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UK ABWR

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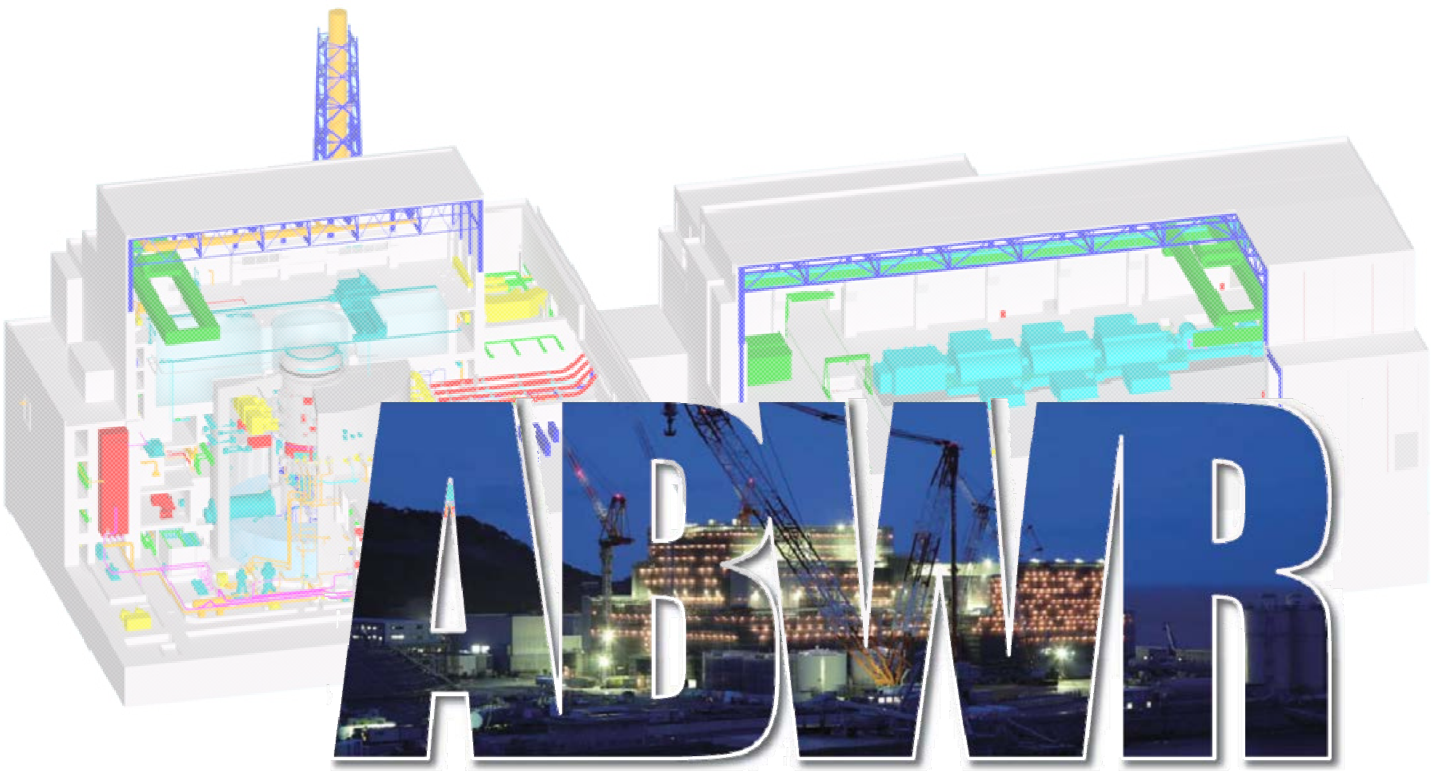
UK ABWR Generic Design Assessment

Preliminary Safety Report on Reactor Core and Fuels



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Abbreviations and Acronyms

Abbreviations and Acronyms	Description
ABWR	Advanced Boiling Water Reactor
ARI	Alternative Rod Insertion
ASME	American Society of Mechanical Engineers
ATWS	Anticipated Transient Without Scram
BWR	Boiling Water Reactor
CBH	Control Blade History
CPR	Critical Power Ratio
CR	Control Rod
CRD	Control Rod Drive
CUW	Clean Up Water
FMCRD	Fine Motion Control Rod Drive
GNF	Global Nuclear Fuel
HCU	Hydraulic Control Unit
LHGR	Linear Heat Generation Rate
LOCA	Loss-Of-Coolant Accident
LPRM	Local Power Range Monitor
MCPR	Minimum Critical Power Ratio
MLHGR	Maximum Linear Heat Generation Rate
NMS	Neutron Monitoring System
RC&IC	Rod Control & Information System
RIP	Reactor Internal Pump
RPS	Reactor Protection System
RPV	Reactor Pressure Vessel
SLCS	Standby Liquid Control System
SLMCPR	Safety Limit Minimum Critical Power Ratio
TIP	Transverse In-core Probe

1. Summary

1.1. Summary

The Advanced Boiling Water Reactor (ABWR) is a forced recirculation direct cycle reactor using light water as a coolant and moderator. It is equipped with an internal steam separator and reactor internal pumps for recirculation. The rated thermal power is 3,926 MW. The major safety systems of the reactor (safety relief valves, emergency core cooling system, primary containment vessel, etc.) are designed for a thermal power of 4,005 MW (about 102% of the rated power).

The ABWR reactor consists of a reactor pressure vessel, reactor internals, core, control rods, fine-motion control rod drives, etc. Figure 1-1 shows an overview of these components. The reactor pressure vessel includes the core support structure, reactor internals, steam separator, steam dryer, etc.

The core is surrounded by a core shroud that is designed to separate the coolant that flows upward in the core from the coolant that flows down in the annular area between the core shroud and the reactor pressure vessel wall.

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

One set of four fuel assemblies in the core interior area is supported by a centre fuel support piece placed on top of a control rod guide tube. The fuel assemblies 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 fuel assembly is supported in the lateral direction by a top guide, which is also supported by the core shroud.

