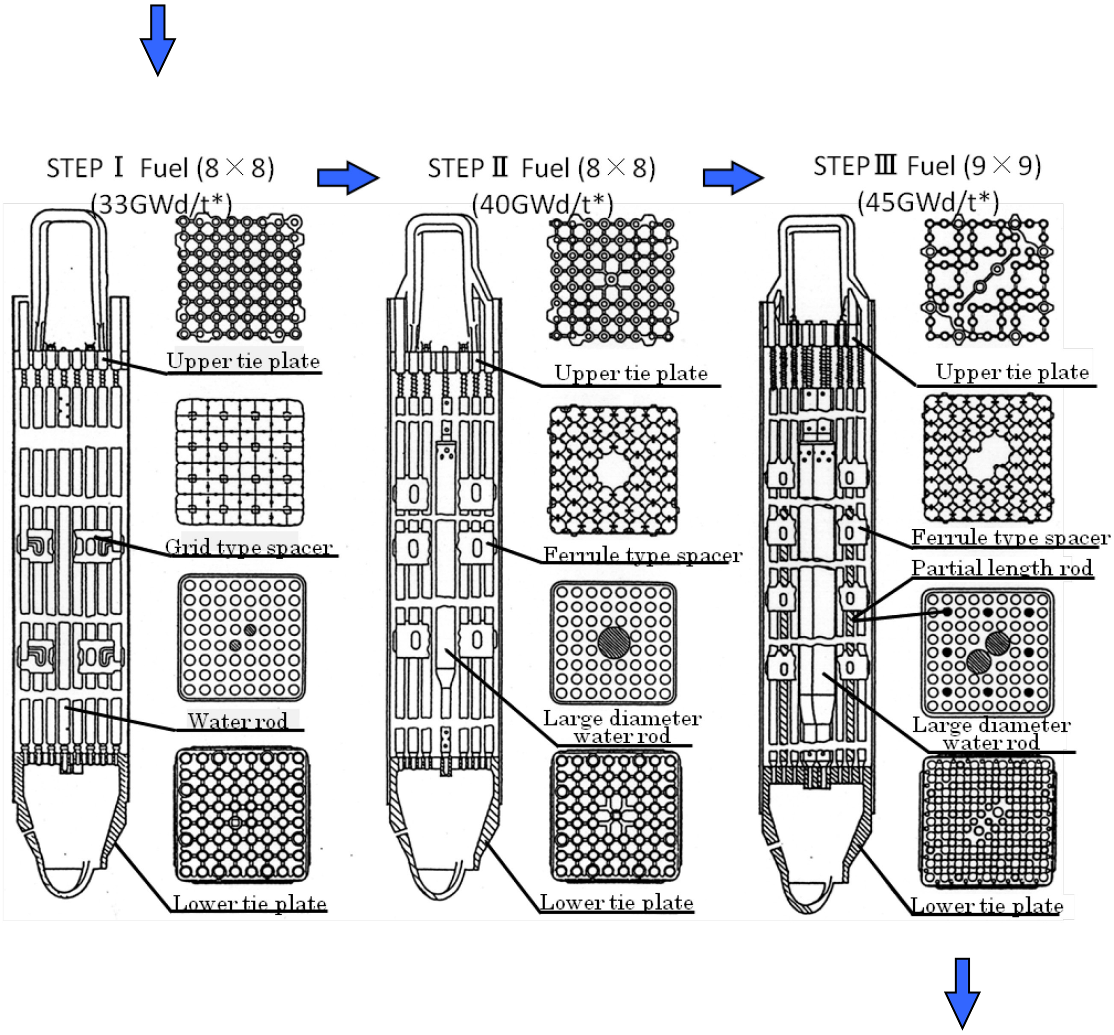
FA parameters are provided in Table 11.5-1.

Further details are given in Chapter 1 of [Ref-11-5].

Table 11.5-1: Fuel Assembly Parameters (example)

Fuel Assembly	
Fuel rod array	10×10
Overall length (mm)	4,468
Number of fuel rods	92
Number of full-length rods (FLR)	78
Number of part-length rods (PLR)	14
Number of water rods	2
Number of spacers	8
Weight of UO_2 per assembly (kg)	204
Weight of fuel assembly (kg) (includes fuel channel)	301
Fuel Rods	
Outside diameter (mm)	10.26
Active fuel length (mm) FLR/PLR	3,810/2,134
Cladding material	Zircaloy-2 / Zirconium
Fuel Pellets	
Material	UO_2 or UO_2+Gd_2O_3
Water rod	
Material	Zircaloy-2
Fuel Channel	
Material	Zirconium alloy
Core Assembly	
Number of fuel assemblies	872
Fuel weight as UO_2 (t)	180
Core diameter (equivalent) (m)	5.16
Core height (active fuel) (m)	3.81

7×7 type



10×10 type

*: Average discharge exposure

Figure 11.5-1 : BWR Fuel Evolution

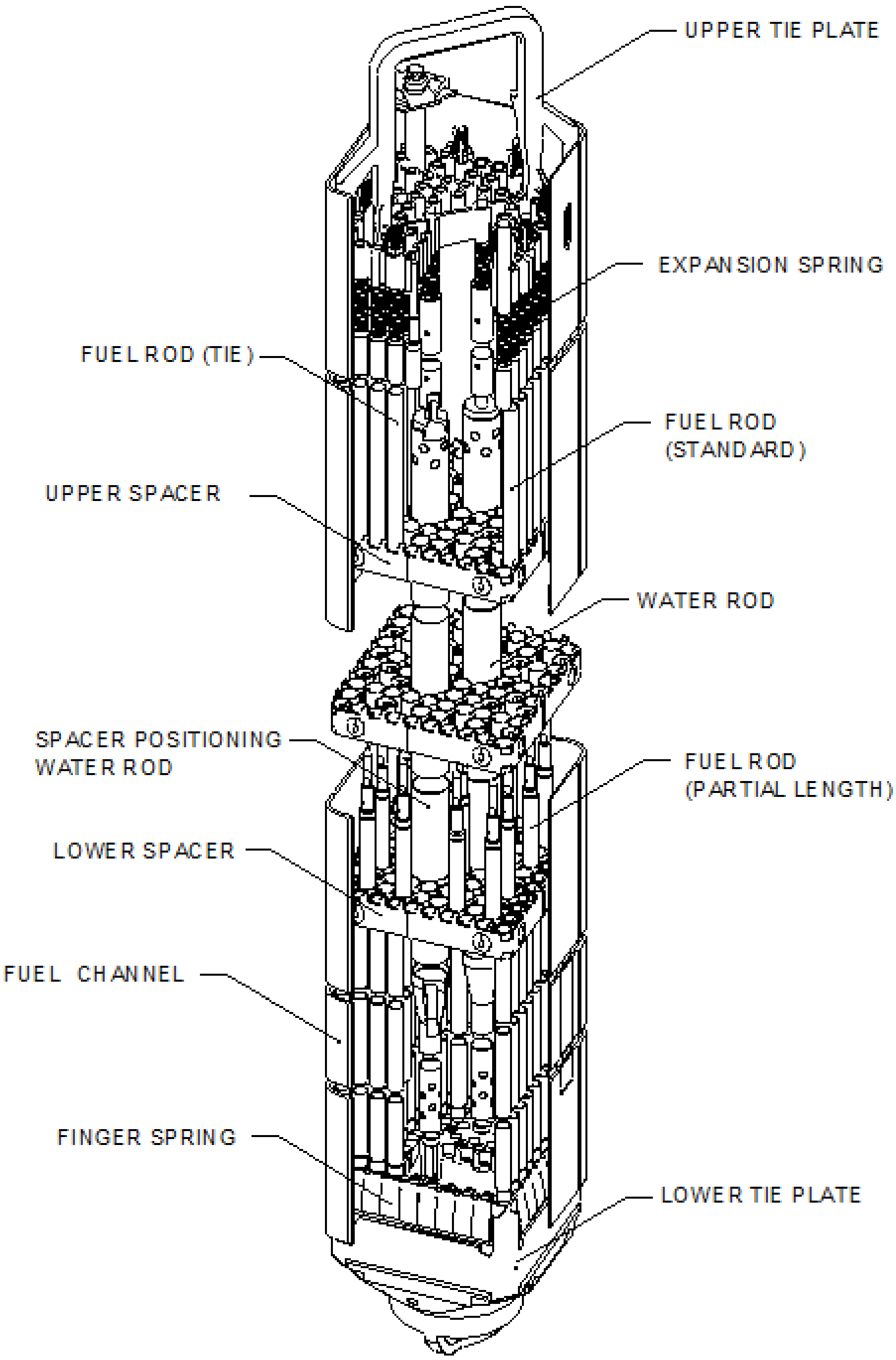


Figure 11.5-2 : Fuel Assembly

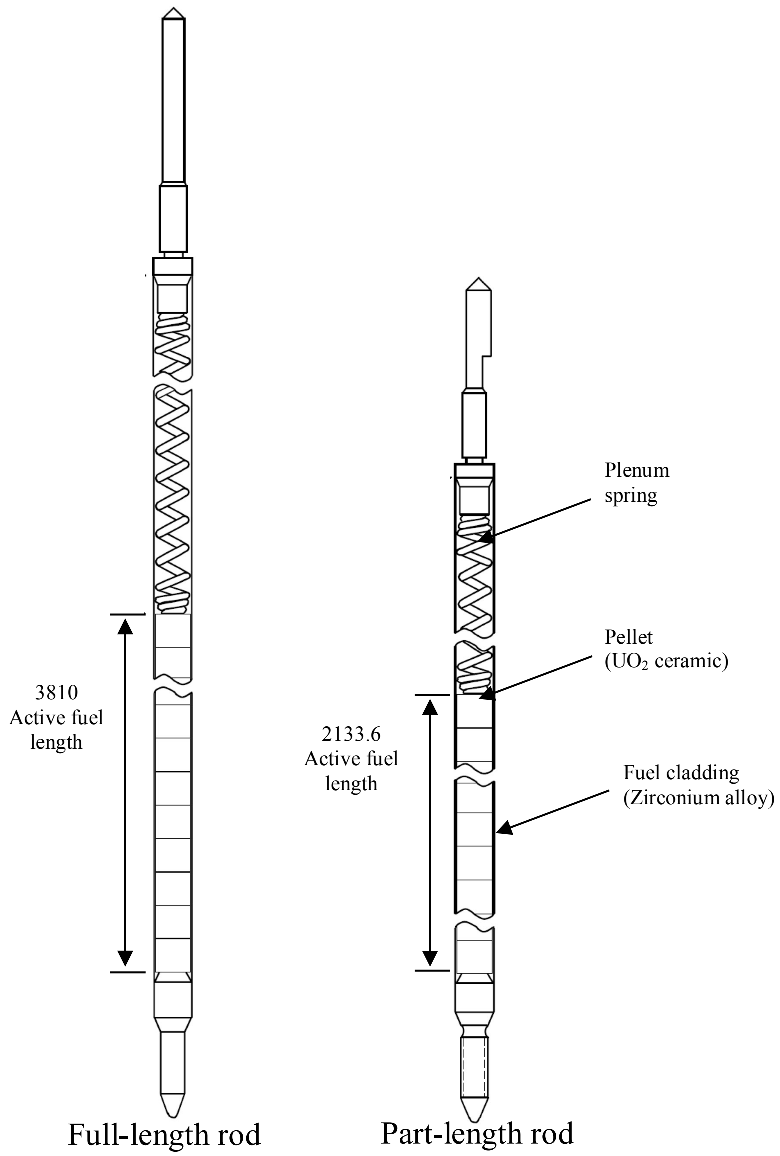


Figure 11.5-3 : Fuel Rods

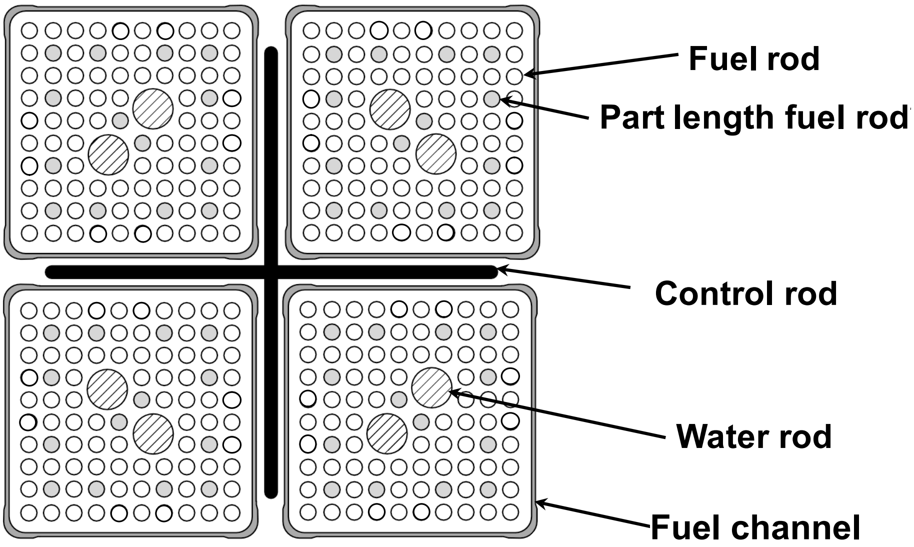


Figure 11.5-4 : Four Bundle Fuel Module (Cell)

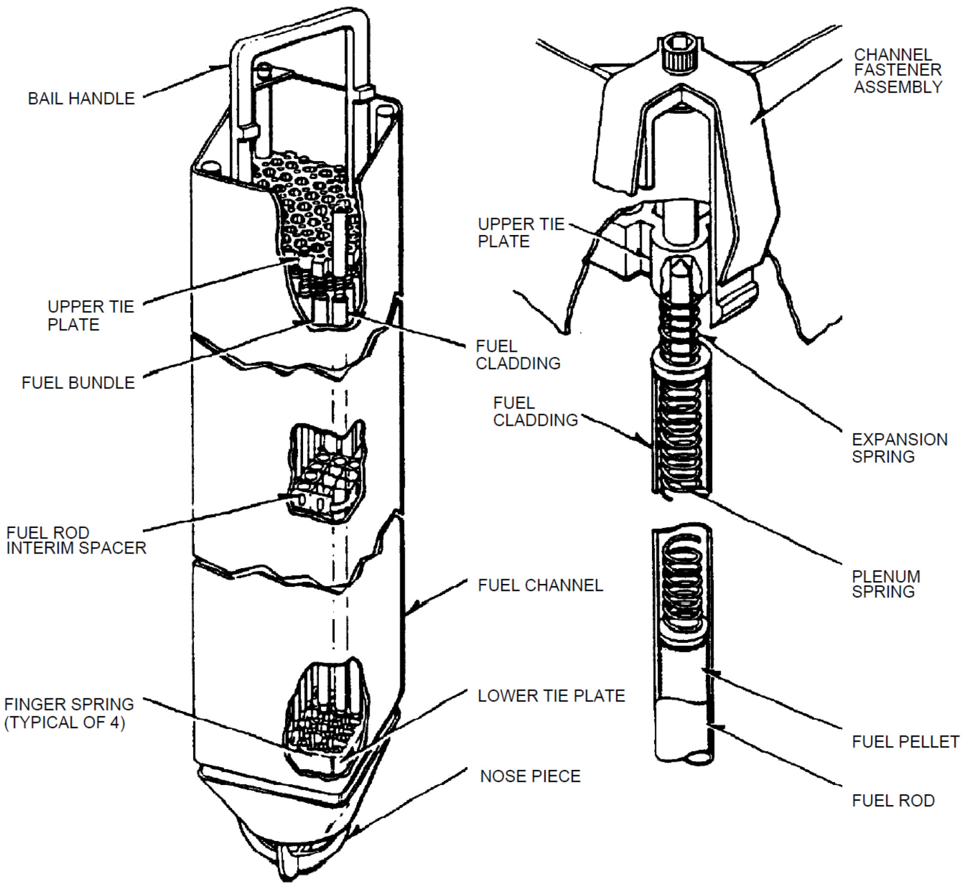


Figure 11.5-5 : Channel Fastener Assembly

Fuel Rod

The basic GE14 fuel rod is comprised of a column of cylindrical fuel pellets enclosed by a cladding tube and sealed gas-tight by plugs inserted in each end of the cladding tube. The plugs are welded after insertion. The fuel pellets consist of uranium-dioxide (UO_2) or UO_2-gadolinia with a ground cylindrical surface, flat ends and chamfered edges. The fuel rod cladding tube is comprised of Zircaloy-2 with a metallurgically bonded inner barrier zirconium layer.

As shown in Figure 11.5-3, two types of fuel rods are used in a fuel bundle; full length and part length rods. Eight of the full length rods (tie rods) have a threaded lower end plug that screw into the lower tie plate and a threaded upper end plug that extends through a boss in the upper tie plate and is fastened with a nut. A locking tab washer is included under the tie rod nut to prevent rotation of the tie rod and nut. The part length rods also have lower end plugs that are threaded into the lower tie plate to prevent movement of the rods during shipping or handling with the bundle oriented horizontally. The tie rods support the weight of the assembly only during fuel handling operations. During operation, the assembly is supported by the lower tie plate.

The upper end plugs of the full length fuel rods and water rods have extended shanks that protrude through bosses in the upper tie plate to accommodate the differential growth expected for high exposure operation. Expansion springs are also placed over each upper end plug shank to assure that the full length fuel rods and water rods remain properly seated in the lower tie plate.

Each fuel rod contains high density ceramic uranium dioxide fuel pellets stacked within Zircaloy cladding. The fuel rod is evacuated, backfilled with helium, and sealed with end plugs welded into each end. U-235 enrichments may vary from fuel rod to fuel rod within a bundle to reduce local peak-to-average fuel rod power ratios.

Selected fuel rods within each bundle include small amounts of gadolinium as a burnable poison along the length of the fuel rod to provide axial power shaping. Gadolinium concentration may also be varied to achieve a desired hot excess reactivity depletion profile.

Adequate free volume is provided within each fuel rod in the form of a pellet-to-cladding gap and a plenum region at the top of each fuel rod to accommodate thermal and irradiation expansion of the column of UO_2 fuel pellets and the internal pressure resulting from the helium fill gas, impurities, and gaseous fission products liberated over the life of the fuel. A plenum spring, or retainer, is provided in the plenum space to minimise the movement of the column of fuel pellets inside the fuel rod during shipping and handling.

Water Rod

Water rods are hollow Zircaloy tubes with several holes around the circumference near each end to allow coolant to flow through. The GEl4 fuel design includes two large central water rods identical in size that occupy eight fuel rod locations and provide improved moderation. One of the water rods has welded tabs that are used to align the fuel spacers and firmly lock them into position.

Fuel Spacer

The primary function of the spacer is to provide lateral support and spacing of the fuel rods, with consideration of thermal hydraulic performance, fretting wear, strength, neutron economy, and manufacturability. The GE14 design includes a high performance spacer developed to meet the low pressure drop requirement for a 10×10 design and to provide excellent critical power performance. Eight spacers are employed in the GE14 design. With appropriate design of the spacer spring strength, all fuel rods and water rods are designed to have free expansion independently in the axial direction.