The core has a right cylindrical shape that is about 3.8 m in height and about 5.2 m in equivalent diameter; it consists of 872 fuel assemblies and 205 control rods. Figure 1-2 shows the core configuration. The fuel assembly type is 10x10, which consists of 92 fuel rods per assembly and 2 large-diameter water rods.

The control rod is inserted into and withdrawn from the core through a control rod guide tube at the lower plenum of the reactor. Each control rod is connected to a fine-motion control rod drive by a coupling. The fine-motion drives allow for small power changes, faster start-up times and improved power manoeuvring.

Ten reactor internal pumps, which are placed at the reactor pressure vessel bottom, penetrate the shroud support plate. The motor of the reactor internal pump is housed within a motor casing, and drives the pump impeller through a shaft. By using pumps attached directly to the reactor vessel, jet pumps and the external recirculation system (i.e., recirculation pumps, piping, valves, etc.) have been eliminated. With no large external piping attached to the vessel below the top of the core, the ABWR core remains submerged and cooled in the event of emergency core cooling system initiation following a pipe rupture.

As shown in Figure 1-3, the coolant from the reactor internal pump is distributed to each fuel assembly appropriately as the coolant passes through the core lower plenum and the orificed fuel support piece. The coolant is heated as it passes through the assembly, exiting 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 then enters the stand pipe of the steam separator.

A centrifugal effect within the steam separator removes water from the mixture. The steam then enters the steam dryer where the remaining moisture is mostly extracted. This dry steam exits the reactor pressure vessel through the 4 main steam pipes. 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 means of the reactor internal pumps.

The reactor has a Neutron Monitoring System (NMS) to monitor reactor power over the full range of operation, from start up to rated power (1).

Table 1-1 shows the major specifications of the reactor and the core.

1.2. References

(1) Section 6.10, ABWR General Description, XE-GD-0126

Table 1-1 Major specifications of reactor and core

Reactor thermal power	3,926 MW
Reactor coolant recirculation pump	10 units
Core flow rate*	52.2 x 10^3 t/h
Core inlet sub-cooling*	54.0 kJ/kg
Average steam quality at core outlet*	14.6wt%
Reactor pressure (Pressure vessel at dome)*	7.07 MPa [gage]
Core Effective height*	3.81 m
Equivalent diameter*	5.16 m
Steam flow rate*	7.64 x 10^3 t/h
Steam pressure*	7.07 MPa [gage]
Steam temperature*	287 ℃

*Values are approximate.

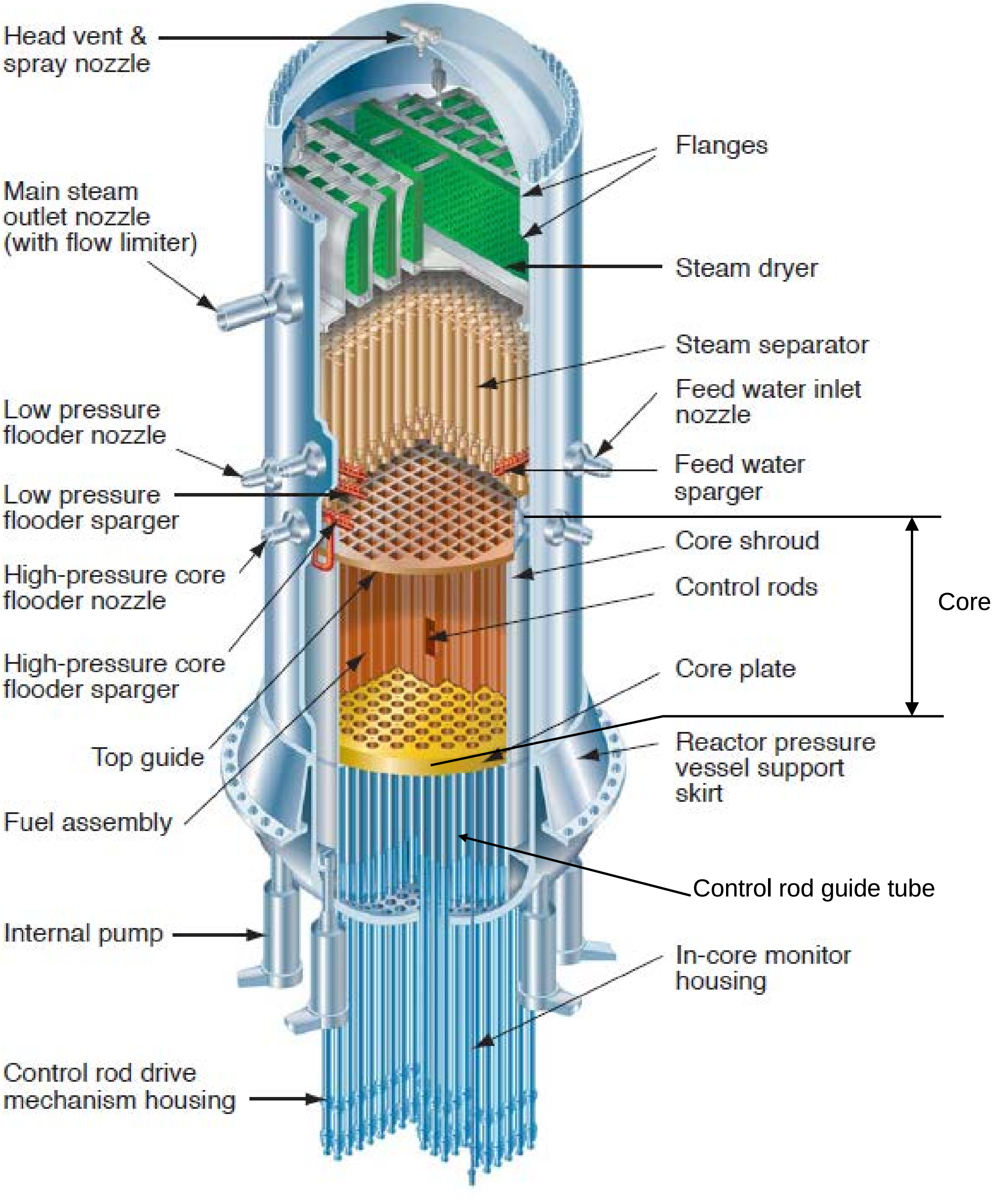

Figure 1-1 Overview of ABWR reactor

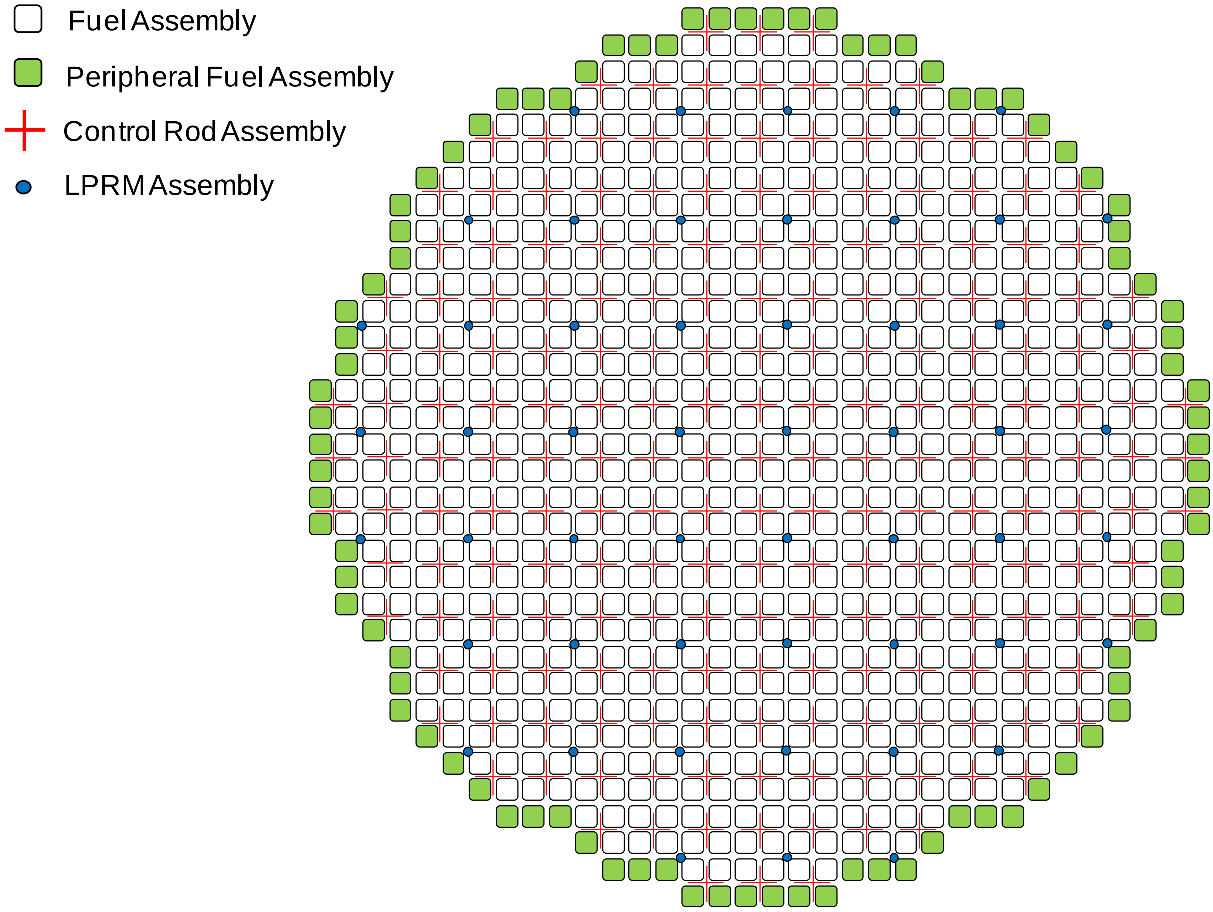

Figure 1-2 ABWR core configuration

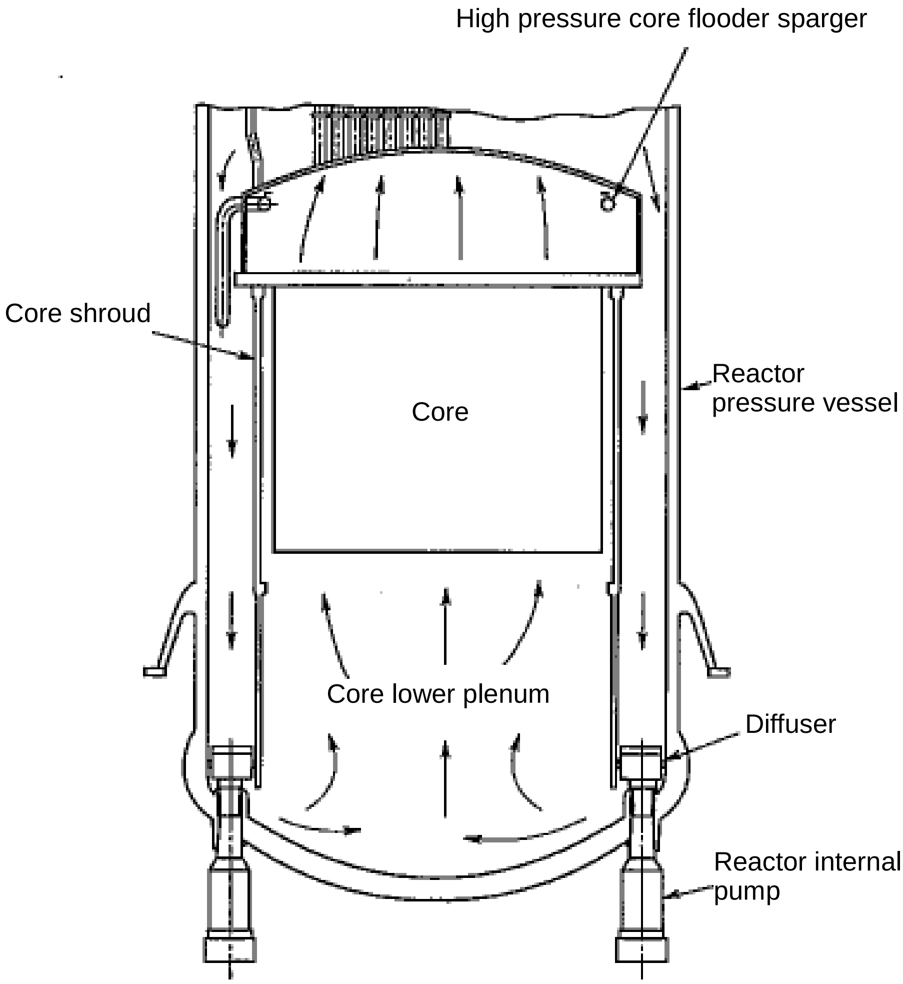

Figure 1-3 Flow of coolant in reactor pressure vessel

2. Fuel System Design

The fuel system is defined as consisting of the fuel assembly and the reactivity control assembly. The fuel assembly is comprised of the fuel bundle, fuel channel and channel fastener. The fuel bundle is comprised of fuel rods, water rods, expansion springs, spacers, and upper and lower tie plates.

The fuel to be loaded in an ABWR, designated as GE14, has been deployed in reload quantities for over fifteen years and has been approved for use in the following countries: Germany, Switzerland, Sweden, Finland, Spain, Mexico and the U.S. GE14 has also been selected as the initial core fuel for two ABWRs in Taiwan.

To demonstrate ABWR system response, a reference core of GE14 fuel is used. This core is shown in Section 3.

Regarding the reactivity control system, Section 2 addresses only the reactivity control elements that extend from the coupling interface of the fine-motion control rod drive (FMCRD) mechanism.

2.1. Design Bases

2.1.1. Fuel Assembly

The fuel system (comprised of the fuel bundle, fuel channel, channel fastener and control rod) is designed to ensure that fuel damage, should it occur, would not result in the release of radioactive materials in excess of limits prescribed in Subsection 2.3.1. Evaluations are performed in conjunction with the core nuclear characteristics, the core hydraulic characteristics, the plant equipment characteristics, and the instrumentation and protection systems to assure that this requirement is met.

The fuel system has the following functional requirement in normal operation and frequent design basis faults:

- Containment of radioactive materials, in particular fission products.

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

- Control of core reactivity and safe core shutdown under all circumstances.
- Residual heat removal through preservation of a coolable geometry.

2.1.2. Control Rod

The control rod is designed to have:

(1) Sufficient mechanical strength to prevent displacement of its reactivity control material.
(2) Sufficient strength to prevent deformation that could inhibit its motion.

2.2. Description

2.2.1. Fuel Assembly

The 10x10 fuel assembly is shown in Figure 2-1. A fuel assembly consists of a fuel bundle and a fuel channel that surrounds the fuel bundle. The GE14 design utilizes a 10x10 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 fuel bundle from one location to another. An identifying fuel assembly serial number is engraved on the top of the handle; no two assemblies bear the same serial number. A boss projects from one side of the handle to ensure proper orientation of the assembly in the core. Finger springs located between the lower tie plate and fuel channel are utilized to control the leakage flow into the bypass region (i.e., the region outside of the fuel channel). Additional bypass flow enters the bypass region through 2 properly sized holes in the lower tie plate casting in order to 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 8 spacer grids that maintain proper spacing of fuel rods along the axial length of the bundle as well as influencing critical power performance.