Finger Spring

Finger springs are employed to control the bypass flow through the fuel channel-to-lower tie plate flow path. They are designed to help maintain essentially the same bypass flow throughout the life of the bundle.

Debris Filter

The lower tie plate of the GE14 bundle houses a debris filter. This filter prevents debris from entering the fuel channel, thereby, providing resistance to debris fretting, and substantially improving fuel reliability.

Expansion Spring

Expansion springs provide sufficient axial spacing for the differential growth of the full length fuel rods and water rods.

Fuel Channel

The fuel channel is composed of Zirconium based material and performs the following functions:

(1) Form the fuel bundle flow path outer perimeter for bundle coolant flow.
(2) Provide surfaces for CR guidance in the reactor core.
(3) Provide structural stiffness to the fuel bundle during lateral loadings applied from fuel rods through the fuel spacers.
(4) Minimise, in conjunction with the finger springs and bundle lower tie plate, coolant bypass flow at the fuel channel/lower tie plate interface.
(5) Transmit FA seismic loadings to the top guide and fuel support of the core internal structures.
(6) Provide a heat sink during loss-of-coolant accident (LOCA).
(7) Provide a stagnation envelope for in-core fuel sipping.

The fuel channel is open at the bottom and makes a sliding seal fit on the lower tie plate surface. The upper end of the fuel assemblies in a four-bundle cell are positioned in the corners of the cell against the top guide beams by the channel fastener springs. At the top of the fuel channel, two diagonally opposite corners have welded tabs that support the weight of the fuel channel on the threaded raised posts of the upper tie plate. One of these raised posts has a threaded hole. The fuel channel is attached to the fuel bundle using the threaded channel fastener assembly, which also includes the FA positioning spring. Channel-to-channel spacing is assured by the fuel bundle spacer buttons located on the upper portion of the fuel channel adjacent to the CR passage area.

Fuel channels for the GE14 design have thinner sides and thicker corners to provide sufficient strength in the regions of highest stress while minimising material for improved neutron economy.

11.5.1.2.Control Rod (CR)

The CRs (Figure 11.5-6 and Figure 11.5-7) perform the functions of power shaping, reactivity control and scram reactivity insertion for safe shutdown response. Power distribution in the core is controlled during operation of the reactor by manipulating selected patterns of CRs to counterbalance steam void effects at the top of the core.

The CR consists of a sheathed cruciform array of either stainless steel tubes filled with boron carbide powder or hafnium metal. The main structure of a CR consists of the following stainless steel components: a top handle, a bottom connector and bayonet coupling socket, a vertical centre post (tie rod), and four U-shaped sheaths that surround the absorber rods. The top handle, bottom connector and tie rod are welded into a single skeletal structure. The U-shaped sheaths are welded to the tie rod, handle and connector to form a rigid housing to contain either the absorber rods or hafnium metal.

The U-shaped sheath has cooling holes to remove heat generated inside the blade by a coolant flow. The cooling holes consequently prevent failures caused by local thermal stress or burnout. Rollers at the top handle and the bottom connector of the CR guide the CR as it is inserted and withdrawn from the core.

CR parameters are provided in Table 11.5-2.

Further details are given in [Ref-11-20].

Table 11.5-2 : Control Rod Parameters

Control Rod	
Cross sectional shape	Cruciform
Overall length (mm)*	4,000
Width (mm)*	250
Blade thickness (mm)*	8
Material of main structural component	Stainless steel / casting
Number of control rods	205
Boron Carbide Type	
Neutron absorbing material	Boron carbide powder
Number of absorber rods	72
Hafnium Type	
Neutron absorbing material	Hafnium metal
Number of hafnium flat-tube	16

*Values are approximate

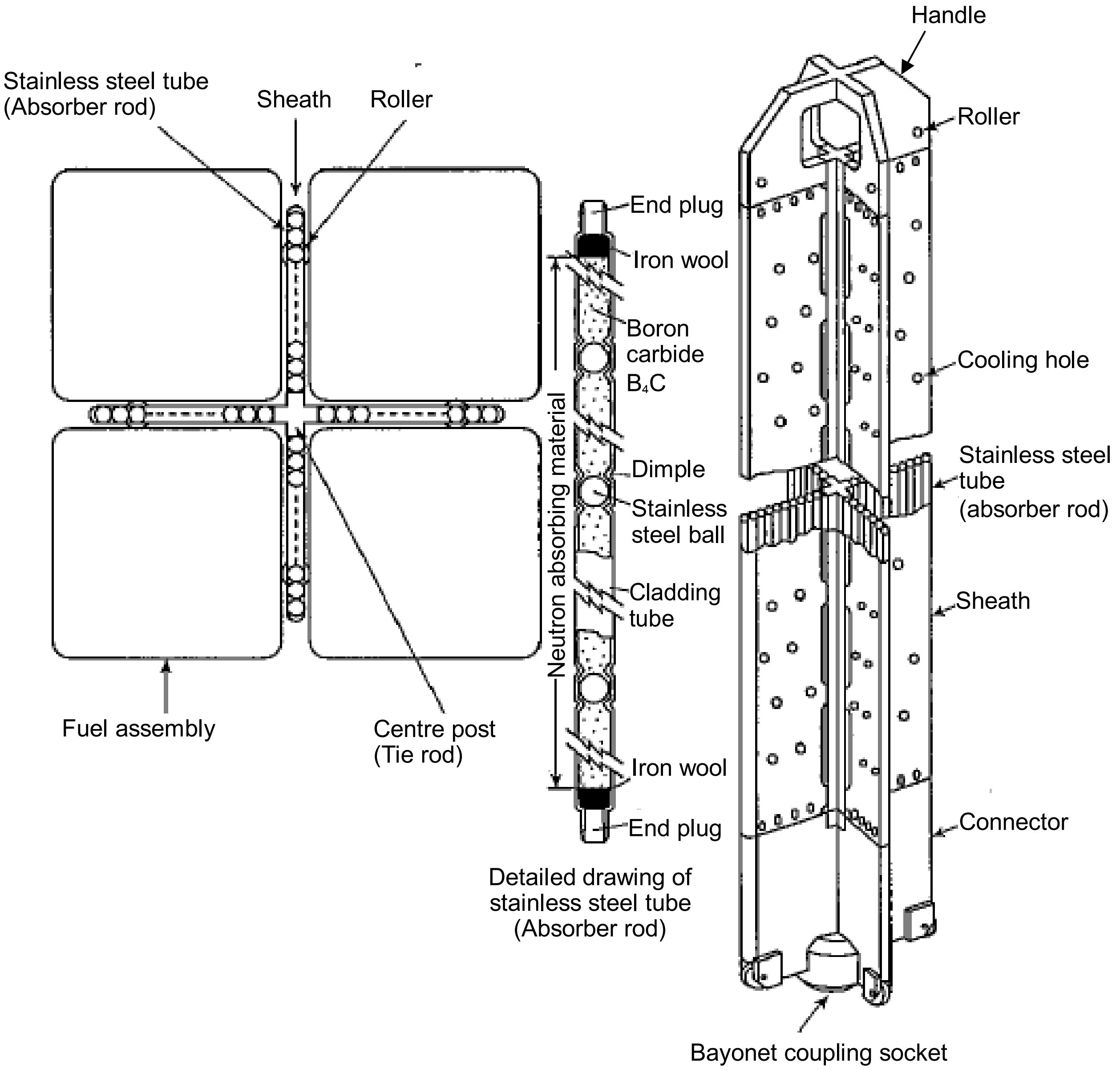


Figure 11.5-6 : Control Rod (Boron Carbide Type)

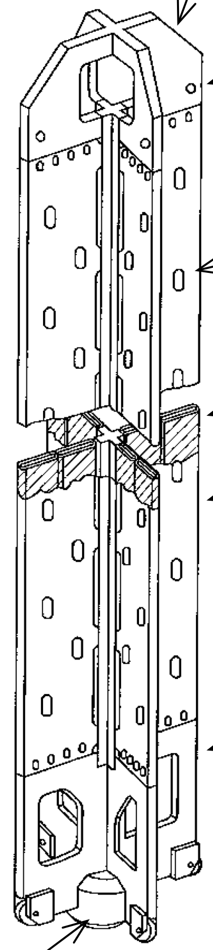


Figure 11.5-7 : Control Rod (Hafnium Type)

11.5.2 Control Rod Drive (CRD) System

The main roles of the CRD are the following:

(1) The CRD drives the electrohydraulic FMCRDs through an electric motor and thereby changes the position of the CRs in the core to control the reactivity during normal operation.

(2) During transients of the plant, the FMCRDs can be hydraulically driven by pressurised water from the HCU to rapidly insert all the CRs into the core, in an action known as a SCRAM, and thereby shut down the reactor safely and quickly.

(3) During normal operation the CRD Pumps supply purge water to the FMCRDs, the Reactor Internal Pumps (RIPs), and the Reactor Water Cleanup System pump (CUW Pump) while continuously maintaining the HCU Accumulators pressurised with water to ensure they are charged at high pressure for possible SCRAMs.

The CRD consists of the following main components:

- FMCRD,
- HCU,
- CRD Pumps,
- CRD Drive Water Heater,
- CRD Pump Suction Filters,
- CRD Charging Header Accumulator,
- HCU Nitrogen Gas Charging Equipment, and
- Valves, piping, instrumentation, and controllers.

Further details are given in Generic PCSR Chapter 12 Subsection 12.4.3.1: Control Rod Drive System.

11.5.3 Standby Liquid Control System (SLC)

The SLC is the secondary means to provide reactor shutdown and maintain sub-criticality if the reactor cannot be shut down by SCRAM action in an event known as Anticipated Transient Without Scram (ATWS).

The main role of the SLC is to inject a neutron absorbing solution to provide sufficient negative reactivity into the core to shut down the reactor in a safe manner from full power operation to cold shutdown conditions in the unlikely event that CRs insertion is not available. Sodium pentaborate solution is used as a neutron absorber.
The SLC injects the neutron absorber into the core from the SLC Storage Tank through the High Pressure Core Flooder System (HPCF) flooder sparger.
The SLC consists of the following components:

- SLC Storage Tank,
- SLC Test Tank,
- SLC Pump,
- Motor-operated injection valve,
- Piping and Valves, and
- Instruments and Control Components.

Further details are given in Generic PCSR Chapter 12 Subsection 12.2.3.2: Standby Liquid Control System.

11.5.4 Nuclear Design

The core employs light-water moderation and is fuelled with low-enriched uranium-dioxide. The use of water as a moderator produces a neutron energy spectrum in which fissions are caused principally by thermal neutrons. Under normal operating conditions, the moderator boils producing a spatially variable distribution of steam voids in the core. The ABWR design provides a system in which the fission rate is reduced by an increase in the steam void content in the moderator. This void feedback effect is one of the inherent safety features of the ABWR system. Any system input that increases reactor power, either in a local or gross sense, produces additional steam voids that reduce reactivity and thereby reduces the power. A reference core loading of 872 fuel bundles is used as the basis for the system dynamic response evaluation.

The equilibrium core of GE14 fuel is shown in Figure 11.5-8. In this core, fresh fuel assemblies are distributed throughout the centre region and thrice burned fuel assemblies are loaded in the peripheral region in order to minimise neutron leakage. The control cell core design concept with once or twice burned fuels within the control cells is adopted in order to mitigate power peaking associated with the withdrawal of CRs and/or alternating the CR pattern using 29 control cell locations.

Further details are given in Section 3.3 of [Ref-11-6] and [Ref-11-7].

	01	03	05	07	09	11	13	15	17	19	21	23	25	27	29	31	33	35	37	39	41	43	45	47	49	51	53	55	57	59	61	63	65	67
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1	Cycle N	224
2	Cycle N-1	224
3	Cycle N-2	224
4	Cycle N-3	196
5	Cycle N-4	4

Control Cell

Figure 11.5-8 : Reference Equilibrium Core Loading Map

11.5.4.1.Fuel Enrichment and Refuelling

The fuel enrichment is determined to sufficiently compensate for the reduction of reactivity due to neutron leakage, heating and boiling of the moderator, increase of the fuel temperature, neutron absorption by xenon and samarium, and burn-up of the fuel. The peak pellet enrichment is 4.9 wt percent; the majority of pellets within the fuel bundles are this enrichment in order to reach the target average enrichment that will achieve the design batch average discharge exposure of 50GWd/t. This batch average discharge exposure of 50GWd/t was determined as a target core design exposure considering the limit of maximum local exposure (peak pellet exposure) established by fuel rod thermal and mechanical design. Gadolinia (Gd_2O_3) is used as burnable poison to control the excess of reactivity of fresh FAs. Gadolinia concentration is optimised to burn out at the end of the operating cycle. The enrichment and gadolinia concentrations are varied axially in order to achieve the desired axial power and local pin power peaking distributions.

The final details of refuelling are determined following reactor shutdown according to the actual operating history. The basic concept is as follows.

(1) Regular scheduled refuelling

The operation cycle length of the reference equilibrium core and fuel design is 18 months. In principle, regular refuelling is performed once every 13 to 24 months. At the end of each cycle, the number of fuel assemblies and enrichment necessary to achieve the desired cycle length are determined, considering operating conditions and capacity factor. The position of the discharged fuel, the loading position of new fuel and the fuel arrangement are determined so that the reconfigured core satisfies cold shutdown margin, thermal limits and target burn-up under normal operation conditions. (Note that for most BWRs, batch size and enrichments are determined many months prior to shutdown based on accurate projections of the end-of-cycle core condition to support manufacturing schedules, customer outage schedules and licensing activities that are performed for each reload).

Although there are some variations in the number of discharged fuel assemblies, it is about 25 percent of the entire core on average for an 18 month equilibrium core. The corresponding batch average burn-up of the discharged fuel is about 50GWd/t.

To prevent the occurrence of a mislocated fuel loading error (MFLE), rigorous verification procedures are applied during and after core reconfiguration. During fuel movement, each move (location and orientation) is observed and checked at the time of completion by the operator and a spotter. After completion of the core load, the core is verified by videotaping the core using an underwater camera. In addition, during fuel movement, the neutron flux in the core is monitored by SRNM (Start-up Range Neutron Monitor). The requirement of lowest counting-rate is set to confirm that SRNM is working. When the count of SRNM is less than this set point, an alarm and Rod Block are given. A procedure is determined to exceed this set point during the fuel loading (e.g., fuel assemblies are placed around SRNM). The count rate of SRNM tends to increase when the core approaches to criticality by MFLE during fuel movement. Therefore any inadvertent criticality during fuel loading can be detected by SRNM. When the reactor period is shorter than a set point due to the significantly high rising rate of SRNM count, an alarm and Rod Block are given. Also when the count exceeds a set point, an alarm and Rod Block are given.

(2) Unscheduled refuelling

An unplanned refuelling outage may occur during an operating cycle due to various causes, for example, high offgas resulting from damaged fuel, although BWRs are capable of suppressing power in the vicinity of the damaged fuel and continue operation without further fuel degradation (Chapter 4 of [Ref-11-5]). In the event of plant shutdown, refuelling is performed such that shutdown margin and thermal limits are always satisfied. The refuelling interval and target burn-up would be adjusted accordingly.

11.5.4.2.Control Rod Withdrawal Procedure and Control Rod Pattern

The CR withdrawal procedure and the CR pattern at reactor start-up are developed to minimise CR worth and obtain adequate power distributions. In principle, all CRs are divided into two sequences that form a checkerboard pattern. Then, the CRs in one sequence are sub-divided into at least four basic groups. Withdrawal is performed for each group. When grouping the CRs, no adjacent CRs belong to the same group.

When all groups in one sequence are withdrawn, the CR fraction in the core becomes about 50 percent. Usually the hot standby condition of the reactor is attained near this CR pattern. The maximum CR worth is also attained in the process of approaching this condition.

The above procedure for CR withdrawal is not unique. The CR worth minimiser may block additional rod withdrawals in the event of a high worth rod at power conditions below a low power set point. Once this low power set point is reached, the preventive function of the rod worth minimiser is bypassed because the CR worth tends to become extremely small.

The CR withdrawal procedure and the CR pattern are determined by calculation and actual measurement. In other words, the CR withdrawal procedure and the CR pattern are determined in advance by the result of calculation, but are adjusted so that the expected power distribution may be obtained from the response values of the reactor neutron monitoring system during reactor operation.

The power distribution calculation value based on the predetermined CR pattern is used as an operating guideline by plant operators. The actual CR pattern, however, is established by operators based on actual core conditions.

The reactor power is controlled if necessary to meet the operational limits such as MLHGR/MCPR during CR pattern change.

11.5.5 Thermal Hydraulic Design

11.5.5.1.Reactor Core

Thermal Hydraulic Characteristic Data

Thermal hydraulic parameters for the ABWR at rated conditions are shown below.

Thermal power	3,926 MW
Steam flow rate*	7.64×10^3 t/h
Core flow rate*	52.2×10^3 t/h
Effective heat transfer area*	9,138 m^2
Reactor pressure*	7.17 MPa [abs]
Average heat flux*	430 kW/m^2
Maximum heat flux*	1,365 kW/m^2
Average power density*	49.2 kW/l

Fuel pellet highest temperature*	1,760 °C
Core inlet sub-cooling*	54.0 kJ/kg
Coolant core outlet temperature*	287 °C
Average steam quality at core outlet*	14.6 wt%
Core average void fraction*	43 %

*Values are approximate.

Critical Power Ratio (CPR)

CPR is the ratio of the FA power at which boiling transition occurs (critical power) to actual FA power. Critical power is evaluated by the GEXL boiling transition correlation described in Chapter 4 of [Ref-11-18]. A description of the CPR calculation is provided in Subsection 11.6.6.3.

Linear Heat Generation Rate (LHGR)

The LHGR limit is bundle type dependent. It is monitored to assure that all mechanical design requirements are met. The fuel will not be operated at LHGR values greater than those found to be acceptable within the body of safety analyses under normal operating conditions. Under abnormal conditions, including the maximum overpower condition, the maximum LHGR will not cause fuel melting or cause the stress and strain limits to be exceeded.

Core Coolant Flow Distribution

The flow distribution to the fuel assemblies and bypass flow paths are calculated on the assumption that the pressure drop across all fuel assemblies and bypass flow paths is the same. This assumption has been confirmed by measuring the flow distribution in boiling water reactors ([Ref-11-8], [Ref-11-9] and [Ref-11-10]). The components of bundle pressure drop considered are friction, local, elevation, and acceleration (Subsection 11.6.6.1). Pressure drop measurements made in operating reactors confirm that the total measured core pressure drop and calculated core pressure drop are in good agreement. Therefore, there is reasonable assurance that the calculated flow distribution throughout the core is in good agreement with the actual flow distribution of an operating reactor.

An iteration is performed on flow through each flow path (fuel assemblies and bypass flow paths), which equates the total differential pressure (plenum to plenum) across each path and matches the sum of the flows through each flow path to the total core flow.

Within the FA, heat balances on the active coolant are performed nodally. Fluid properties are expressed as the bundle average at the particular node of interest. A constant pressure model is used to evaluate fluid properties.

The relative radial and axial power distributions are used with the bundle flow to determine the axial coolant property distribution, which gives sufficient information to calculate the pressure drop components within each FA type. When the equal pressure drop criterion described above is satisfied, the flow distributions are established.