2.2.1.1. Fuel Rods

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 screws 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 and cold shutdown zone shaping characteristics (note that the part length rod geometry of the GE14 design reduces the need to use gadolinia for shutdown zone 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 UO_2 and the internal pressure resulting from the helium 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 minimize the movement of the column of fuel pellets inside the fuel rod during shipping and handling.

2.2.1.2. Water Rods

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 lock them into position.

2.2.1.3. 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 10x10 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.

2.2.1.4. Finger Springs

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.

2.2.1.5. Debris Filter

The lower tie plate of the GE14 bundle houses a debris filter. This filter provides resistance to debris fretting, thereby substantially improving fuel reliability.

2.2.1.6. Expansion Springs

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

2.2.1.7. Fuel Channels

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

(1) Forms the fuel bundle flow path outer periphery for bundle coolant flow.
(2) Provides surfaces for control rod guidance in the reactor core.
(3) Provides structural stiffness to the fuel bundle during lateral loadings applied from fuel rods through the fuel spacers.
(4) Minimizes, in conjunction with the finger springs and bundle lower tie plate, coolant bypass flow at the fuel channel/lower tie plate interface.
(5) Transmits fuel assembly seismic loadings to the top guide and fuel support of the core internal structures.
(6) Provides a heat sink during loss-of-coolant accident (LOCA).
(7) Provides 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, that also includes the fuel assembly 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 control rod 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 minimizing material for improved neutron economy.

2.2.2. Control Rods

The control rods (Figure 2-2) 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 control rods to counterbalance steam void effects at the top of the core.

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

Rollers at the top handle and bottom connector of the control rod guide the control rod as it is inserted and withdrawn from the core.

The boron carbide powder in the absorber tubes is compacted to about 70% of its theoretical density and contains a minimum of 76.5% by weight of natural boron. The boron-10 minimum content of the boron is 18% by weight. The absorber tubes are sealed by a plug welded into each end. The boron carbide is longitudinally separated into individual compartments by stainless steel balls at approximately 40 cm intervals. The steel balls are held in place by a slight crimp of the tube.

2.3. Design Evaluation

2.3.1. Fuel Assembly

2.3.1.1. Fuel Rod Thermal-Mechanical Design

(1) Evaluations Methods

The fuel rod thermal-mechanical evaluations are all performed using the PRIME03 fuel rod thermal-mechanical performance model. The stress/strain methodology is described later in this subsection.