Void Fraction Distribution

The coolant flows into the inlet of the FA as single-phase water. It is heated as it passes through the FA, and becomes a two-phase flow mixture of steam and water. A typical axial void fraction distribution of the equilibrium core is shown in Figure 11.5-9 and Figure 11.5-10.

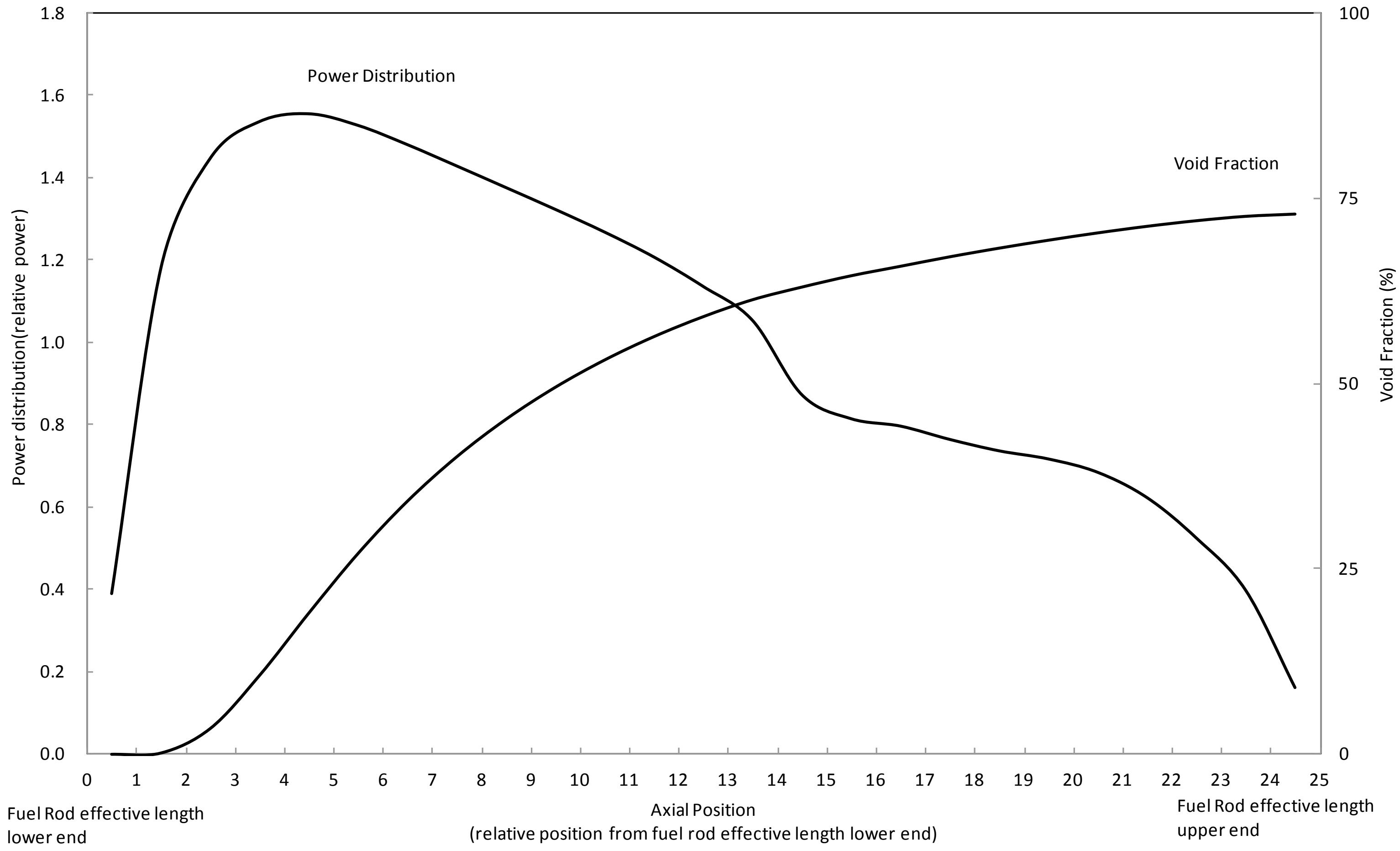

Figure 11.5-9 : Axial Power and Void Fraction Distribution (BOC)

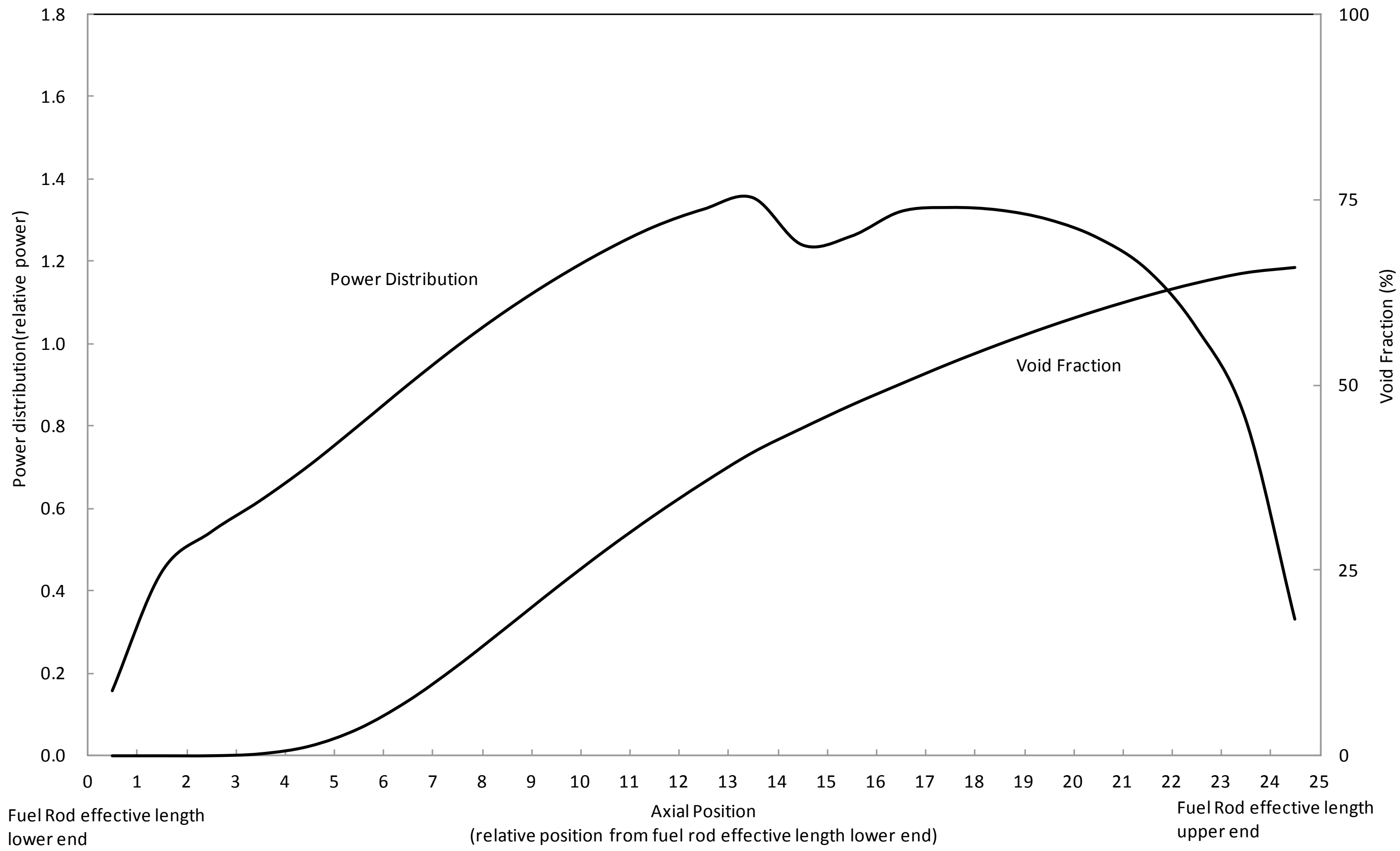

Figure 11.5-10 : Axial Power and Void Fraction Distribution (EOC)

11.5.5.2.Reactor Coolant System

Reactor Coolant System Configuration

The Reactor Coolant System is composed of components such as reactor coolant pressure boundary and reactor recirculation system. These systems are described in Generic PCSR Chapter 12: Reactor Coolant Systems, Reactivity Control Systems and Associated Systems.

Power/Flow Operating Map

(1) Limits for Normal Operation

The ABWR's thermal power and core flow conditions have certain restrictions because of overall plant control characteristics, core thermal power limits, etc. The power-flow map with 10 RIP's in operation is shown in Figure 11.5-11. This power-flow map illustrates the power range of operation used in the system response analyses. The nuclear system equipment, nuclear instrumentation, and the Reactor Protection System, in conjunction with operating procedures, operate within the area of the map for normal operating conditions. The boundaries on this map are as follows.

(a) **Minimum Pump Speed Line:** This line shows the change in flow associated with power changes while maintaining a constant RIP minimum speed of 30 percent.

(b) **Power Rod Line of 102 percent and 100 percent or Rated Power (Whichever is lower):** The 102 percent power rod line passes through 102 percent power at 90 percent flow. The operating state for the reactor follows this rod line (or similar ones) during recirculation flow changes with a fixed CR pattern; however, rated power may not be exceeded.

The 100 percent power rod line passes through 100 percent power at 90 percent core flow. This line defines the boundary of nominal power-flow operating points.

(c) **Steam Separator Limit Line:** This line results from the requirements to have an acceptable moisture carryover fraction from the steam separator.

(d) **Constant Pump Speed Lines:** These lines show the change in flow associated with power changes while constant RIP speeds are maintained.

(e) **Natural Circulation Line:** The operating state of the reactor moves along this line for the normal CR withdrawal sequence in the absence of recirculation pump operation.

The normal operation area for nominal power-flow with 10 RIP's in operation is defined by the lines of 100 percent power rod line, (1), (3) and (4).

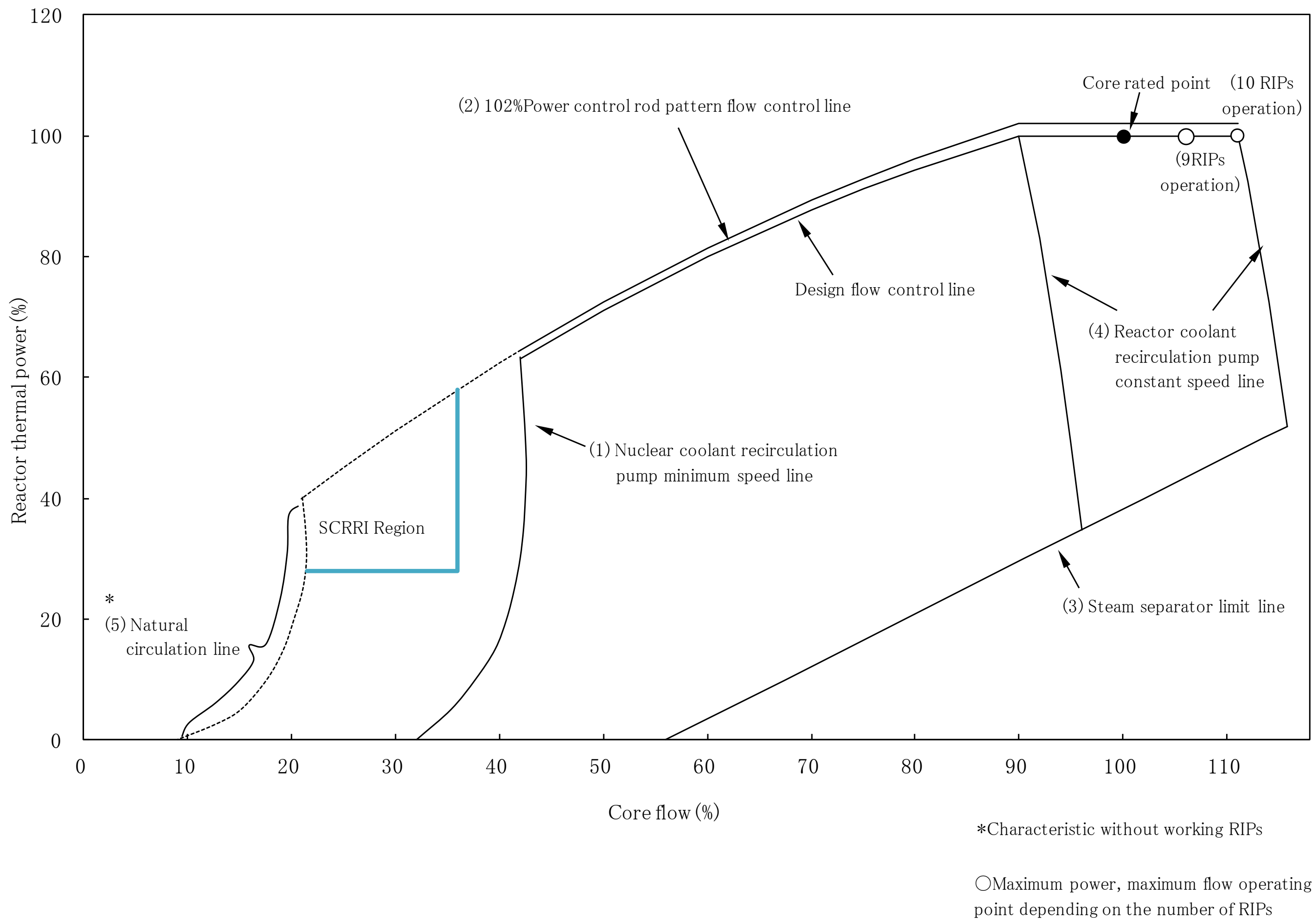

Figure 11.5-11 : Operating Power-Flow Operating Map

Flow Control

The normal plant start-up procedure requires the start-up of all 10 RIP's first, maintained at their minimum pump speed, at which point reactor heat-up and pressurisation can commence. When operating pressure has been established, reactor power can be increased. The system is then brought to the desired power/flow level within the normal operating area by increasing the RIP speeds and by withdrawing CRs.

CR withdrawal with constant pump speed will result in power/flow changes along lines of constant pump speed. Pump speed changes with constant CR position will result in power/flow changes along, or nearly parallel to, the rated power control line.

11.5.6 Core Monitoring System

A core monitoring system is one of components of the Process Computer and provides estimates of power distributions and thermal limits. These estimates are calculated by the core simulator based on three-dimensional diffusion theory and adapted to signals from plant Traversing Incore Probe (TIP) and Local Power Range Monitor (LPRM) instrumentation shown in Figures 2-1 and 2-2 of [Ref-11-21].

11.6 Design Evaluation

11.6.1 Fuel Assembly (FA)

The evaluation of the FA is described below. Further details are given in Section 2.3 and 3.3 of [Ref-11-5].

11.6.1.1.Fuel Rod Thermal-Mechanical Design

Evaluations Methods

Fuel rod thermal-mechanical evaluations are all performed using the PRIME03 fuel rod thermal-mechanical performance model ([Ref-11-11]). The stress/strain methodology is described later in this subsection.

(1) Fuel rod thermal-mechanical evaluations

The PRIME03 fuel rod performance model performs best estimate coupled thermal and mechanical analyses of a fuel rod experiencing a variable operating history. The model explicitly addresses the effects of:

- Fuel and cladding thermal expansion,
- Fuel and cladding creep and plasticity,
- Cladding irradiation growth,
- Cladding irradiation hardening and thermal annealing of irradiation hardening,
- Fuel irradiation swelling,
- Fuel irradiation-induced densification,
- Fuel cracking and relocation,
- Fuel hot pressing,
- Fission gas generation and exposure-enhanced fission gas release including fission product helium release,
- Differential axial expansion of the fuel and cladding reflecting axial slip or lockup of the fuel pellets with the cladding,
- Fuel phase change volumetric expansion upon melting, and
- The PRIME03 material properties and component models represent the latest experimental information available.

(2) Stress/Strain Analyses methods

The fuel rod cladding stress analyses are performed using a Monte Carlo statistical method in conjunction with distortion energy theory. Fuel cladding plasticity analyses are also performed when required by the loading conditions.

(3) Design basis power versus exposure limit (typical)

Design basis power versus exposure limit (typical) is:

- Used for all fuel rod thermal mechanical design analyses to evaluate the fuel rod design features and demonstrate conformance to the design criteria,
- Applied as a design constraint to the reference core loading nuclear design analyses, and
- Applied as an operating constraint to ensure that actual operation is maintained within the fuel rod thermal and mechanical design basis.

The proposed exposure limit is within the data used to validate the fuel performance codes used in fuel design and is well within the actual operating experience for the GE14 fuel design.

Design criteria

A set of design limits are defined and applied in the fuel rod thermal-mechanical design analyses to ensure that fuel rod mechanical integrity is maintained throughout the fuel rod design lifetime. The design criteria were developed to focus on the parameters most significant to fuel performance and operating occurrences that can realistically limit fuel performance.

(1) Cladding lift-off/Fuel rod internal pressure

The fuel rod internal pressure and cladding lift-off design ratio are determined statistically. The analysis is performed such that the fuel rod cladding will not creep outward at a rate greater than the fuel pellet irradiation swelling rate.

(2) Fuel Temperature

The fuel pellet centre-line temperature for the maximum duty fuel rod is statistically determined. The evaluations reflect operation on the bounding power-exposure operating envelop prior to the frequent design basis faults.

(3) Cladding strain

The fuel rod cladding circumferential plastic strain is determined by the 'worst case' analysis. Evaluations are performed reflecting continuous operation on the bounding power-exposure operating envelop prior to the frequent design basis faults. Based on the results of these evaluations, the mechanical overpower limits are applied to the fuel design to prevent cladding permanent (plastic plus creep) strain equal to or greater than the limit value.

(4) Dynamic loads /Cladding fatigue

The fuel rod is evaluated to ensure that the cumulative duty from cladding strains due to cyclic loadings will not exceed the cladding fatigue capability. The Zircaloy fatigue curve employed represents a statistical lower bound to the existing fatigue experiment measurements. The design limit for fatigue cycling is established such that the value of calculated fatigue usage is less than the material fatigue capability.

(5) Cladding creep collapse

Analysis results confirm that the fuel design will not experience cladding creep collapse.

(6) Fuel rod stresses /strains

The fuel rod is evaluated to ensure that fuel rod failure will not occur due to stresses or strains exceeding the fuel rod mechanical capability. In addition to the loads imposed by the difference between the external coolant pressure and the fuel rod internal gas pressure, a number of other stresses or strains can occur in the cladding tube. The value of the design ratio must be less than 1.0.

(7) Cladding hydrogen content

This evaluation (relative to hydriding of the fuel rod cladding) is based on the substantial operating and manufacturing experience to date with fuel designs fabricated to the same specifications limit on the amount of hydrogen permitted in a manufactured fuel rod. The experience with fuel manufactured since 1972 demonstrates that hydriding is not an active failure mechanism for current fuel designs.