(a) The 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
- The PRIME03 material properties and component models represent the latest experimental information available.

(b) 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.

(2) Evaluation results

Fuel rod thermal-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.

2.3.2. Control Rods

2.3.2.1. Evaluation Results

The control rod evaluations have been completed for the reference control rod. The evaluations demonstrate that the criteria are satisfied for the reference B_4C control rod.

2.4. Testing, Inspection, and Surveillance Plans

GNF-A has an active program of surveillance of fuel, both production and developmental.

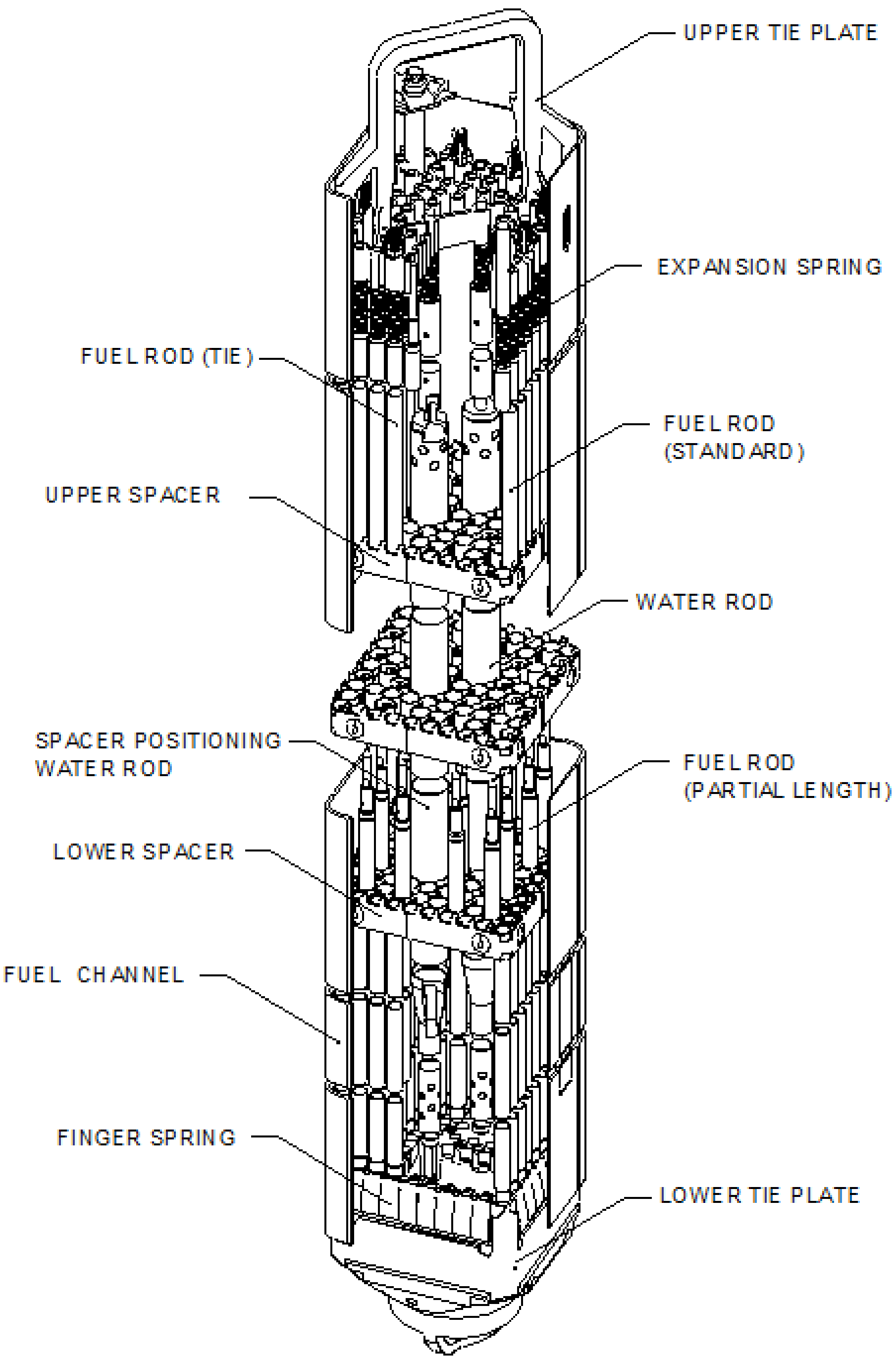

Figure 2-1 Fuel Assembly

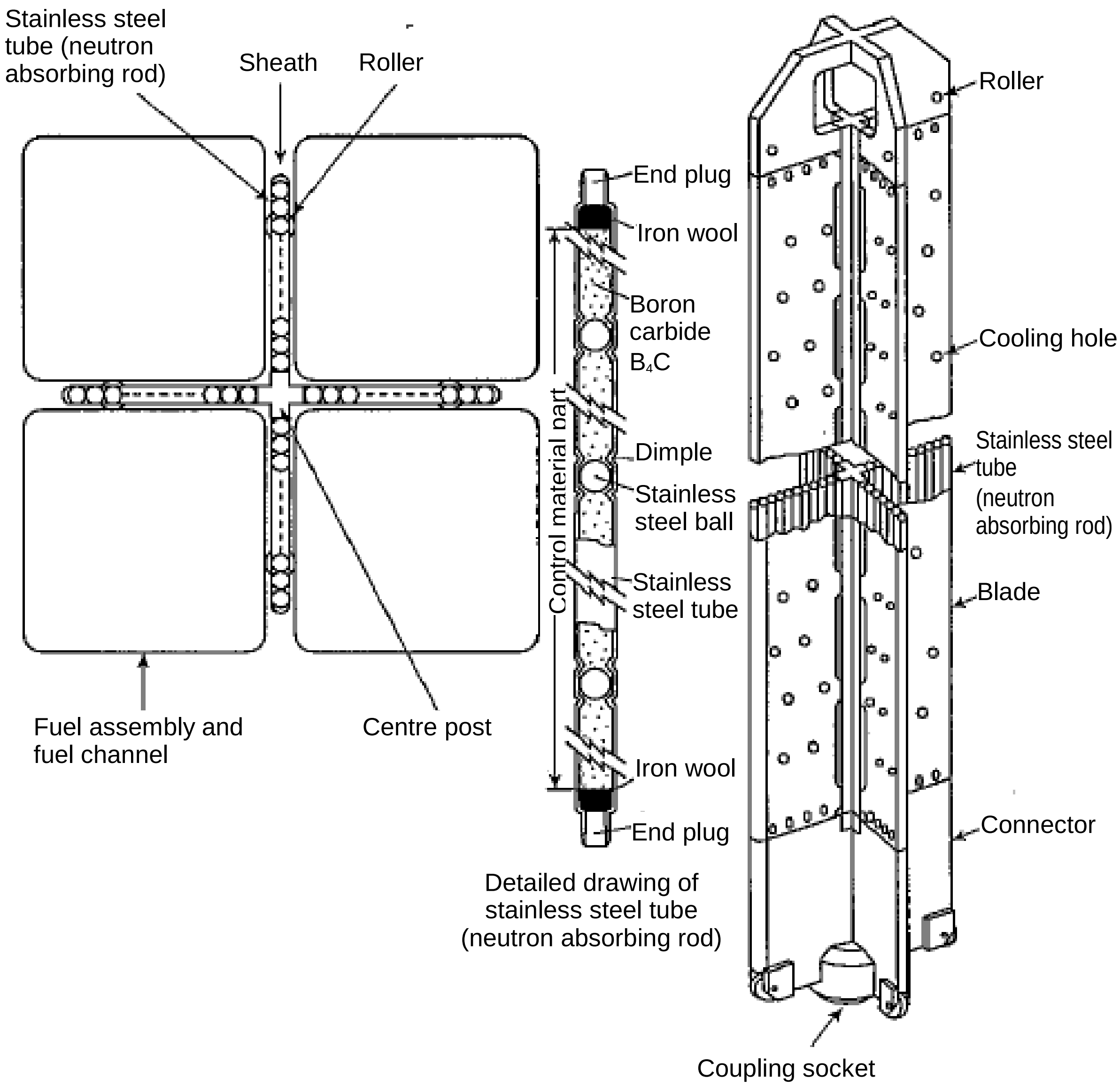

Figure 2-2 Control Rod Assembly

3. Nuclear Design

This section describes the nuclear core design bases and the models used to analyze the fuel.

3.1. Functional requirement

The safety functional requirements met by the neutronic core design are:

- Control of core reactivity to enable the chain reaction to be stopped under all circumstances and to return the reactor to a safe shutdown state
- Removal of heat produced in the fuel via the coolant fluid
- Containment of radioactive substances (actinides and fission products)

The nuclear design must ensure these safety functions are achieved for all design basis operating conditions.

3.2. Design base

3.2.1. Design base for safety

The following items for the design are considered with regards to reactor safety:

(a) Nuclear power restriction characteristics of reactor
With relevant functions of the reactor cooling system, the reactor shutdown system, the instrumentation and control system and the safety protection system, the reactor has inherent power restriction characteristics so as not to exceed the allowable fuel design limits for frequent design basis faults, and not to exceed allowable fuel design limits related to the fuel enthalpy for reactivity insertion transients (e.g., unplanned withdrawal of control rod at the start of the reactor).

(b) Power oscillation control characteristics of reactor
Power oscillations are readily detected and suppressed, should they occur.

(c) Maximum reactivity worth and reactivity insertion rate of control rod
The maximum reactivity worth and the reactivity insertion rate of the control rod are small enough that 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.

(d) Independence of reactor shutdown system
The reactor shutdown system is able to bring the core to a subcritical state from a hot stand-by or power operation condition. It has two independent systems - the control rod and fine-motion control rod drive system; and the Standby Liquid Control System (SLCS) - that can maintain a subcritical state at hot stand-by condition.

(e) Shutdown margin with control rod
The control rod and fine-motion control rod drive system of the reactor shutdown system are able to bring the core to a subcritical state at hot or cold conditions when one or more control rods with the largest reactivity worth (specifically, one rod or a pair of rods belonging to the same hydraulic control unit) are completely withdrawn out of the core and cannot be inserted.

(f) Hot shutdown capability of reactor shutdown system
For frequent design basis faults, the control rod and fine-motion control rod drive system of an independent system included in the reactor shutdown system prevent exceeding the allowable fuel design limit, and are able to bring the core to a subcritical state at hot conditions and maintain this state at hot conditions.

(g) Cold shutdown capability of reactor shutdown system
At least one of the independent systems included in the reactor shutdown system is able to bring the core to a subcritical state at cold conditions and maintain this state at cold conditions.

Items (d), (f) and (g) are described in Section 5 Reactivity Control System.

3.2.2. Design base for operation

The following items for the design are considered with regards to reactor operation:

(a) Operation performance of reactor
Under normal operation, the reactor is able to be operated without exceeding the thermal limits and to adjust the power level.

(b) Required reactivity of reactor
There is sufficient reactivity to achieve the target burn-up under rated power operation.

(c) Function of reactivity control system
The reactivity control system is capable of adjusting core reactivity during normal operation and to maintain the reactor at a specified operation status.

3.2.3. Specific items in design base

To satisfy the design bases in 3.2.1 and 3.2.2, the following specific items for the design are considered:

(a) Reactivity coefficients
The reactivity coefficient, defined as the moderator void coefficient and the Doppler coefficient, is always negative, and always satisfies the following conditions. Validity of its value is confirmed through core characteristics evaluation, rather than restricting the value itself.

i The overall moderator void coefficient is negative.

ii The Doppler coefficient is negative and has sufficient reactivity feedback characteristics for infrequent design basis faults (i.e., reactivity insertion accident).

iii The moderator temperature coefficient is not a value that may pose a problem in the current design range and it has no specific requirement.

(b) Spatial oscillation of xenon
The power reactivity coefficient is large enough to reduce the spatial oscillation of xenon.