(8) Cladding corrosion

The effects of cladding oxidation and corrosion product build-up (crud) on the fuel rod surface are included in the fuel rod thermal-mechanical design evaluations. The growth rate of the crud and the oxide thickness are input parameters for the statistical analyses. The input parameters for cladding corrosion are derived from data collected from plants with a range of saturation temperatures and from fuel operating over a wide range of powers. Reactor coolant system chemistry, which affects cladding corrosion, is discussed in Generic PCSR Chapter 23: Reactor Chemistry.

Evaluation results

Fuel rod thermal-mechanical evaluations have been completed for the GE14 fuel design using the methodologies described in this subsection. Further details are given in Section 2.3 of [Ref-11-5].

11.6.1.2.Cladding Overheating

It is traditional practice to assume that failures will not occur if the thermal margin criterion is satisfied. This is a conservative assumption for events that cause failures by high temperature cladding mechanisms. For BWR fuel, thermal margin is stated in terms of the minimum value of the critical power ratio (MCPR), which corresponds to the most limiting FA in the core. Adequate thermal margin is maintained by establishing an MCPR based on a statistical analysis. Further details are given in Subsection 11.6.6.3.

11.6.1.3.Pellet and Cladding Interaction

Measures to reduce damage due to local distortion of the cladding and deformation of the pellet include use of pellets with a short chamfer, use of cladding having a zirconium liner with large ductility on the inner wall, etc.
Tests have proven that zirconium has low sensitivity to stress corrosion cracking. Furthermore, mitigation of PCI events is based on the defined allowable design basis power versus exposure limit, core design practices, plant operating procedures, and fuel operating guidelines.

11.6.1.4.Fuel Assembly Mechanical Design

Design criteria

(1) Stress

The FA structural components are evaluated to ensure that the components will not fail due to stresses exceeding the FA component mechanical capability. The limits are typically applied to un-irradiated material conditions because irradiation increases the material strength properties. The stress limits are applicable to the combined effective stress.

(2) Fatigue

FA components with significant cyclic loading are evaluated to ensure that the material fatigue capability will not be exceeded.

(3) Fretting Wear

Testing is performed to assure that the mechanical features of the design do not result in significant vibration and consequent fretting wear. The vibration response of a new design is compared to a design that has demonstrated satisfactory performance through discharge burn-up. GE14 design vibration response was compared to the conventional design, the performance of which is considered acceptable based upon its performance in reactor operation, for this purpose.

Design loads

The structural adequacy of the FA components is demonstrated by evaluations (analysis or testing) that specifically address the operational duty that results from the BWR environment. This duty results from steady-state operation (including handling loads), mechanical loads associated with frequent design basis faults, and mechanical loads with infrequent design basis faults due to external conditions.

(1) Upper tie plate

The design loading occurs from bundle handling. Three times the bundle weight is applied at the tie plate handle to grapple attachment. The load is transmitted at the tie rod locations.

(2) Lower tie plate

The design loading occurs from bundle handling. The load is applied uniformly to the lower tie plate grid boss locations.

(3) Fuel rod end plug

The design loading occurs from bundle handling. The bundle weight with margin plus the sum of all the axial expansion spring forces are applied to the tie rods.

(4) Plenum spring

The plenum spring is designed to resist an acceleration of the fuel pellet column while being transported without deflecting the spring beyond a certain value.

(5) Expansion spring

The expansion spring is designed to resist downward forces from grappling and the weight of the suspended components (tie plate, fuel channel, and channel fastener) while allowing expansion from irradiation growth of the individual fuel rods and considering loss of load carrying capacity resulting from irradiation induced stress relaxation.

(6) Water rod

The water rod tubing is evaluated for a steady state differential wall pressure. The maximum load that a water rod tab can experience due to operating effects of spacer lift forces from flow or differential irradiation/thermal expansion between the fuel rods and water rods is the load required to simultaneously slide all fuel rods through a spacer.

(7) Spacer

Tests are performed to demonstrate that the GE14 spacer design can withstand significant lateral loading before any significant deformation occurs.

(8) Fuel channel

The design loads for the fuel channel include steady state and transient operating pressure differentials.

Evaluation results

The FA mechanical evaluations have been completed for the GE14 design using the methodologies described in this subsection. The evaluations demonstrate that the criteria are satisfied for the GE14 design. The following provides an explanation of the spacer mechanical test and flow induced vibration.

(1) Spacer mechanical test

The spacer mechanical tests were performed by assembling a test spacer onto a short section of a fuel bundle with empty fuel rods. The ends of the fuel rods were held in place by fixed lower tie plates. The test spacer was loaded by a section of fuel channel, which was attached and driven by the hydraulic piston of a tensile machine.

(2) Flow induced vibration

The GE14 FA was tested to assure that new features did not result in a significant increase in flow induced vibration (FIV) response and increase the potential for fretting wear. The test results demonstrated the acceptable performance of the GE14 FA.

11.6.1.5.Compatibility/Dimensional Changes

The GE14 FA has been mechanically designed to be compatible with the ABWR core configuration considering existing top guides, fuel supports, and fuel assemblies. In addition, allowances are made for dimensional changes in the FA components throughout the operating lifetime due to such considerations as irradiation growth and creep deformation.

The design limits for acceptable component dimensional changes are as follows:

(1) The fuel rod upper end plug initial engagement into the upper tie plate must be large enough such that at end–of–life the minimum growth fuel rod in the bundle will maintain engagement.

(2) The initial expansion space between the top of the fuel rod and the upper tie plate must be large enough such that at end–of–life the maximum growth fuel rod in the bundle will not compress an expansion spring beyond the solid height.

(3) The water rod upper end plug initial engagement into the upper tie plate must be large enough such that at end–of–life the cylindrical engagement with the tie plate is maintained.

(4) The water rod lower end plugs must not become disengaged from the lower tie plate during their lifetime, so the water rod lower end plugs are threaded and screw into the lower tie plate.

Hydrogen uptake associated with corrosion has the potential to cause fuel spacer lateral growth. The effects potentially include transverse loads on the fuel channel. In normal operation, there is some residual gap between the outer spacer perimeter and the inner channel surface at end-of-life. When the channel is removed from the FA for fuel rod inspection, some operational rule, for example limiting the channel lift load, will be proposed to prevent spacer and/or channel damage (e.g., stuck channel or broken spacer).

11.6.1.6.Fuel Channel Distortion Management

Fuel channel distortion may significantly affect operational and safety parameters. Specifically, the fuel channel distortion may lead to channel – control blade interference, which may obstruct the CR scram performance, and also lead to the decrease of thermal margin. In order to avoid the influence of fuel channel distortion, channel management, including evaluation of control blade cell friction and surveillance testing, will be performed.

(1) Two types of channel distortion, bulge and bow, are observed in BWRs and ABWR. Channel bulge is the result of elastic strain and irradiation creep strain from stress that results from the differential pressure across the channel wall. Bulge is always outward on all four sides of the channel since the pressure in-channel is greater than outside the channel. Bow is the result of a difference in length from one side of the channel to the opposite side. The channel bow distorts toward the longer side. Two distortion mechanisms cause channel bow independently; fluence gradient-induced bow and shadow corrosion-induced bow.

(2) Potential effects of channel distortion include mainly implications for CR scram performance and the thermal margin of fuel. CR scram performance is monitored through surveillance testing of CRs, which is performed by observing any significant CR friction when the CR is withdrawn. Scram time testing is periodically performed to verify the CRD scram performance and potentially any resistance to scram performance that might be caused by friction from channel distortion.
The impact of channel distortion on thermal margin is considered in the core design.

Further details are given in Section 3.5 of [Ref-11-5].

11.6.1.7.Manufacturing and Inspection of Fuel

Figure 11.6-1 shows the general manufacturing process of a fuel bundle.

Quality control is strictly performed during all stages of the fuel manufacturing process to confirm that design specifications are satisfied. Quality control for each stage is specified in the manufacturing process documents and the quality control plan.

For the pellets, properties of UO_2 powder, density, chemical composition, surface finish and etc. are checked. For the cladding, dimensional inspection, ultrasonic and other non-destructive tests are performed to detect defects in the cladding wall. Destructive tests, chemical analysis, tensile strength test, bursting test and etc. are also performed. Integrity of the end plug welding is checked by X-ray photography or ultrasonic tests. For the fuel rod, a helium leak test is performed to ensure that there is no leak of helium from the cladding and the end plug weld. During the assembly of the fuel bundle, each fuel rod serial number is checked against the fuel rod location plan to ensure that there is no mis-location. After assembling the fuel rods, dimensional and visual inspections are performed on key areas such as the clearance between fuel rods.

After being delivered to the installation site, the fuel bundle is re-inspected for deformation and its integrity confirmed.

The QATS (Quality at the Source) Programme is as follows:

(1) The Quality department certifies operations personnel to perform an in-process inspection to accept the material.
(2) Operations personnel have to meet minimum training requirements and demonstrate a proficiency in that role prior to being certified (as per the training and evaluation plan).
(3) The Quality department then performs surveillances to ensure that the product that is being manufactured is acceptable for release to the Customer.
(4) Surveillances are performed by Quality personnel (Quality Surveillance Controllers).

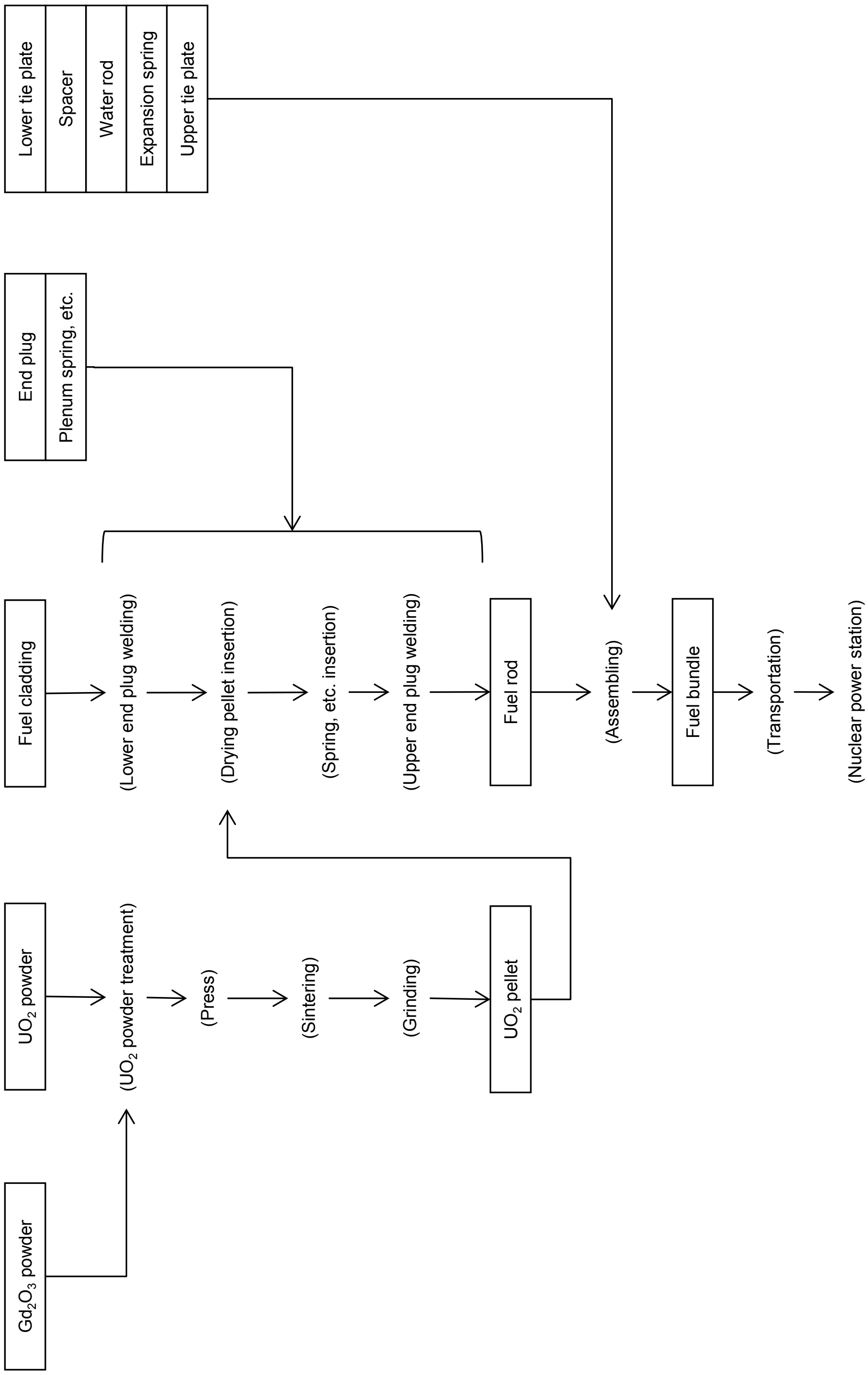

Figure 11.6-1 : Outline of manufacturing process of fuel

11.6.1.8.Post-irradiation Examination and Inspection

After each cycle of fuel irradiation, some of the fuel bundles are inspected to confirm their integrity. The fuel channel is detached from the FA for fuel inspection. After inspection, the fuel channel is re-attached to the fuel bundle and the fuel is reloaded into the core.
Generally, the post-irradiation examination of fuel is performed to confirm the expected fuel performance. The data from post-irradiation examination is reflected in the fuel design and safety analyses.

11.6.1.9.Fuel Reliability Record

GNF fuel has achieved superior reliability under the design and quality control programme described above. There have been no fuel failure events due to mechanical design problems or component failures, and no failures due to crud, corrosion, or flow induced vibration /wear. Most fuel failure events (almost 80 percent) have been due to debris fretting, but the introduction of an improved debris filter lower tie plate (Defender) has resulted in a reduction in failures from debris by approximately a factor of 5 in GNF 10×10 fuel. The only other significant failure mechanism observed has been a total of 11 PCI-type, "duty-related" failures, in a small number of reactor manoeuvres (most > 10 years ago) which have been mitigated through several measures including improved operating guidelines and improved pellet quality. The FMCRD system of the ABWR inherently includes more margin to these type of failures. In case of damaged fuel, BWRs are capable of suppressing the power in the vicinity of the damaged fuel and continue operation without further fuel degradation (see Chapter 4 of [Ref-11-5]).

11.6.2 Control Rod (CR)

A set of acceptance criteria has been established for evaluating CR designs. The requirements on CR and their bases are provided below.

11.6.2.1.Requirements for Design Basis

The following comprise the requirements for design basis given in Subsection 11.4.2.5.

Stresses

The CR is evaluated to assure that it does not fail because of loads due to shipping, handling, and normal, abnormal, emergency, and faulted operating modes. To assure that the CR does not fail, these loads must not exceed the allowable limit for the core support structure as a guideline.
The evaluated loads include those due to normal operational transients [scram and jogging (step wise motion)], pressure differentials, thermal gradients, flow- and system-induced vibration, and irradiation growth in addition to the lateral and vertical loads expected for each condition. Fatigue usage is based upon the cumulative effect of the cyclic loadings. The analyses include corrosion and crud deposition as a function of time as appropriate.

Conservatism is included in the analyses by including a margin to the limit or by assuming loads greater than expected for each condition. Higher loads can be incorporated into the analyses by increasing the load itself or by statistically considering the uncertainties in the value of the load.

Control Rod Insertion

The CR is evaluated to confirm that it can be inserted during normal, abnormal, emergency and faulted modes of operation assumed in the plant analyses and throughout its design life. These evaluations include a combination of analyses of the geometrical clearance and actual testing.

Control Rod Material

The external CR materials must be capable of withstanding the reactor coolant environment for the design life of the CR. Effects of crud, crevices, stress corrosion and irradiation upon the material must be included in the CR and core evaluations.
Irradiation effects to be considered include material hardening and absorber depletion and swelling.

Reactivity

The reactivity worth of the CR is determined by the initial amount and type of absorber material and irradiation depletion. Scram time insertion performance and CR drop times affect the total reactivity inserted into the core. All of these effects are included in the plant core analyses including frequent and infrequent design basis faults. The reactivity worth of the rod provides, under conditions of normal operation and frequent design basis faults, appropriate margin for malfunctions, such as two stuck CRs or accidental CR withdrawal, without exceeding specified acceptable fuel design limits. Nuclear design limit of the CR is defined as 10 percent reduction in relative CR worth; in other words, the depletion of the neutron absorbing material which meets design criteria.

11.6.2.2.Evaluation Results

The evaluations demonstrate that the criteria are satisfied for the reference CR design as follows.

Stress

Details are given in Section 3 of [Ref-11-20].

Control Rod Insertion

Details are given in Generic PCSR Chapter12 Subsection 12.4.3.1: Control Rod Drive System.

Control Rod Material

Details are given in Chapter 3 of [Ref-11-20].

Reactivity

Details are given in Subsection 11.6.5.

Considering the above criteria, the CR lifetime is managed appropriately, which is applied even at power suppression operation. Details are given in Section 2 and 3 of [Ref-11-20] and Section 5 of [Ref-11-5].

11.6.3 Control Rod Drive (CRD) System

The evaluation of CRD system has been completed to confirm the requirements are satisfied. Details are given in Chapter 6 of [Ref-11-23].

11.6.4 Standby Liquid Control System (SLC)

The evaluation of SLC has been completed to confirm the requirements are satisfied. Details are given in Subsection 11.6.5.3 for shutdown capability and in Chapter 6 of [Ref-11-24] for the remaining design requirements.

11.6.5 Nuclear Design

The evaluation of nuclear design is described below. Further details are given in Section 3.3 of [Ref-11-6] and [Ref-11-7].

11.6.5.1.Power Distribution

The core power distribution is a function of fuel bundle design, core loading, CR pattern, core exposure distributions and core coolant flow rate. The thermal performance parameters, MLHGR and MCPR, assure that the fuel thermal design limits will not be exceeded during operation.

Power Distribution Anomalies

Stringent inspection procedures are utilised to ensure the correct arrangement of the core following fuel loading. A fuel loading error (a mislocated or a misoriented fuel bundle in the core) would be a very improbable event (Section B.7 of [Ref-11-25]), but calculations have been performed to determine the effects of such events on CPR.

The inherent design characteristics of the BWR are well suited to limit gross power tilting. The stabilising nature of the large moderator void coefficient effectively reduces the effect of perturbations on the power distribution. In addition, the in-core instrumentation system, together with the online computer, provides the operator with prompt information on the power distribution so that the operators can readily use CRs or other means to limit the undesirable effects of power tilting. Because of these design characteristics, it is not necessary to allocate a specific margin in the peaking factor to account for power tilt. In the unlikely event that the power distribution could not be maintained within normal limits using CRs and flow, then the total core power would need to be reduced.

11.6.5.2.Reactivity Coefficients

Reactivity coefficients, the differential changes in reactivity produced by differential changes in core conditions, are useful in assessing the severity of reactor response to instability and transient events. The reactivity coefficients are classified as follow:

(1) Fuel rod temperature coefficient (Doppler coefficient)
(2) Moderator void coefficient
(3) Moderator temperature coefficient
(4) Power reactivity coefficient

Reactivity coefficients ((1), (2) and (3)) are combined to give the power reactivity coefficient.

The system is designed so that the power reactivity coefficient is always negative during the cycle.