(c) Maximum reactivity worth and reactivity insertion rate of control rod
For the maximum reactivity worth and the reactivity insertion rate of the control rod, the following restrictions are applied:

i. In selection of the operating control rod pattern and the control rod withdrawal sequence, the reactivity worth for simultaneous withdrawal of multiple control rods belonging to the same control rod group while approaching criticality is less than the maximum worth allowed by the control rod worth minimiser. If one control rod drops while approaching criticality, the maximum worth of the dropped control rod is less than the maximum worth allowed by the control rod worth minimiser. A dropped control rod 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.

ii. The control rod movement step is designed so that an operator can safely control the reactor with an adequate reactor period when multiple control rods belonging to the same withdrawal group are simultaneously moved by one step.

(d) Shutdown capability of Standby Liquid Control System
The Standby Liquid Control System is capable of bringing the core to a subcritical state from the hot stand-by or power operation condition and to achieve and maintain a subcritical state at a temperature of 20 degree-C while compensating for the reactivity insertion due to the xenon decay and change of the coolant temperature.

(e) Shutdown margin with control rod
The control rod design and the concentration of the burnable neutron absorber of the fuel assembly are designed so that the criteria in 3.2.1(e) are satisfied.

(f) 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) are satisfied.

(g) Operation cycle length
Operation cycle length is determined based on rated power operation considering refuelling duration and capacity factor.

(h) Function of reactivity control system
With adjustment of the control rod position and the recirculation flow, the reactivity of reactor is adjusted and the reactor is maintained at the specified operation status.

3.3. Description

The ABWR core design consists of a light-water moderated reactor, fuelled with slightly enriched uranium-dioxide. The use of water as a moderator produces a neutron energy spectrum in which fissions are caused principally by thermal neutrons. At 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 reduce the power. A reference core loading of 872 fuel bundles is used as the basis for the system dynamic response evaluation.

This reference core loading pattern (an equilibrium core of GE14 fuel) is provided in Figure 3-1. 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 minimize 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 after withdrawing control rods and/or alternating the control rod pattern using 29 control cell locations.

3.3.1. 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 of xenon and samarium, and burn-up of the fuel.

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

(a) 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 the operating conditions including the capacity factor, etc. 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 at 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 1/4 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.

(b) Unscheduled refuelling
An unplanned refuelling outage may occur during an operating cycle due to various causes, for example, high offgas from damaged fuel. In this case, refuelling is performed such that shutdown margin and thermal limits are always satisfied. However, the refuelling interval and target burn-up may be adjusted.

3.3.2. Control rod withdrawal procedure and control rod pattern

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

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

The above procedure for control rod withdrawal is not unique. The control rod 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 control rod worth tends to become extremely small.

The control rod withdrawal procedure and the control rod pattern are determined by calculation and actual measurement. In other words, the control rod withdrawal procedure and the control rod pattern are initially determined 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 control rod pattern is used as an operating guideline by plant operations. The actual control rod pattern is, however, determined for each operation.

The hot excess reactivity is suppressed primarily by control rod insertion. Core flow is maintained at 90% of rated and increased at the end of cycle for neutron spectrum operation, often referred to as spectral shift operation. Two control rod patterns comprised of 12 to 16 rods are alternated every 3.3GWd/t to mitigate the power peaking of the fuel adjacent to control rods, often referred to as control rod blade history (CBH) effects.

3.3.3. 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 control rod, 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 control rod.

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

3.3.4. Power Distribution

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

3.3.4.1. Power Distribution Anomalies

Stringent inspection procedures are utilized 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, 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 stabilizing 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 control rods 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 control rods and flow, then the total core power would have to be reduced.

3.3.5. Reactivity Coefficients

Reactivity coefficients, the differential changes in reactivity produced by differential changes in core conditions, are useful in calculating stability and evaluating the response of the core to external disturbances. The reactivity coefficients are shown below:

(a) Fuel rod temperature coefficient (Doppler coefficient)
(b) Moderator void coefficient
(c) Moderator temperature coefficient

These reactivity coefficients are combined and called the power reactivity coefficient.

In general, the power reactivity value is negative. The moderator temperature coefficient at low temperature may become less negative at the end of the cycle (and may become positive at higher core average burn-up). However, the system is designed so that the power reactivity coefficient is always negative during the cycle.

(a) 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 UO2 fuel temperature is increased but negative reactivity is inserted with 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 runaway. The Doppler effect plays an important role for safety of the reactor.

The insertion of reactivity with 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:

$$\Delta k_{DOP} = C(\sqrt{T} - \sqrt{T_o}) \qquad (3\text{-}1)$$

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:

$$\frac{1}{k}\cdot\frac{dk}{dT} = \frac{C}{2[k_o + C(\sqrt{T} - \sqrt{T_o})]\sqrt{T}} \qquad (3\text{-}2)$$

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 as burn-up progresses, the Doppler coefficient is negative and the absolute value increases.

(b) 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:

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

V:	Channel void fraction
C:	Control rod insertion ratio
M^2:	Neutron migration area
B^2:	Buckling
k_{∞}:	Infinite multiplication factor (UC: without control rod, C: with control rod)
k_{eff}:	Effective multiplication factor

As moderator density decreases, the neutron leakage is increased and the control rod 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 control rod worth are increased monotonously. The control rod insertion ratio C is reduced along with the decrease in the control rod density; this reduction occurs throughout the operating cycle with increasing burn-up.

(c) 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 smallness of the value, 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 the design.

(d) 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 about -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.

3.3.6. Control Requirements

The ABWR control rod 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, xenon-free core.

3.3.6.1. 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 control rod or any control rod pair with same hydraulic control unit (HCU), fully withdrawn and all other rods fully inserted. The shutdown margin is determined by using the BWR simulator code (see Section 6.3) to calculate the core multiplication at selected exposure points with the strongest rod or strongest rod pair with 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.