(1) Fuel rod temperature coefficient (Doppler coefficient)

The Doppler effect occurs with an increase of uranium 238 neutron absorption. This effect plays an important role in the dynamic characteristics and safety of the reactor. If the power is suddenly increased, the UO_2 fuel temperature is increased but negative reactivity is inserted because of the Doppler effect associated with temperature increase, and a further increase in power is prevented. Therefore, if the power is suddenly increased by means of a large reactivity insertion, a large negative Doppler effect is immediately generated to prevent a nuclear excursion. The Doppler effect plays an important role for reactor safety.

The insertion of reactivity from the Doppler effect is given as the reactivity change when parameters other than the fuel temperature are fixed and only the fuel temperature is changed. It is accurately expressed with the formula below:

$$\Delta k_{DOP} = C(\sqrt{T} - \sqrt{T_0}) \qquad (11.6\text{-}1)$$

Δk_{DOP}: Reactivity change when the fuel temperature is changed from T_0 to T (reactivity insertion with Doppler effect)

C: Constant

Therefore, the Doppler coefficient is expressed as follows:

$$\frac{1}{k}\cdot\frac{dk}{dT}=\frac{C}{2[k_0+C(\sqrt{T}-\sqrt{T_0})]\sqrt{T}} \quad (11.6\text{-}2)$$

C: Constant

k_0: Reactivity when temperature is T_0

The absolute value of the Doppler coefficient is larger when the fuel temperature is low. The absolute value is increased if the density of the moderator is reduced due to an increase of moderator temperature or voiding.

When plutonium 240 is accumulated throughout the cycle, the Doppler coefficient is negative and the absolute value increases.

(2) Moderator void coefficient

In normal operation, the core pressure is maintained at a constant value. As a result, the coolant temperature is constant regardless of the power level, except for minor variation in the sub-cooled zone. The void fraction varies depending on the power level and the core flow rate. Adjusting void fraction (or moderator density) by means of core flow changes is frequently done during power operation.

The moderator void coefficient has a large negative value and it is effective in mitigating the power increase during reactivity insertions.

The moderator void coefficient can be expressed as follows:

$$\frac{1}{k_{eff}}\cdot\frac{dk_{eff}}{dV} = \frac{1}{k_{eff}}\left[(1-C)\frac{d}{dV}\left(\frac{k_\infty^{UC}}{1+M^2B^2}\right)+C\frac{d}{dV}\left(\frac{k_\infty^{C}}{1+M^2B^2}\right)\right] \quad (11.6\text{-}3)$$

V: Channel void fraction

C: CR insertion ratio

M^2: Neutron migration area

B^2: Buckling

k_∞: Infinite multiplication factor (UC: without CR, C: with CR)

k_{eff}: Effective multiplication factor

As moderator density decreases, the neutron leakage increases and the CR worth becomes greater. These two effects always have negative impacts on the core. As the moderator density is reduced, the absolute values of the neutron leakage and the CR worth are increased monotonically. The CR insertion ratio C is reduced along with the decrease in the CR fraction; this reduction occurs throughout the operating cycle with increasing burnup.

A detailed discussion of the methods used to calculate void reactivity coefficients, their accuracy and their application to plant transient analyses, is presented in [Ref-11-12].

(3) Moderator temperature coefficient

The moderator temperature coefficient varies depending on temperature and core burn-up, but the reactivity is small in magnitude. Considering delay of heat transfer from the fuel to the coolant as well as magnitude, this coefficient does not have any significant impact on fast transients.

Therefore, during an operating cycle the moderator temperature coefficient is of minor concern from a safety point-of-view due to the relatively slow nature of moderator temperature change relative to fuel temperature change. Also, the coefficient is small when compared to void reactivity feedback such that at normal operating conditions the core dynamic behaviour will be dominated by the negative void coefficient.

The moderator temperature coefficient is not specified in design.

(4) Power reactivity coefficient

The power reactivity coefficient is the integral of all factors of variations in reactivity due to minimal change of the core thermal power. According to the analyses performed for the initial and equilibrium cycles, the power reactivity coefficient at rated power operation differs by approximately -0.03 $(\Delta k/k)/(\Delta p/p)$ throughout the cycles. The magnitude of the power reactivity coefficient is sufficient for damping power oscillations, including the spatial oscillation of xenon.

11.6.5.3.Control Requirements

The ABWR CR system is designed to provide adequate control of the maximum excess reactivity anticipated during plant operation. The shutdown capability is conservatively evaluated assuming a cold and xenon-free core.

Shutdown Reactivity

The core must be capable of being made sub-critical, with margin, in the most reactive condition throughout the operating cycle with the highest worth CR or any CR pair with the same hydraulic control unit (HCU), fully withdrawn and all other rods fully inserted. The shutdown margin is determined by using the BWR simulator code ([Ref-11-14]) to calculate the core multiplication factor at selected exposure points with the strongest rod or strongest rod pair with the same HCU fully withdrawn. The core is assumed to be in the cold, xenon-free condition in order to ensure that the calculated values are conservative.

Reactivity Variations

The excess reactivity designed into the core is controlled by the CR system and supplemented by gadolinia-urania fuel rods. CRs are used during the cycle partly to compensate for burnup and partly to control the power distribution.

Standby Liquid Control System (SLC)

The SLC shutdown margin is evaluated and confirmed to meet the design limit. Further details are given in Subsection 3.3.9.3 of [Ref-11-6] and [Ref-11-7].

11.6.5.4.Maximum Reactivity Worth and Reactivity Insertion Rate of Control Rod

For the maximum reactivity worth and the reactivity insertion rate of the CR, the following restrictions are applied:

(1) For the selection of the operating CR pattern and the CR withdrawal sequence, the reactivity worth for simultaneous withdrawal of multiple CRs belonging to the same CR group (i.e., ganged withdrawal) while approaching criticality is less than the maximum worth limited by the withdrawal procedures. If one CR drops while approaching criticality, the maximum worth of the dropped CR is less than the maximum worth limited by the withdrawal procedures. A dropped CR shall not damage the reactor coolant pressure boundary and shall not adversely affect the core, core support structure and pressure vessel internal structure such that core cooling is impaired.
For ganged withdrawal, the above maximum CR group worth limited by the withdrawal procedures is considered in the safety analysis of Rod Withdrawal Error. The above maximum single rod worth limited by the withdrawal procedures is considered in the safety analysis of CR Drop Accident.
(2) The CR movement step is designed so that an operator can safely control the reactor with an adequate reactor period when multiple CRs belonging to the same withdrawal group are simultaneously moved step by step.

11.6.5.5.Stability

Xenon Transients

Boiling water reactors do not have instability problems due to xenon. This has been demonstrated by:

(1) Never having observed xenon instabilities in operating BWRs.
(2) Special tests that have been conducted on operating BWRs in an attempt to force the reactor into xenon instability.
(3) Calculations.

All of these indicators have proven that xenon transients are highly damped in a BWR due to the large negative power coefficient. Experiments conducted in this area are reported in [Ref-11-13].

Thermal Hydraulic Stability

Thermal hydraulic stability is described in Subsection 11.6.6.4.

11.6.5.6.Effect of Xenon

The maximum reactivity due to xenon is attained when the reactor is shut down from rated power operation. The burn-up fraction of xenon, i.e., the maximum reactivity increase ratio due to xenon burn-up, is obtained at the maximum xenon concentration assuming that a sudden change of the reactor operation from the standby condition to rated power operation occurs.
Based on the above scenario, the reactivity increase rate due to xenon burn-up is calculated to be about 0.0001Δk/min. The reactivity decrease rate due to insertion of the CR, on the other hand, may be expected to be about 0.0005Δk/min. Therefore, the xenon burn-up may be controlled with sufficient margin by inserting the CR.

In designing the Standby Liquid Control System used for backup shutdown of the reactor, the required injection speed of the boric acid solution is determined considering the possible increase of the reactivity due to xenon decay.

11.6.5.7. Analytical Methods

The ABWR core design is performed using the analytical tools (TGBLA06 and PANAC11) and methods that are used for the steady-state and kinetic-state nuclear evaluations of BWR cores. These nuclear physics methods are described in detail in [Ref-11-14].

TGBLA06 is a lattice design computer programme for conventional BWRs that can model the following lattices: 7×7, 8×8, 9×9, and 10×10. TGBLA06 solves 2-D diffusion equations with diffusion parameters corrected by transport theory to provide the multiplication factor, the fission density distribution, the neutron balance, and the homogenised cross-sections. TGBLA06 also performs burn-up calculations for generating input to the PANAC11.

PANAC11 is a static, three-dimensional coupled nuclear-thermal-hydraulic computer programme representing the BWR core exclusive of the external flow loop. The programme is used for detailed three-dimensional design and operational calculations of BWR neutron flux and power distributions and thermal performance as a function of control rod position, refuelling pattern, coolant flow, reactor pressure, and other operational and design variables. The nuclear model is based on coarse-mesh nodal, one-group, static diffusion theory. One energy group is used to represent fast energy neutron diffusion. Resonance energy and thermal energy neutronic effects are included in the model by relating the resonance and thermal energy fluxes to the fast energy flux. Eigenvalue iteration yields the fundamental mode solution. This is coupled to static parallel channel thermal-hydraulics containing a modified Zuber-Findlay void-quality correlation. Pressure drop balancing yields the flow distribution.

11.6.6 Thermal Hydraulic Design

The evaluation of thermal hydraulic design is described below. Further details are given in Section 4.3 of [Ref-11-6] and [Ref-11-7].

11.6.6.1.Core Pressure Drop and Hydraulic Loads

The components of bundle pressure drop considered are friction, local, elevation and acceleration pressure drops. Pressure drop measurements made in operating reactors confirm that the total measured core pressure drop and calculated core pressure drop are in good agreement.

Friction Pressure Drop

Friction pressure drop is calculated with a basic model as follows:

$$\Delta P_f = \frac{w^2}{2g\rho}\frac{fL}{D_H {A_{ch}}^2}\phi_{TPF}^2 \qquad (11.6\text{-}4)$$

where

ΔP_f	=	friction pressure drop
w	=	mass flow rate
g	=	acceleration of gravity
ρ	=	average nodal liquid density
D_H	=	channel hydraulic diameter
A_{ch}	=	channel flow area
L	=	increment length
f	=	friction factor
ϕ_{TPF}	=	two-phase friction multiplier

The single phase friction factor and two phase friction multiplier were validated by comparisons to full scale bundle pressure drop test data by GNF.

Local Pressure Drop

The local pressure drop is defined as the irreversible pressure loss associated with an area change, such as the orifice, lower tie plate, and spacers of a FA.

The general local pressure drop model is similar to the friction pressure drop and is expressed as:

$$\Delta P_L = \frac{w^2}{2g\rho}\frac{K}{A^2}\phi_{TPL}^2 \qquad (11.6\text{-}5)$$

where

ΔP_L	=	local pressure drop
K	=	local pressure drop loss coefficient
A	=	reference area for local loss coefficient
ϕ_{TPL}	=	two-phase local multiplier

and w, g_c, and ρ are defined above. The formulation for the two-phase multiplier is similar to that reported in [Ref-11-15]. The local loss component of the total pressure drop across a region inside the FA is deduced from the measured total pressure drop by subtracting the frictional, elevation, and acceleration components. The corresponding local loss coefficient is then determined using the above formula. These data were obtained from tests performed in single-phase water to calibrate the orifice, the lower tie plate, and the holes in the lower tie plate, and in both single- and two-phase flow to derive the best fit design values for spacer and upper tie plate pressure drop. The range of test variables was specified to cover the range of interest for the ABWR.

Elevation Pressure Drop

The elevation pressure drop is based on the relation:

$$\Delta P_E = \bar{\rho}\Delta L$$

$$\bar{\rho} = \rho_f(1-\alpha) + \rho_g\alpha \qquad (11.6\text{-}6)$$

where

ΔP_E	=	elevation pressure drop

ΔL = incremental length

$\bar{\rho}$ = average mixture density

α = nodal average void fraction

ρ_f , ρ_g = liquid and saturated vapour density, respectively

The void fraction model used is an extension of the Zuber-Findlay model [Ref-11-16], and uses an empirical fit constant to predict a large block of steam void fraction data.

Acceleration Pressure Drop

A reversible pressure change occurs when an area change is encountered, and an irreversible loss occurs when the fluid is accelerated through the boiling process. The basic formulation for the reversible pressure change resulting from a flow area change in the case of single-phase flow is given by:

$$\Delta P_{ACC} = \left(1 - \sigma_A^2\right)\frac{W^2}{2g\ \rho_f A_2^2}$$

$$\sigma_A = \frac{A_2}{A_1} = \frac{final\ flow\,area}{initial\ flow\,area} \tag{11.6-7}$$

where

ΔP_{ACC} = acceleration pressure drop

ρ_f = liquid density

A_2 = final flow area

A_1 = initial flow area

w = mass flow rate

In the case of two-phase flow, the liquid density is replaced by a density ratio so that the reversible pressure change is given by:

$$\Delta P_{ACC} = \left(1 - \sigma_A^2\right)\frac{w^2 \rho_H}{2g\ \rho_{KE}^2 A_2^2} \tag{11.6-8}$$

where

$$\frac{1}{\rho_H}=\frac{x}{\rho_g}+\frac{1-x}{\rho_f}$$, homogeneous density

$$\frac{1}{\rho_{KE}^2}=\frac{x^3}{\rho_g^2\alpha^2}+\frac{(1-x)^3}{\rho_f^2(1-\alpha)}$$, kinetic energy density (11.6-9)

α = void fraction at A_2

x = steam quality at A_2

and other terms are as previously defined. The basic formulation for the acceleration pressure change due to density change is:

$$\Delta P_{ACC}=\frac{w^2}{g_c A_{ch}^2}\left(\frac{1}{\rho_{out}}-\frac{1}{\rho_{in}}\right) \quad (11.6\text{-}10)$$

where

ρ_{out} = outlet coolant density

ρ_{in} = inlet coolant density

where ρ is either the homogeneous density, ρ_H, or the momentum density, ρ_M

$$\frac{1}{\rho_M}=\frac{x^2}{\rho_g\alpha}+\frac{(1-x)^2}{\rho_f(1-\alpha)} \quad (11.6\text{-}11)$$

and is evaluated at the inlet and outlet of each axial node. Other terms are as previously defined. The total acceleration pressure drop in the ABWR is of the order of a few percent of the total pressure drop.

11.6.6.2. Correlation and Physical Data

Substantial amounts of physical data in support of the determination of pressure drop and thermal hydraulic loads discussed in Subsection 11.6.6.1 have been obtained. Correlations have been developed to fit these data to the formulations discussed.

Pressure Drop Correlations

Significant amounts of friction pressure drop data have been taken in multi-rod geometries representative of BWR fuel bundles and correlated to both the friction factor and two-phase multipliers on a best fit basis using the pressure drop formulations described in Subsection 11.6.6.1 Tests were performed in single-phase water to calibrate the orifice and the lower tie-plate, and in both single- and two-phase flow to arrive at best fit design values for spacer and upper tie plate pressure drop. The range of test variables was specified to cover the range interest for the ABWR. New data are taken whenever there is a significant design change to ensure the most applicable methods are in use at all times.

Applicability of the single-phase and two-phase hydraulic models previously discussed in Subsection 11.6.6.1 for the GE14 fuel design described in Subsection 11.5.1, was confirmed by full scale prototype flow tests.

Void Fraction Correlation

The void fraction correlation includes effects of pressure, flow direction, mass velocity, quality, and subcooled boiling.

Heat Transfer Correlation

The Jens-Lottes [Ref-11-17] heat transfer correlation is used in fuel design to determine the cladding-to-coolant heat transfer coefficients for nucleate boiling.

11.6.6.3. Critical Power Ratio (CPR) Calculation

CPR Calculation is described below. Further details are given in Chapter 4 of [Ref-11-18].

Critical Power

Operating limits are specified to maintain adequate margin to the onset of boiling transition. The figure of merit utilised for plant operation is the critical power ratio (CPR). This is defined as the ratio of the critical power (bundle power at which some point within the assembly experiences the onset of boiling transition) to the operating bundle power. The critical power is determined at the same mass flux level, inlet temperature, and pressure which exist at the specified reactor condition. Thermal margin is stated in terms of the minimum value of the critical power ratio (MCPR), which corresponds to the most limiting FA in the core. To ensure that adequate margin is maintained, a design requirement based on a statistical analysis was selected as follows:

Considering uncertainties in manufacturing and monitoring the core operating state, at least 99.9 percent of the fuel rods would be expected to avoid boiling transition.

Fuel Cladding Integrity Safety Limit

The generation of the Minimum Critical Power Ratio (MCPR) limit requires a statistical analysis of the core near the limiting MCPR condition. The MCPR Fuel Cladding Integrity Safety Limit applies not only for core wide frequent design basis faults, but also applies to the localised rod withdrawal error. Uncertainties in thermal hydraulic parameters are considered in the statistical analysis below.

(1) Statistical Model

The statistical analysis utilises a model of the BWR core that simulates the process computer function. This code produces a critical power ratio (CPR) map of the core based on inputs of power distribution, flow and heat balance information. Random Monte Carlo selections of all operating parameters based on the uncertainty ranges of manufacturing tolerances, uncertainties in measurement of core operating parameters, calculation uncertainties, and statistical uncertainty associated with the critical power correlations are imposed upon the analytical representation of the core and the resulting bundle critical power ratios are calculated.

The minimum allowable critical power ratio is set to correspond to the criterion that 99.9 percent of the rods are expected to avoid boiling transition by interpolation among the means of the distributions formed by all the trials.

(2) Bounding BWR Statistical Analysis

Statistical analyses have been performed that provide fuel cladding integrity safety limit MCPRs for the ABWR equilibrium and initial core cycles. The results of the analyses show that at least 99.9 percent of the fuel rods in the core are expected to avoid boiling transition if the MCPR is equal to or greater than a certain value. This value is called Safety Limit MCPR (SLMCPR). Typical values are 1.06 to 1.09.

(3) MCPR Operating Limit Calculation

A plant-specific MCPR operating limit is established to provide adequate assurance that the fuel cladding integrity safety limit for that plant is not violated for frequent design basis faults. This operating requirement is obtained by the addition of the limiting ΔCPR value for frequent design basis faults to the SLMCPR.

(4) Evaluation Procedure for Frequent Design Basis Faults

The evaluation of frequent design basis faults are described in Generic PCSR Chapter 24: Design Basis Analysis.

GEXL Correlation

The GEXL correlation equation is used to predict the point where boiling transition starts using critical steam quality (critical quality) and is expressed as follows:

$$X_c = f(L_B, D_Q, G, L, L_A, P, R) \qquad (11.6\text{-}12)$$

where,

X_c:	Critical quality
L_B:	Boiling length
D_Q:	Thermal diameter (i.e., 4A/(total rodded perimeter))
G:	Mass flow rate per unit area
L:	Heated length
L_A:	Annular flow length
P:	System pressure
R:	Parameter that characterizes local peaking in a bundle

The influence of the axial power distribution on critical quality is characterised by the length of boiling and the influence of local power distribution via the R-factor. Critical quality is defined by a random point in the axial direction of a FA, and boiling transition begins when actual steam quality at that point is larger than the critical quality. The FA power at that time is defined as the critical power.

Evaluation Results

The MCPR for the GDA reference core design is evaluated and confirmed to meet the MCPR operating limit with a margin.

11.6.6.4.Thermal Hydraulic Stability Performance

In general, BWRs have a large negative power coefficient and inherent stability against disturbances of reactivity caused by various operations such as CR movement. On the other hand, as BWRs have a positive pressure coefficient, power is controlled by adjusting the recirculation flow rate with constant steam pressure. Also, design features such as a void reactivity coefficient designed not to be excessively negative, forced circulation that prevents hydraulic disturbances, sintered uranium dioxide pellets with sufficient heat conductivity, and introduction of part length fuel rods that contribute to reduced two-phase pressure drop ratio, contribute to the stability.

The ABWR has the following design features compared to previous BWR plant types.

(1) A 154.9 mm fuel lattice pitch that is wider than a BWR5 lattice pitch of 152.4mm lattice (i.e., the ABWR has a larger intra-assembly gap). This provides more non-boiling area in the core and less negative void coefficient.

(2) A smaller core inlet orifice diameter compared to the standard BWR5. This results in a larger single phase pressure drop ratio (i.e., a larger single phase to two phase pressure drop ratio).

(3) The inner width of the fuel channel is the same as the early BWR3/4 type fuel and is larger than the latest BWR5 fuel. This contributes to a lower core pressure drop.

(4) A larger number of low ΔP steam separators (AS-2B). This reduces the two phase pressure drop of the recirculation system.

With the characteristics described above, the ABWR has sufficient suppression capability against power oscillation within the normal operating domain area specified in Subsection 11.5.5.2.

For high power and low core flow conditions outside the normal operating domain, the probability of generating a sustainable power oscillation is relatively small. To increase the stability margin and to prevent power oscillation, the RPS (Reactor Protection System) scram "APRM simulated thermal power high" and selected CRs run-in are automatically initiated at high core power / low core flow conditions. The initiation region of this "selected CRs run-in" feature (SCRRI) is shown in Figure 11.5-11.

As such, ABWR stability characteristics will be confirmed with analyses at the intersection of the 102 percent power rod line and the minimum pump speed line. The results of the analysis will determine the core decay ratio and channel decay ratio that satisfy the design criteria (decay ratio). The magnitude of the core and channel decay ratio provides an indication of the propensity of the core power to oscillate in either a core-wide (in-phase) or regional (out-of-phase) mode, respectively. Furthermore, the fuel integrity is maintained and the reactor is led to a stable condition by the RPS scram and SCRRI even if the high power and low core flow region is inadvertently entered. Further details are given in Subsection 4.4.3 of [Ref-11-6] and [Ref-11-7], and in [Ref-11-19].

11.6.7 Core Monitoring System

The uncertainties of core monitoring system were evaluated by the comparisons of measurements and predictions. These uncertainties are less than the assumption used in the core design methodology. Further details are given in Chapter 4 of [Ref-11-21] and in [Ref-11-26]

11.7 Assumptions, Limits and Conditions for Operation

11.7.1 Purpose

One of purpose of this PCSR is to identify the constraints that must be applied by a future licensee of the UK ABWR plant to ensure safety operation during normal, fault and accident conditions. Some of these constraints are maximum or minimum limits on the values of the plant parameters (e.g. pressure or temperature) whilst others are conditional, which prohibit certain operational states or require a minimum level of availability of a specified equipment. Those constraints are collectively defined in this GDA PCSR as Assumptions, Limits and Conditions for Operation (LCOs).

This section considers the LCOs that apply specifically to the Reactor Core of the generic UK ABWR design reference. These LCOs have been identified following the standard procedure for GDA detailed in [Ref-11-28], and are collated in the Generic Technical Specifications [Ref-11-29]. Note that [Ref-11-29] is only systematically applied to safety Class 1 and 2 systems, which are within the scope of GDA.

11.7.2 LCOs Specified for Reactor Core and Fuel

The list below is based on the Topic Report [Ref-11-27]that specify the LCOs.

- Shutdown Margin (SDM)－The reactor core shall be subcritical at all operation modes with the highest worth control rod or control rod pair fully withdrawn and all other control rods fully inserted, based on Surveillance (SR) verifying SDM.

- Reactivity Anomalies－The reactivity difference between the monitored and predicted value shall be within criterion based on SR verifying reactivity difference.

- Control Rod Programme Controls－All operable control rods shall comply with the requirement at low power based on SR verifying all control rods comply with predetermined withdrawal sequence restrictions.

- Minimum Critical Power Ratio (MCPR)－All MCPRs shall be greater than or equal to the operating limits based on SR verifying MCPRs.

- Linear Heat Generation Rate (LHGR) －All LHGRs shall be less than or equal to the limits based on SR verifying LHGRs.

- Plant Control System－The control rod withdrawals shall be monitored to prevent overpowering a fuel rod based on SRs verifying Control Rod Block function.

- Refueling Equipment Interlocks－During fuel movement and before refueling operation, Refueling Equipment Interlocks shall be operable based on SRs verifying reactor mode switch.

- Reactor Coolant System (RCS) Specific Activity － These limits ensure the specific iodine activity and gross specific activity are kept below DBA assumption, based on SRs verifying reactor coolant I-131 and gross specific activity.

- Main Condenser Off-gas － This limit ensures the gross gamma activity rate of the noble gases is kept below DBA assumption, based on SR verifying gross gamma activity rate of the noble gases.

11.7.3 Assumptions Specified for Reactor Core and Fuel

The following are the key assumptions made in each of the systems structures and components (SSCs) described within this chapter.

- Applied Fuel Design is GE14, and
- Operating Power-Flow Map is as assumed in DBA analysis.

11.8 Summary of ALARP Justification

This section presents a high level overview of how the ALARP principle has been applied for the fuel and core, and how this contributes to the overall ALARP argument for the UK ABWR.

Generic PCSR Chapter 28: ALARP Evaluations presents the high level approach taken for demonstrating ALARP across all aspects of the design and operation. It presents an overview of how the UK ABWR design has evolved, the further options that have been considered across all technical areas resulting in a number of design changes, and how these contribute to the overall ALARP case. The approach to undertaking ALARP Assessment during GDA is described in the GDA ALARP Methodology [Ref-11-30] and in the Safety Case Development Manual [Ref-11-31]

The most significant nuclear safety risks associated specifically with the fuel (whilst in the core) and other core components are:

- Fuel failures during normal operation or in fault conditions – in which breaches of the fuel cladding lead to fission product release to the reactor coolant, and possibly fuel washout in the most severe cases,
    - The potential root causes of fuel failures in normal operation include debris fretting, Pellet-Cladding Interaction (PCI) during load manoeuvres, fuel manufacturing defects, flow induced vibrations, and cladding corrosion.
    - Additional root causes during faults include overheating of the cladding, excessive cladding strain, significant clad oxidation leading to embrittlement in Loss of Coolant Accidents, or sudden rupture in Reactivity Insertion Accidents.
- Insufficient shutdown capability of the control rods, and
- Inability to insert control rods as required due to loss of structural integrity of the rods or interference of the control blades with the surrounding fuel channels.

It should be noted that this chapter does not discuss the ALARP evaluations of any fuel handling operations or interim storage of spent fuel. Such discussions are included in PCSR chapters 19 and 32 respectively.

Extensive work has been performed to minimise the risks of fuel failure by improvements that have been introduced during successive fuel development programmes. These include the following aspects:

- Fuel Assembly Design,
  - Fuel pellet (including use of Gadolinia as burnable poison)
  - Cladding
  - Fuel assembly configuration
  - Debris filter
  - Channel box
  - Soft duty rules (operational rules to mitigate PCI failures)
  - Management of damaged fuel (to prevent the degradation and reduce the risk of worker exposure)
- Control Rods,
- Nuclear and Thermal-hydraulic design, and
  - Core configuration and loading pattern
  - Reactivity control with core flow
  - Instability countermeasure
- Core Monitoring System.

For each aspects listed above, the ALARP considerations include the adoption of relevant good practice and potential options for risk reduction. These are briefly discussed in the following paragraphs. References are also provided.

Fuel Pellet Design

One of the drivers for the evolution in fuel pellet design is the protection against PCI. PCI failures have been correlated to fuel rods with chipped pellets (or areas of MPS, Missing Pellet Surface). For this issue, a tighter MPS specification was established to eliminate the flaws associated with PCI failures, with multiple inspections established to assure compliance with the specification. The introduction of larger chamfered pellet edges has further reduced the likelihood that pellets will be damaged during the manufacturing process.
Further discussions about the reference design are provided in Section 2.3.9 of [Ref-11-5].

Following decades of manufacturing and operational experience gadolinium oxide (Gd_2O_3), or gadolinia, has become the industry standard material for use as a burnable neutron absorber (poison) in BWRs. Gadolinia is effective for additional control during shutdown and for power shaping particularly during early parts of the cycle. Gadolinia was selected for use in the UK ABWR over

alternative burnable poisons because the residual gadolinia after burn-up is smaller compared to others.
Further discussions about the reference design are provided in Section 1.1.1 of [Ref-11-5] and Section 3.3.9.1 of [Ref-11-7].

Cladding

GNF offers Zircaloy-2 as cladding material, an improvement which is supported by operational experience especially for corrosion protection. GNF introduced the so called Process 8 (P8) fuel cladding in 2003, which combined and enhanced the best features of previous successful P5 and P7 cladding designs. P9 was introduced in 2007 and became the standard cladding design for all customers in mid-2008. P9 is identical to P8 with the addition of a tighter iron specification biased toward the high end of American Society for Testing and Materials (ASTM) limits in the Zircaloy-2 portion of the cladding to provide enhanced corrosion margin. Currently P9 cladding utilizes an optimum nodular corrosion resistant microstructure and has achieved an experience base of well over several million rods with no corrosion or shadow corrosion related failures.
GNF also introduced an cladding inner liner of pure zirconium for PCI protection, which serves as a buffer between the Zircaloy-2 and the swelling of the fuel pellet.
Further discussions about the reference design are provided in Section 2.2.2.7, 2.2.2.8 and 2.3.9 of [Ref-11-5].

Fuel Assembly Configuration

Fuel design has undergone numerous evolutions, which initially employed 7x7 fuel bundle configuration and now employs 10x10 bundle configuration. This change allowed extension of thermal margins and mitigation of PCI due a to a reduced Maximum Linear Heat Generation rate, having led to substantial increase in performance and reliability.
Further discussions about the reference design are provided in [Ref-11-5].

Debris Filter

Debris fretting occurs when various types of debris in the coolant pass through the lower tie plate (LTP) and cause through wall fretting of the cladding.
In 1990 GNF offered its first intra-bundle debris protection by introducing an LTP that has an entry hole that is one third the size used in the previous LTP design. This reduced the size of debris that could enter the bundle. In 1996, GNF introduced a debris filter LTP that reduced the size of the debris that could enter the bundle by another factor of three. The next iteration of the design was to include a debris shield. The debris shield further reduced the size of the debris that could enter the

bundle with no pressure drop penalty. GNF then developed a next generation "Defender^{TM}" filter that further reduced the size of debris, but specifically targeted wires or wire-like debris that have commonly been associated with cladding perforations.

A feed water strainer is a further prospective countermeasure to reduce the occurrence of debris-induced fuel failures. This will be evaluated at a later stage by making use of operational experience gathered in the interim period.

Further discussions about the reference design are provided in Section 4 of [Ref-11-5].

Channel Box

The current channel material selection is Zircaloy-2. While the corrosion performance of Zircaloy-2 in a BWR environment is excellent, it is susceptible to channel bow especially at high exposure. In the near future, a channel bow resistant material called NSF will be available.

Further discussions about the reference design are provided in Section 3.5 of [Ref-11-5].

The strategy for mitigating the effects of channel bow has also evolved over time. A method was developed to explicitly account for the effects of channel bow on the Critical Power Ratio (CPR). Then for shadow corrosion-induced bow a more detailed methodology for minimizing the susceptibility to channel distortion (both bow and bulge) was implemented in the design process. The objective of this strategy has been to ensure that the channels did not significantly interfere with control rod motion, thus ensuring the operability of the control rod drives.

Further discussions about the reference design are provided in Section 4.3.9.1.2 of [Ref-11-7].

Soft Duty Rules

For modern (barrier) fuel under normal operating conditions, the primary protection against fuel failure by PCI is operation within the allowable steady-state operating envelope. With this protection only a small number of PCI failures have occurred. Although these failures were rare, they strongly correlated with certain specific types of operation; nearly all of the failure events had power histories that were in the tail of the distribution of possible power histories. The soft duty rules were introduced to address these power histories, including exposure dependent threshold power levels, with controlled ramp rates for power increases above a threshold or envelope.

Further discussions about the reference design are provided in Section 2.3.9 of [Ref-11-5].

Management of Damaged Fuel

The basic approach following failed fuel detection is a continued operation of the reactor at power by performing Power Suppression Testing and local power suppression. The feasible risk reduction options following failed fuel detection are considered to be:

- Operate the reactor within a lower primary coolant activity level limit, or
- Perform immediate reactor shutdown and removal of failed fuel.

The study concluded that the earlier shutdown of the reactor and the subsequent outage necessary for the removal of the failed fuel assembly would result in an increase of dose to the workforce, introduce nuclear safety risks related to the shutdown, start-up and outage operations, and a prolonged lost generation all of which translated in to a significant cost dis-benefit.
Further discussions about the reference design are provided in Section 5 of [Ref-11-5].

Control Rods

The boron carbide control rod design employed for the ABWR is a conventional design utilising a stainless steel sheath. The hafnium control rod design was introduced later with hafnium replacing the boron carbide powder as neutron absorbing material. An advantage of hafnium control rods is that they have a longer operational life than the boron carbide control rods since most of the isotopes produced by the neutron capture reactions still have the capability to capture neutrons. For the UK ABWR hafnium control rods will be used in the control cells, whereas boron carbide control rods will be used in the shutdown cells. The purpose of the control cells is to control reactivity during power operations and to shape the radial power profile of the reactor core in order to achieve an even burn-up of the loaded fuel. The other control rods are used for reactor shutdown operations.
Further discussions about the reference design are provided in Section 11.5.1.2.

Core Configuration and Loading Pattern

The UK ABWR has adopted an N-lattice geometry that provides a small increase in the intra-assembly bypass gap width relative to BWR/5-6 designs. The N-lattice benefits include an increased cold shutdown margin, moderate void reactivity coefficients, and increased margin to channel/control rod interference.
In UK ABWR core the fresh fuel assemblies are distributed throughout the centre region, alongside assemblies that have already been partially burned. The lower reactivity, higher burn-up assemblies are placed in the outer peripheral region to minimise neutron leakage.
Further discussions about the reference design are provided in Section 11.5.4.

Reactivity Control with Core Flow

Reactivity within the UK ABWR can be controlled by changing the core flow. Increasing the flow will increase the volume of moderator in the reactor consequently increasing reactivity. Conversely reducing the flow will decrease the volume of moderator in the reactor and therefore reduce reactivity. This core flow adjustment capability can be used to enhance a “spectral shift” operating strategy. In comparison, PWRs reactivity is controlled by the addition of boron to the reactor coolant, where the nuclear reactions involving boron and associated chemical species result in the generation of tritium. By using the recirculation flow control, the ABWR reduces significantly the generation of tritium.

Further discussions about the reference design are provided in Section 11.5.4.

Instabilities Countermeasures

The ABWR has the following advantages compared to the previous BWR plant types

- Fuel lattice pitch wider than previous BWR, which provides less negative void coefficient.
- A core inlet orifice diameter smaller than previous BWR, which results in a larger single phase pressure drop.
- The inner width of the fuel channel larger than previous BWR, which contributes to a lower core pressure drop.
- A larger number of low ΔP separators, which reduces the two phase pressure drop of the recirculation system.

With the characteristics described above and the mitigation system of RPS SCRAM “APRM simulated thermal power high (TPM high)” and Selected Control Rod Run-In (SCRRI), the ABWR has sufficient suppression capability against power oscillation.

Further discussions about the reference design are provided in Section 4.4.3 of [Ref-11-7].

Core Monitoring System

The regular C&I systems of the BWR have been complemented by the addition of a computerised core monitoring system that is capable of undertaking core-wide calculations of power distribution and thermal margins. This capability has resulted in a design that has detailed, near real-time monitoring of core parameters and provides to the operator additional information to support the plant operation within the thermal margins and other applicable limits.

Based on the extensive good operating experience and the ALARP evaluations of specific aspects described above, it is concluded that the fuel and core in the UK ABWR currently satisfies the ALARP principle.

11.9 Conclusions

The description of the UK ABWR reactor core given in this chapter is based on design specifications and the established design basis. Limits and other design features established are supported by evaluations reported in the supporting references to the Chapter.

The main Structures, Systems and Components (SSCs) of the reactor core are described in detail and these are identified as:

- Fuel assemblies (fuel bundle and fuel channel),
- Boron carbide and hafnium control rods, and
- Core monitoring system.

Overview descriptions and design evaluations are also presented for Control Rod Drive and Standby Liquid Control systems.

The Chapter uses the initial and equilibrium cores developed for the UK ABWR as the basis for all core physics related evaluations. Computational methods are discussed, as well as the parameters relevant to ensuring safe and reliable operation.

This Chapter lists the Safety Functional Claims (SFCs) that are made on fuel and core components and core loading patterns, to maintain the High Level Safety Functions (HLSFs) during normal operation and design basis fault conditions.

A summary is provided of the justification that the risks associated with operation of the UK ABWR core are acceptable (in terms of radiation dose consequences) and have been reduced to levels that are As Low As Reasonably Practicable (ALARP). A summary is also given of the Assumptions, Limits and Conditions for Operation for the fuel and core.

11.10 References

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[Ref-11-18] Hitachi-GE Nuclear Energy, Ltd., "Description of Thermal Analysis Methodology", GA91-9201-0003-00008 (UE-GD-0094) Rev. 1, June 2015.

[Ref-11-19] Hitachi-GE Nuclear Energy, Ltd., "Description of Stability Analysis Code", GA91-9201-0003-00007 (UE-GD-0095) Rev. 1, June 2015.

[Ref-11-20] Hitachi-GE Nuclear Energy, Ltd., "Topic Report on Control Rod Design", GA91-9201-0001-00133 (KCE-GD-2014) Rev. 0, May 2015.

[Ref-11-21] Hitachi-GE Nuclear Energy, Ltd., "Description of Core Monitoring System", GA91-9201-0003-00608 (UE-GD-0345) Rev. 0, June 2015.

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Appendix A: Safety Functional Claims Table

	Top Claim for Fuel Assembly													
	Fundamental Safety Function (FSF) Table 5.6-1: High Level Safety Functions in UK ABWR		High Level Safety Function (HLSF) Table 5.6-1: High Level Safety Functions in UK ABWR				Fault Schedule (Bounding Fault) Topic Report on Fault Assessment Table.4.2-1 Fault Schedule [Ref-11-33]		Safety Functional Claim (SFC)					
			No.	HLSF	Cat.	Class	No.	Contents	State	Claim ID	Claim Contents	Cat.	Class	
1	4	Confinement/Containment or radioactive materials	4-10	Functions to prevent the dispersion of fission products into reactor coolant, spent fuel pool and canister	C	3	-	Conformity to the claims is evaluated in the fuel mechanical design (see subsection 11.6.1.1 and [Ref-11-5]). No corresponding event	Normal Conditions and Fault Conditions	NF SC 1 (FA SFC 4-10.1)	Cladding creep out rate due to fuel rod internal pressure is less than or equal to the fuel pellet irradiation swelling rate in normal operation and all frequent design basis faults so as to preclude creep deformation (for cladding lift-off).	C	3	
2	4	Confinement/Containment or radioactive materials	4-10	Functions to prevent the dispersion of fission products into reactor coolant, spent fuel pool and canister	C	3	1.1, 1.2, 1.3, 1.4, 1.5, 1.6, 1.7, 1.8, 1.9, 2.1, 2.2, 2.3, 3.1, 4.2, 4.4, 4.5, 4.6, 5.1, 5.2, 5.3, 6.1, 7.2, 1.1.1/12.1, 1.1.2/12.2, 1.1.3/12.3, 11.1, 13.2, 17.1, 17.2, 17.3, 17.4, 18.1, 18.2	Conformity to the claims is evaluated with the design basis analysis (see [Ref-11-25]). Event corresponding to IDs are described in Table 24.4-1.	Normal Conditions and Fault Conditions	NF SC 2 (FA SFC 4-10.2)	The fuel centerline temperature is less than the melting temperature in normal operation and all frequent design basis faults so as to preclude subsequent potential cladding damage.	C	3	

	Top Claim for Fuel Assembly													
	Fundamental Safety Function (FSF) Table 5.6-1: High Level Safety Functions in UK ABWR		High Level Safety Function (HLSF) Table 5.6-1: High Level Safety Functions in UK ABWR				Fault Schedule (Bounding Fault) Topic Report on Fault Assessment Table.4.2-1 Fault Schedule [Ref-11-33]		Safety Functional Claim (SFC)					
			No.	HLSF	Cat.	Class	No.	Contents	State	Claim ID	Claim Contents	Cat.	Class
3	4	Confinement/Containment or radioactive materials	4-10	Functions to prevent the dispersion of fission products into reactor coolant, spent fuel pool and canister	C	3	1.1, 1.2, 1.3, 1.4, 1.5, 1.6, 1.7, 1.8, 1.9, 2.1, 2.2, 2.3, 3.1, 4.2, 4.4, 4.5, 4.6, 5.1, 5.2, 5.3, 6.1, 7.2, 1.1.1/12.1, 1.1.2/12.2, 1.1.3/12.3, 11.1, 13.2, 17.1, 17.2, 17.3, 17.4, 18.1, 18.2	Conformity to the claims is evaluated with the design basis analysis (see [Ref-11-25]). Event corresponding to IDs are described in Table 24.4-1.	Normal Conditions and Fault Conditions	NF SC 3 (FA SFC 4-10.3)	The cladding circumferential strain due to pellet-clad mechanical interaction is less than the design limit at normal operation and all frequent design basis faults so as to preclude mechanical failure due to cladding strain.	C	3
4	4	Confinement/Containment or radioactive materials	4-10	Functions to prevent the dispersion of fission products into reactor coolant, spent fuel pool and canister	C	3	-	Conformity to the claims is evaluated in the fuel mechanical design (see subsection 11.6.1.1 and [Ref-11-5]). No corresponding event	Normal Conditions and Fault Conditions	NF SC 4 (FA SFC 4-10.4)	The cumulative duty from cladding strain due to cyclic loadings in normal operation and all frequent design basis faults is less than the cladding fatigue capability.	C	3
5	4	Confinement/Containment or radioactive materials	4-10	Functions to prevent the dispersion of fission products into reactor coolant, spent fuel pool and canister	C	3	-	Conformity to the claims is evaluated in the fuel mechanical design (see subsection 11.6.1.1 and [Ref-11-5]). No corresponding event	Normal Conditions and Fault Conditions	NF SC 5 (FA SFC 4-10.5)	The fuel cladding damage from cladding collapse into a fuel column axial gap is precluded in normal operation and all frequent design basis faults.	C	3

	Top Claim for Fuel Assembly												
	Fundamental Safety Function (FSF) Table 5.6-1: High Level Safety Functions in UK ABWR		High Level Safety Function (HLSF) Table 5.6-1: High Level Safety Functions in UK ABWR				Fault Schedule (Bounding Fault) Topic Report on Fault Assessment Table.4.2-1 Fault Schedule [Ref-11-33]		Safety Functional Claim (SFC)				
			No.	HLSF	Cat.	Class	No.	Contents	State	Claim ID	Claim Contents	Cat.	Class
6	4	Confinement/Containment or radioactive materials	4-10	Functions to prevent the dispersion of fission products into reactor coolant, spent fuel pool and canister	C	3	-	Conformity to the claims is evaluated in the fuel mechanical design (see subsection 11.6.1.1 and [Ref-11-5]). No corresponding event	Normal Conditions and Fault Conditions	NF SC 6 (FA SFC 4-10.6)	The cladding stresses or strains are less than material failure limits at normal operation and all frequent design basis faults so as to preclude mechanical failure due to cladding stresses or strains.	C	3
7	4	Confinement/Containment or radioactive materials	4-10	Functions to prevent the dispersion of fission products into reactor coolant, spent fuel pool and canister	C	3	-	Conformity to the claims is evaluated in the fuel mechanical design (see subsection 11.6.1.1 and [Ref-11-5]). No corresponding event	Normal Conditions and Fault Conditions	NF SC 7 (FA SFC 4-10.7)	The fuel cladding damage from the localised reduction in cladding ductility due to hydriding is precluded in normal operation and all frequent design basis faults.	C	3
8	4	Confinement/Containment or radioactive materials	4-10	Functions to prevent the dispersion of fission products into reactor coolant, spent fuel pool and canister	C	3	-	Conformity to the claims is evaluated in the fuel mechanical design (see subsection 11.6.1.1 and [Ref-11-5]). No corresponding event	Normal Conditions and Fault Conditions	NF SC 8 (FA SFC 4-10.8)	The fuel cladding damage from cladding oxidation is precluded in normal operation and all frequent design basis faults.	C	3
9	4	Confinement/Containment or radioactive materials	4-10	Functions to prevent the dispersion of fission products into reactor coolant, spent fuel pool and canister	C	3	-	Conformity to the claims is evaluated in the fuel mechanical design (see subsection 11.6.1.4 and [Ref-11-5]). No corresponding event	Normal Conditions and Fault Conditions	NF SC 9 (FA SFC 4-10.9)	The fuel cladding damage from fretting wear is precluded in normal operation.	C	3

	Top Claim for Fuel Assembly												
	Fundamental Safety Function (FSF) Table 5.6-1: High Level Safety Functions in UK ABWR		High Level Safety Function (HLSF) Table 5.6-1: High Level Safety Functions in UK ABWR				Fault Schedule (Bounding Fault) Topic Report on Fault Assessment Table.4.2-1 Fault Schedule [Ref-11-33]		Safety Functional Claim (SFC)				
			No.	HLSF	Cat.	Class	No.	Contents	State	Claim ID	Claim Contents	Cat.	Class
10	4	Confinement/Containment or radioactive materials	4-10	Functions to prevent the dispersion of fission products into reactor coolant, spent fuel pool and canister	C	3	-	Conformity to the claims is evaluated in the fuel mechanical design (see subsection 11.6.1.3 and [Ref-11-5]). No corresponding event	Normal Conditions and Fault Conditions	NF SC 10 (FA SFC 4-10.10)	The fuel cladding damage from PCI/SCC is precluded in normal operation and any frequent design basis faults.	C	3
11	2	Fuel Cooling	2-1	Functions to cool reactor core	A	1	7.1, 8.1, 8.2, 9.1, 9.2, 9.3, 10.1, 10.2, 10.3, 1.1.1, 1.4.2, 1.5.1, 2.1.1, 2.2.1, 2.3.1, 5.1.1, 5.1.3, 5.2.1, 11.2, 11.3, 11.4, 11.5, 11.8.1, 11.9, 11.10.1, 11.11.1, 11.12.1, 17.5, 17.6, 18.3	Conformity to the claims is evaluated with the design basis analysis (see [Ref-11-25]). Event corresponding to IDs are described in Table 24.4-1.	Fault Conditions	NF SC 11 (FA SFC 2-1.1)	In all infrequent design basis faults, peak cladding temperature is less than 1,200 °C .	A	1
12	2	Fuel Cooling	2-1	Functions to cool reactor core	A	1	7.1, 8.1, 8.2, 9.1, 9.2, 9.3, 10.1, 10.2, 10.3, 1.1.1, 1.4.2, 1.5.1, 2.1.1, 2.2.1, 2.3.1, 5.1.1, 5.1.3, 5.2.1, 11.2, 11.3,	Conformity to the claims is evaluated with the design basis analysis (see [Ref-11-25]). Event corresponding to IDs are described in Table 24.4-1.	Fault Conditions	NF SC 12 (FA SFC 2-1.2)	In all infrequent design basis faults, peak cladding oxidation is less than 15 percent equivalent cladding reacted (ECR).	A	1

	Top Claim for Fuel Assembly													
	Fundamental Safety Function (FSF) Table 5.6-1: High Level Safety Functions in UK ABWR		High Level Safety Function (HLSF) Table 5.6-1: High Level Safety Functions in UK ABWR				Fault Schedule (Bounding Fault) Topic Report on Fault Assessment Table.4.2-1 Fault Schedule [Ref-11-33]		Safety Functional Claim (SFC)					
			No.	HLSF	Cat.	Class	No.	Contents	State	Claim ID	Claim Contents	Cat.	Class	
							11.4, 11.5, 11.8.1, 11.9, 11.10.1, 11.11.1, 11.12.1, 17.5, 17.6, 18.3							
13	2	Fuel Cooling	2-1	Functions to cool reactor core	A	1	4.1, 4.3, 13.1, 13.2	Conformity to the claims is evaluated with the design basis analysis (see [Ref-11-25]). Event corresponding to IDs are described in Table 24.4-1.	Fault Conditions	NF SC 13 (FA SFC 2-1.3)	Fuel enthalpy meets the prescribed design limit for all frequent and infrequent design basis faults.	A	1	
14	1	Control of Reactivity	1-2	Functions to maintain core geometry	A	1	-	Conformity to the claims is evaluated in the FA mechanical design (see subsection 11.6.1.4 and [Ref-11-5]). No corresponding event	Normal Conditions and Fault Conditions	NF SC 14 (FA SFC 1-2.1) (FA SFC 2-1.4) (FA SFC 5-6.1)	The FA structural components (fuel rods, upper tie plate, lower tie plate, spacers, water rods and fuel channel) do not fail due to stresses less than or equal to the FA component mechanical capability.	A	1	
	2	Fuel Cooling	2-1	Functions to cool reactor core	A	1								
	5	Others	5-6	Functions to handle fuel and heavy equipment safely	A	1								
15	1	Control of Reactivity	1-2	Functions to maintain core geometry	A	1	-	Conformity to the claims is evaluated in the FA mechanical design (see subsection 11.6.1.4 and [Ref-11-5]). No corresponding event	Normal Conditions and Fault Conditions	NF SC 15 (FA SFC 1-2.2) (FA SFC 2-1.5)	The cumulative duty on FA structural components from cyclic loadings is less than the material fatigue capability.	A	1	
	2	Fuel Cooling	2-1	Functions to cool reactor core	A	1								

	Top Claim for Fuel Assembly														
	Fundamental Safety Function (FSF) Table 5.6-1: High Level Safety Functions in UK ABWR		High Level Safety Function (HLSF) Table 5.6-1: High Level Safety Functions in UK ABWR				Fault Schedule (Bounding Fault) Topic Report on Fault Assessment Table.4.2-1 Fault Schedule [Ref-11-33]		Safety Functional Claim (SFC)					
			No.	HLSF	Cat.	Class	No.	Contents	State	Claim ID	Claim Contents	Cat.	Class
16	1	Control of Reactivity	1-2	Functions to maintain core geometry	A	1	-	Conformity to the claims is evaluated in the fuel mechanical design (see subsection 11.6.1.4 and [Ref-11-5]). No corresponding event	Normal Conditions and Fault Conditions	NF SC 16 (FA SFC 1-2.3) (FA SFC 2-1.6)	The fuel cladding damage from fretting wear is precluded by the FA structural components.	A	1
	2	Fuel Cooling	2-1	Functions to cool reactor core	A	1							
17	4	Confinement/Containment of radioactive materials	4-10	Functions to prevent the dispersion of fission products into reactor coolant, spent fuel pool and canister	C	3	Dose evaluation	Conformity to the claims is evaluated with the design basis analysis for the infrequent faults which may cause fuel leak and in the fuel mechanical design (see [Ref-11-25]).	Normal Conditions and Fault Conditions	NF SC 17 (FA SFC 4-10.11)	Fuel assembly and its components are designed so that the effective dose received by any person is less than the prescribed limit, and within the functions of relevant plant components.	C	3
18	4	Confinement/Containment or radioactive materials	4-10	Functions to prevent the dispersion of fission products into reactor coolant, spent fuel pool and canister	C	3	-	Conformity to the claims is evaluated in the fuel mechanical design (see section 19.8, chapter 32). No corresponding event	Normal Conditions	NF SC 18 (FA SFC 4-10.12)	When fuel assemblies are stored in the spent fuel storage pool which is surrounded by water, fuel cladding damage from corrosion is precluded.	C	3
19	4	Confinement/Containment or radioactive materials	4-10	Functions to prevent the dispersion of fission products into reactor coolant, spent fuel pool and canister	C	3	-	Conformity to the claims is evaluated in the fuel mechanical design (see [Ref-11-22]). No corresponding event	Normal Conditions and Fault Conditions	NF SC 19 (FA SFC 4-10.13)	Under normal operation and any frequent design basis faults the inert environment and maximum temperature are controlled and kept less than the prescribed limit so as to preclude cladding failure.	C	3

	Top Claim for Fuel Assembly													
	Fundamental Safety Function (FSF) Table 5.6-1: High Level Safety Functions in UK ABWR		High Level Safety Function (HLSF) Table 5.6-1: High Level Safety Functions in UK ABWR				Fault Schedule (Bounding Fault) Topic Report on Fault Assessment Table.4.2-1 Fault Schedule [Ref-11-33]		Safety Functional Claim (SFC)					
			No.	HLSF	Cat.	Class	No.	Contents	State	Claim ID	Claim Contents	Cat.	Class
20	2	Fuel Cooling	2-1	Functions to cool reactor core	A	1	1.2, 1.5, 1.6, 1.7, 1.8, 1.9, 2.1, 2.2, 2.3, 3.1, 4.2, 4.4, 4.5, 4.6, 5.1, 5.2, 5.3, 6.1, 7.2, 11.1, 1.1.1/12.1, 1.1.2/12.2, 1.1.3/12.3, 13.2, 17.1, 17.2, 17.3, 17.4, 18.1, 18.2	Conformity to the claim is evaluated with the design basis analysis (see [Ref-11-25]). Event corresponding to IDs are described in Table 24.4-1.	Normal Conditions and Fault Conditions	NF SC 33 (THD SFC 2-1.1)	At normal operation and frequent design basis faults, MCPR is equal to or has margin to the safety limit so as to preclude thermal failure with overheating.	A	1
21	2	Fuel Cooling	2-1	Functions to cool reactor core	A	1	1.1, 1.3, 1.4, 4.2	Conformity to the claim is evaluated with the design basis analysis (see [Ref-11-25]). Event corresponding to IDs are described in Table 24.4-1.	Normal Conditions and Fault Conditions	NF SC 34 (THD SFC 2-1.2)	At normal operation and frequent design basis faults, when MCPR becomes less than the safety limit, cladding temperature is less than 800 °C such that ballooning rupture failure is precluded.	A	1

	Top Claim for Control Rods													
	Fundamental Safety Function (FSF) Table 5.6-1: High Level Safety Functions in UK ABWR		High Level Safety Function (HLSF) Table 5.6-1: High Level Safety Functions in UK ABWR				Fault Schedule (Bounding Fault) Topic Report on Fault Assessment Table.4.2-1 Fault Schedule [Ref-11-33]		Safety Functional Claim (SFC)					
			No	HLSF	Cat.	Class	No.	Contents	State	Claim ID	Claim Contents	Cat	Class
1	1	Control of Reactivity	1-2	Functions to maintain core geometry	A	1	-	Conformity to the claim is evaluated in the CR mechanical design (see subsection 11.6.2 and [Ref-11-20]). No corresponding event	Normal Conditions and Fault Conditions	NF SC 20 (CR SFC 1-2.1)	The CR stresses are less than the allowable limit.	A	1
2	1	Control of Reactivity	1-2	Functions to maintain core geometry	A	1	-	Conformity to the claim is evaluated in the CR mechanical design (see subsection 11.6.2 and [Ref-11-20]) No corresponding event	Normal Conditions and Fault Conditions	NF SC 21 (CR SFC 1-2.2)	The material of the CR shall be shown to be compatible with the reactor environment.	A	1
3	1	Control of Reactivity	1-4	Functions to maintain sub-criticality	A A	1 2	-	Conformity to the claim is evaluated in the nuclear design (see subsection 11.6.5.3, [Ref-11-6] and [Ref-11-7]). No corresponding event	Normal Conditions and Fault Conditions	NF SC 27 (ND SFC 1-4.1)	The reactor shutdown system is able to bring the core to a subcritical state, with adequate margin that bounds inherent biases and uncertainties associated with calculational methods, from a hot stand-by or power operation condition.	A A	1 2

	Top Claim for Control Rods												
	Fundamental Safety Function (FSF) Table 5.6-1: High Level Safety Functions in UK ABWR		High Level Safety Function (HLSF) Table 5.6-1: High Level Safety Functions in UK ABWR				Fault Schedule (Bounding Fault) Topic Report on Fault Assessment Table.4.2-1 Fault Schedule [Ref-11-33]		Safety Functional Claim (SFC)				
			No	HLSF	Cat.	Class	No.	Contents	State	Claim ID	Claim Contents	Cat	Class
4	1	Control of Reactivity	1-4	Functions to maintain sub-criticality	A	1	-	Conformity to the claim is evaluated in the nuclear design (see subsection 11.6.5.3, [Ref-11-6] [Ref-11-7]). No corresponding event	Normal Conditions and Fault Conditions	NF SC 28 (ND SFC 1-4.2) (ND SFC 1-5.1)	The reactor shutdown system consists of two independent systems: CR with the FMCRD system that provides a hydraulic fast scram capability and a redundant electric run-in insertion method of CRs (elements that contribute to the excellent safety characteristics of the ABWR), and the Standby Liquid Control System (SLC) which can maintain a subcritical state at hot stand-by assuming a zero xenon condition (i.e., no shutdown credit is taken for xenon).	A	1
					A	2							
	1	Control of Reactivity	1-5	Function of alternative reactivity control	A	2						A	2
					C	3						C	3
5	1	Control of Reactivity	1-3	Emergency shutdown of the reactor	A	1	Reactor Faults initiating a scram	Conformity to the claim is evaluated with the design basis analysis ([Ref-11-25]).	Normal Conditions and Fault Conditions	NF SC 30 (ND SFC 1-3.1) (ND SFC 1-4.4)	For frequent design basis faults, the CR and FMCRD system is able to bring the core to a subcritical state at hot conditions and maintain this state at hot conditions, preventing the allowable fuel design limit from being exceeded.	A	1
	1	Control of Reactivity	1-4	Functions to maintain sub-criticality	A	1						A	2
					A	2							

	Top Claim for Control Rods												
	Fundamental Safety Function (FSF) Table 5.6-1: High Level Safety Functions in UK ABWR		High Level Safety Function (HLSF) Table 5.6-1: High Level Safety Functions in UK ABWR				Fault Schedule (Bounding Fault) Topic Report on Fault Assessment Table.4.2-1 Fault Schedule [Ref-11-33]		Safety Functional Claim (SFC)				
			No	HLSF	Cat.	Class	No.	Contents	State	Claim ID	Claim Contents	Cat	Class
6	1	Control of Reactivity	1-3	Emergency shutdown of the reactor	A	1	-	Conformity to the claim is evaluated in the nuclear design (see subsection 11.6.5.3, [Ref-11-6] and [Ref-11-7]). No corresponding event	Normal Conditions and Fault Conditions	NF SC 31 (ND SFC 1-3.2) (ND SFC 1-4.5)	At least one of the independent reactor shutdown systems included is able to bring the core to a subcritical state at cold conditions and maintain this state at cold conditions.	A	1
	1	Control of Reactivity	1-4	Functions to maintain sub-criticality	A	1						A	2
					A	2							
7	1	Control of Reactivity	1-1	Functions to prevent excessive reactivity insertion	A	1	4.1, 4.3, 13.1, 13.2	Conformity to the claim is evaluated with the design basis analysis (see [Ref-11-25]). Event corresponding to IDs are described in Table 24.4-1.	Normal Conditions and Fault Conditions	NF SC 32 (ND SFC 1-1.1)	The maximum reactivity worth and the reactivity insertion rate of the CR, which are less than the design limits used in the safety analyses of anticipated reactivity insertion events, do not damage the reactor coolant pressure boundary and do not cause damage to the core, core support structure and pressure vessel internal structure that may impair cooling of the fuel.	A	1

	Top Claim for Nuclear Design													
	Fundamental Safety Function (FSF) Table 5.6-1: High Level Safety Functions in UK ABWR		High Level Safety Function (HLSF) Table 5.6-1: High Level Safety Functions in UK ABWR				Fault Schedule (Bounding Fault) Topic Report on Fault Assessment Table.4.2-1 Fault Schedule [Ref-11-33]		Safety Functional Claim (SFC)					
			No.	HLSF	Cat.	Class	No.	Contents	State	Claim ID	Claim Contents	Cat.	Class
1	1	Control of Reactivity	1-8	Functions to suppress reactor power increase with other system	C	3	Reactor Faults	Conformity to the claim is evaluated with the design basis analysis (see [Ref-11-25]).	Normal Conditions and Fault Conditions	NF SC 22 (ND SFC 1-8.1)	The power distribution is controlled so that the operational thermal limits are not exceeded at operation and all frequent design basis faults.	C	3
2	1	Control of Reactivity	1-8	Functions to suppress reactor power increase with other system	C	3	-	Conformity to the claim is evaluated in the nuclear design (see subsection 11.6.5.2, [Ref-11-6] and [Ref-11-7]). No corresponding event	Normal Conditions and Fault Conditions	NF SC 23 (ND SFC 1-8.2)	The overall moderator void coefficient is negative within a range that pressurization transients do not threaten fuel integrity	C	3
3	1	Control of Reactivity	1-8	Functions to suppress reactor power increase with other system	C	3	-	Conformity to the claim is evaluated in the nuclear design (see subsection 11.6.5.2, [Ref-11-6] and [Ref-11-7]). No corresponding event	Normal Conditions and Fault Conditions	NF SC 24 (ND SFC 1-8.3)	The Doppler coefficient is negative and has sufficient reactivity feedback characteristics to terminate infrequent design basis faults (i.e., reactivity insertion accident).	C	3
4	1	Control of Reactivity	1-8	Functions to suppress reactor power increase with other system	C	3	-	Conformity to the claim is evaluated in the nuclear design (see subsection 11.6.5.2, [Ref-11-6] and [Ref-11-7]). No corresponding event	Normal Conditions and Fault Conditions	NF SC 25 (ND SFC 1-8.4)	The moderator temperature coefficient is not a value that may pose a safety concern in the current design range and it has no specific requirement.	C	3
5	1	Control of Reactivity	1-8	Functions to suppress reactor power increase with other system	C	3	-	Conformity to the claim is evaluated in the nuclear design (see subsection 11.6.5.2, [Ref-11-6] and [Ref-11-7]).	Normal Conditions and Fault Conditions	NF SC 26 (ND SFC 1-8.5)	The power reactivity coefficient is negative and its feedback is large enough to reduce the spatial oscillation of xenon.	C	3

	Top Claim for Nuclear Design												
	Fundamental Safety Function (FSF) Table 5.6-1: High Level Safety Functions in UK ABWR		High Level Safety Function (HLSF) Table 5.6-1: High Level Safety Functions in UK ABWR				Fault Schedule (Bounding Fault) Topic Report on Fault Assessment Table.4.2-1 Fault Schedule [Ref-11-33]		Safety Functional Claim (SFC)				
			No.	HLSF	Cat.	Class	No.	Contents	State	Claim ID	Claim Contents	Cat.	Class
								No corresponding event					
6	1	Control of Reactivity	1-4	Functions to maintain sub-criticality	A	1	-	Conformity to the claim is evaluated in the nuclear design (see subsection 11.6.5.3, [Ref-11-6] and [Ref-11-7]).	Normal Conditions and Fault Conditions	NF SC 27 (ND SFC 1-4.1)	The reactor shutdown system is able to bring the core to a subcritical state, with adequate margin that bounds inherent biases and uncertainties associated with calculational methods, from a hot stand-by or power operation condition.	A	1
					A	2		No corresponding event				A	2
7	1	Control of Reactivity	1-4	Functions to maintain sub-criticality	A	1	-	Conformity to the claim is evaluated in the nuclear design (see subsection 11.6.5.3, [Ref-11-6] and [Ref-11-7]).	Normal Conditions and Fault Conditions	NF SC 28 (ND SFC 1-4.2) (ND SFC 1-5.1)	The reactor shutdown system consists of two independent systems: CR with the FMCRD system that provides a hydraulic fast scram capability and a redundant electric run-in insertion method of CRs (elements that contribute to the excellent safety characteristics of the ABWR), and the Standby Liquid Control System (SLC) which can maintain a subcritical state at hot stand-by assuming a zero xenon condition (i.e., no	A	1
					A	2		No corresponding event					
	1	Control of Reactivity	1-5	Function of alternative reactivity control	A	2						A	2
					C	3						C	3

	Top Claim for Nuclear Design												
	Fundamental Safety Function (FSF) Table 5.6-1: High Level Safety Functions in UK ABWR		High Level Safety Function (HLSF) Table 5.6-1: High Level Safety Functions in UK ABWR				Fault Schedule (Bounding Fault) Topic Report on Fault Assessment Table.4.2-1 Fault Schedule [Ref-11-33]		Safety Functional Claim (SFC)				
			No.	HLSF	Cat.	Class	No.	Contents	State	Claim ID	Claim Contents	Cat.	Class
											shutdown credit is taken for xenon).		
8	1	Control of Reactivity	1-4	Functions to maintain sub-criticality	A	1	-	Conformity to the claim is evaluated in the nuclear design (see subsection 11.6.5.3, [Ref-11-6] and [Ref-11-7]).	Normal Conditions and Fault Conditions	NF SC 29 (ND SFC 1-4.3)	The CR and FMCRD system of the reactor shutdown system is able to bring the core to a subcritical state at hot or cold conditions when one or more CRs with the largest reactivity worth are completely withdrawn from the core and cannot be inserted.	A	1
					A	2		No corresponding event				A	2
9	1	Control of Reactivity	1-3	Emergency shutdown of the reactor	A	1	Reactor Faults initiating a scram	Conformity to the claim is evaluated with the design basis analysis [Ref-11-25].	Normal Conditions and Fault Conditions	NF SC 30 (ND SFC 1-3.1) (ND SFC 1-4.4)	For frequent design basis faults, the CR and FMCRD system is able to bring the core to a subcritical state at hot conditions and maintain this state at hot conditions, preventing the allowable fuel design limit from being exceeded.	A	1

	Top Claim for Nuclear Design												
	Fundamental Safety Function (FSF) Table 5.6-1: High Level Safety Functions in UK ABWR		High Level Safety Function (HLSF) Table 5.6-1: High Level Safety Functions in UK ABWR				Fault Schedule (Bounding Fault) Topic Report on Fault Assessment Table.4.2-1 Fault Schedule [Ref-11-33]		Safety Functional Claim (SFC)				
			No.	HLSF	Cat.	Class	No.	Contents	State	Claim ID	Claim Contents	Cat.	Class
	1	Control of Reactivity	1-4	Functions to maintain sub-criticality	A A	1 2						A	2
10	1	Control of Reactivity	1-3	Emergency shutdown of the reactor	A	1	-	Conformity to the claim is evaluated in the nuclear design (see subsection 11.6.5.3, [Ref-11-6] and [Ref-11-7]). No corresponding event	Normal Conditions and Fault Conditions	NF SC 31 (ND SFC 1-3.2) (ND SFC 1-4.5)	At least one of the independent reactor shutdown systems included is able to bring the core to a subcritical state at cold conditions and maintain this state at cold conditions.	A A	1 2
	1	Control of Reactivity	1-4	Functions to maintain sub-criticality	A A	1 2							
11	1	Control of Reactivity	1-1	Functions to prevent excessive reactivity insertion	A	1	4.1, 4.3, 13.1, 13.2	Conformity to the claim is evaluated with the design basis analysis (see [Ref-11-25]). Event corresponding to IDs are described in Table 24.4-1.	Normal Conditions and Fault Conditions	NF SC 32 (ND SFC 1-1.1)	The maximum reactivity worth and the reactivity insertion rate of the CR, which are less than the design limits used in the safety analyses of anticipated reactivity insertion events, do not damage the reactor coolant pressure boundary and do not cause damage to the core, core support structure and pressure vessel internal structure that may impair cooling of the fuel.	A	1

	Top Claim for Nuclear Design													
	Fundamental Safety Function (FSF) Table 5.6-1: High Level Safety Functions in UK ABWR		High Level Safety Function (HLSF) Table 5.6-1: High Level Safety Functions in UK ABWR				Fault Schedule (Bounding Fault) Topic Report on Fault Assessment Table.4.2-1 Fault Schedule [Ref-11-33]		Safety Functional Claim (SFC)					
			No.	HLSF	Cat.	Class	No.	Contents	State	Claim ID	Claim Contents	Cat.	Class
12	1	Control of Reactivity	1-8	Functions to suppress reactor power increase with other system	C	3	-	Conformity to the claim is evaluated in the nuclear design (see subsection 11.6.6.4 and [Ref-11-6][Ref-11-7]). No corresponding event	Normal Conditions and Fault Conditions	NF SC 35 (THD SFC 1-8.1)	The calculated decay ratio for thermal hydraulic stability does not exceed the limiting criteria in the normal operating region. In addition, in case that power oscillation due to thermal hydraulic instability occurs, it is controlled and the reactor is led to a stable state.	C	3
13	1	Control of Reactivity	1-8	Functions to suppress reactor power increase with other system	C	3	-	Conformity to the claim is evaluated in the nuclear design (see subsection 11.6.6.4 and [Ref-11-6][Ref-11-7]). No corresponding event	Normal Conditions and Fault Conditions	NF SC 36 (THD SFC 1-8.2)	In case that power oscillation due to thermal hydraulic instability occurs, it is controlled and the reactor is led to a stable state.	C	3

	Top Claim for Standby Liquid Control System												
	Fundamental Safety Function (FSF) Table 5.6-1: High Level Safety Functions in UK ABWR		High Level Safety Function (HLSF) Table 5.6-1: High Level Safety Functions in UK ABWR				Fault Schedule (Bounding Fault) Topic Report on Fault Assessment Table.4.2-1 Fault Schedule [Ref-11-33]		Safety Functional Claim (SFC)				
			No.	HLSF	Cat.	Class	No.	Contents	State	Claim ID	Claim Contents	Cat.	Class
1	1	Control of Reactivity	1-4	Functions to maintain sub-criticality	A	1	-	Conformity to the claim is evaluated in the nuclear design (see subsection 11.6.5.3, [Ref-11-6] and [Ref-11-7]). No corresponding event	Normal Conditions and Fault Conditions	NF SC 27 (ND SFC 1-4.1)	The reactor shutdown system is able to bring the core to a subcritical state, with adequate margin that bounds inherent biases and uncertainties associated with calculational methods, from a hot stand-by or power operation condition.	A	1
					A	2						A	2
2	1	Control of Reactivity	1-4	Functions to maintain sub-criticality	A	1	-	Conformity to the claim is evaluated in the nuclear design (see subsection 11.6.5.3, [Ref-11-6] and [Ref-11-7]). No corresponding event	Normal Conditions and Fault Conditions	NF SC 28 (ND SFC 1-4.2) (ND SFC 1-5.1)	The reactor shutdown system consists of two independent systems: CR with the FMCRD system that provides a hydraulic fast scram capability and a redundant electric run-in insertion method of CRs (elements that contribute to the excellent safety characteristics of the ABWR), and the Standby Liquid Control System (SLC) which can maintain a subcritical state at hot stand-by assuming a zero xenon condition (i.e., no shutdown credit is taken for xenon).	A	1
					A	2						A	2
	1	Control of Reactivity	1-5	Function of alternative reactivity control	A	2							
					C	3						C	3

	Top Claim for Standby Liquid Control System														
	Fundamental Safety Function (FSF) Table 5.6-1: High Level Safety Functions in UK ABWR		High Level Safety Function (HLSF) Table 5.6-1: High Level Safety Functions in UK ABWR				Fault Schedule (Bounding Fault) Topic Report on Fault Assessment Table.4.2-1 Fault Schedule [Ref-11-33]		Safety Functional Claim (SFC)					
			No.	HLSF	Cat.	Class	No.	Contents	State	Claim ID	Claim Contents	Cat.	Class
3	1	Control of Reactivity	1-3	Emergency shutdown of the reactor	A	1	-	Conformity to the claim is evaluated in the nuclear design (see subsection 11.6.5.3,[Ref-11-6] and [Ref-11-7]). No corresponding event	Normal Conditions and Fault Conditions	NF SC 31 (ND SFC 1-3.2) (ND SFC 1-4.5)	At least one of the independent reactor shutdown systems included is able to bring the core to a subcritical state at cold conditions and maintain this state at cold conditions.	A	1
	1	Control of Reactivity	1-4	Functions to maintain sub-criticality	A	1						A	2
					A	2							

	Top Claim for Thermal Hydraulic Design													
	Fundamental Safety Function (FSF) Table 5.6-1: High Level Safety Functions in UK ABWR		High Level Safety Function (HLSF) Table 5.6-1: High Level Safety Functions in UK ABWR				Fault Schedule (Bounding Fault) Topic Report on Fault Assessment Table.4.2-1 Fault Schedule [Ref-11-33]		Safety Functional Claim (SFC)					
			No.	HLSF	Cat.	Class	No.	Contents	State	Claim ID	Claim Contents	Cat.	Class
1	2	Fuel Cooling	2-1	Functions to cool reactor core	A	1	1.2, 1.5, 1.6, 1.7, 1.8, 1.9, 2.1, 2.2, 2.3, 3.1, 4.2, 4.4, 4.5, 4.6, 5.1, 5.2, 5.3, 6.1, 7.2, 11.1, 1.1.1/12.1, 1.1.2/12.2, 1.1.3/12.3, 13.2, 17.1, 17.2, 17.3, 17.4, 18.1, 18.2	Conformity to the claim is evaluated with the design basis analysis (see [Ref-11-25]). Event corresponding to IDs are described in Table 24.4-1.	Normal Conditions and Fault Conditions	NF SC 33 (THD SFC 2-1.1)	At normal operation and frequent design basis faults, MCPR is equal to or has margin to the safety limit so as to preclude thermal failure with overheating.	A	1
2	2	Fuel Cooling	2-1	Functions to cool reactor core	A	1	1.1, 1.3, 1.4, 4.2	Conformity to the claim is evaluated with the design basis analysis (see [Ref-11-25]). Event corresponding to IDs are described in Table 24.4-1.	Normal Conditions and Fault Conditions	NF SC 34 (THD SFC 2-1.2)	At normal operation and frequent design basis faults, when MCPR becomes less than the safety limit, cladding temperature is less than 800 °C such that ballooning rupture failure is precluded.	A	1
3	1	Control of Reactivity	1-8	Functions to suppress reactor power increase with other system	C	3	-	Conformity to the claim is evaluated in the nuclear design (see subsection 11.6.6.4 and [Ref-11-6][Ref-11-7]). No corresponding event	Normal Conditions and Fault Conditions	NF SC 35 (THD SFC 1-8.1)	The calculated decay ratio for thermal hydraulic stability does not exceed the limiting criteria in the normal operating region.	C	3

	Top Claim for Thermal Hydraulic Design													
	Fundamental Safety Function (FSF) Table 5.6-1: High Level Safety Functions in UK ABWR		High Level Safety Function (HLSF) Table 5.6-1: High Level Safety Functions in UK ABWR				Fault Schedule (Bounding Fault) Topic Report on Fault Assessment Table.4.2-1 Fault Schedule [Ref-11-33]		Safety Functional Claim (SFC)					
			No.	HLSF	Cat.	Class	No.	Contents	State	Claim ID	Claim Contents	Cat.	Class
4	1	Control of Reactivity	1-8	Functions to suppress reactor power increase with other system	C	3	-	Conformity to the claim is evaluated in the nuclear design (see subsection 11.6.6.4 and [Ref-11-6][Ref-11-7]). No corresponding event	Normal Conditions and Fault Conditions	NF SC 36 (THD SFC 1-8.2)	In case that power oscillation due to thermal hydraulic instability occurs, it is controlled and the reactor is led to a stable state.	C	3

	Top Claim for Core Monitoring													
	Fundamental Safety Function (FSF) Table 5.6-1: High Level Safety Functions in UK ABWR		High Level Safety Function (HLSF) Table 5.6-1: High Level Safety Functions in UK ABWR				Fault Schedule (Bounding Fault) Topic Report on Fault Assessment Table.4.2-1 Fault Schedule [Ref-11-33]		Safety Functional Claim (SFC)					
			No.	HLSF	Cat.	Class	No.	Contents	State	Claim ID	Claim Contents	Cat.	Class
1	5	Others	5-12	Supporting functions for management of normal operation	-	3	-	Conformity to the claim is evaluated in core monitoring system design (see subsection 11.6.7, [Ref-11-21] and [Ref-11-26]). No corresponding event	Normal Conditions	NF SC 37 (CMS SFC 5-12.1)	Core performance parameters such as power, flow, MLHGR, MCPR and exposure are monitored so that the fuel integrity is maintained during the reactor operation. These parameters are calculated using the measurement values such as neutron flux, pressure and flow.	-	3

Appendix B: Document Map