3.3.6.2. Reactivity Variations

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

3.3.6.3. Standby Liquid Control System

The Standby Liquid Control System (SLCS) is designed to provide the capability of bringing the reactor, at any time in a cycle, from a full power and minimum control rod inventory (which is defined to be at the peak of the xenon transient) to a sub-critical condition with the reactor in the most reactive xenon-free state. The requirements of this system are dependent primarily on the reactor power level and on the reactivity effects of voids and temperature between the full-power and cold, xenon-free condition.

3.3.7. Stability

3.3.7.1. Xenon Transients

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

1. Operating experiences that has no 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 Reference (3).

3.3.7.2. Thermal Hydraulic Stability

Thermal hydraulic stability is described in subsection 4.4.3.

3.3.8. Analytical Methods

The nuclear evaluations of all cores are performed using the analytical tools and methods.

The calculation method and the nuclear data used for analysis of the nuclear performance of the ABWR are similar to those usually used in the design of the power generation light water reactor.

In general, calculation of the reactor nuclear performance falls into two categories. They are the unit fuel assembly nuclear calculation (5) that calculates characteristics of the unit fuel assembly including the external water gap, and the total core nuclear thermal-hydraulic calculation (6) (7) that calculates nuclear thermal-hydraulic characteristics of the total core. The former is classified into the calculation to obtain the nuclear constant of energy minority group at each part of the fuel assembly with the fuel lattice analysis model, and the calculation to obtain characteristics of the fuel assembly using this constant.

In the unit fuel assembly nuclear calculation, the average nuclear constant of the fuel assembly or the relative power of each fuel rod in the fuel assembly is obtained by changing the parameters of fuel, moderator temperature, burn-up, void ratio, availability of adjacent control rod, etc. In the total core nuclear thermal-hydraulic calculation, the result of the unit fuel assembly nuclear calculation is used.

These calculation methods are briefly outlined below.

In the fuel lattice analysis model, the fast energy group obtains the calculation result from the energy multi-group B-1 approximation method considering the geometrical shape effect for fast neutron fission with the collision probability method of the energy multi-group. The intermediate energy range is treated with the energy multi-group B-1 approximation method. Resonance absorption is treated with the GAM (8) type. The Dancoff effect, self-shielding effect by fuel and change of neutron spectrum are also considered. The thermal energy group performs a THERMOS (9) type calculation to obtain the neutron spectrum distribution in the fuel assembly. For a strong neutron absorber like control rod or fuel rod with Gd_2O_3, the effect of the ambient medium is particularly considered.

In the unit fuel assembly nuclear calculation, the power distribution in the unit fuel assembly, infinite multiplication factor and average nuclear constant are calculated by the energy minority group 2-D dispersion model. To calculate burn-up, the equation indicating the burn-up process of the fuel is solved. At this time, necessary neutron distribution is calculated for each appropriate burn-up step with the fuel lattice analysis model and the energy minority 2-D dispersion model according to the number of elements.

In the total core nuclear thermal-hydraulic calculation, the power distribution of the total reactor, burn-up, thermal-hydraulic characteristics, shutdown margin of reactor, etc. are calculated. Usually, the 3-D boiling water reactor simulation calculation code (6) (7) that can cover the control rod, void rate, spatial distribution of Gd_2O_3, etc. is used. This calculation code can be used for reviewing the control rod pattern during the operating cycle and calculating power response to changes in core flow rate, etc.

3.4. References

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(6) “Three-dimensional nuclear thermal hydraulic power calculation method - Boiling water type nuclear power station”
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(9) Honeck, H. C., “THERMOS - A Thermalization Transport Theory Code for Reactor Lattice Calculations”, BNL - 5826, June 1961.

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	01	03	05	07	09	11	13	15	17	19	21	23	25	27	29	31	33
01															4	4	4
03												4	4	4	4	4	4
05											4	4	2	4	2	4	2
07								4	4	4	4	2	2	2	2	2	4
09							4	4	2	2	2	2	1	4	1	2	2
11						4	4	2	4	1	4	1	2	1	3	1	3
13					4	4	3	2	1	2	1	2	3	2	1	3	1
15				4	4	2	2	1	2	1	3	1	2	1	3	1	3
17				4	2	4	1	2	2	3	1	3	2	3	1	3	2
19				4	2	1	2	1	3	3	3	1	3	2	3	1	3
21			4	4	2	4	1	3	1	3	1	3	1	3	1	3	1
23		4	4	2	2	1	2	1	3	1	3	1	3	1	3	1	3
25		4	2	2	1	2	3	2	2	3	1	3	2	3	1	3	2
27		4	4	2	4	1	2	1	3	2	3	1	3	2	3	1	3
29	5	4	2	2	1	3	1	4	1	3	1	3	1	3	2	3	1
31	4	4	4	2	2	1	3	1	3	1	3	1	3	1	3	1	3
33	4	4	2	4	2	3	1	3	2	3	1	3	2	3	1	3	3

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 3-1 Reference Equilibrium Core Loading Map

NOT PROTECTIVELY MARKED

4. Thermal–Hydraulic Design

4.1. Functional Requirement

The safety functional requirements met by the neutronic core design are:

- Control of core reactivity to enable the chain reaction to be stopped under all circumstances and to return the reactor to a safe state
- Removal of heat produced in the fuel via the coolant fluid
- Containment of radioactive substances (actinides and fission products)

The thermal hydraulic design must ensure these safety functions are achieved for all design basis operating conditions.

4.2. Design Basis

4.2.1. Design basis for safety

The following items for the design are considered in view of the safety of the reactor:

(a) Nuclear power restriction characteristics of reactor
With relevant functions of the reactor cooling system, the reactor shutdown system, the instrumentation and control system and the safety protection system, the reactor has inherent power restriction characteristics so as not to exceed the allowable fuel design limits for frequent design basis faults, and not to exceed allowable fuel design limits related to the fuel enthalpy for reactivity insertion transients (abnormal withdrawal of control rod at the start of the reactor).

(b) Power oscillation control characteristics of reactor
Power oscillations are readily controlled, should they occur.

4.2.2. Design basis for operation

The following items for the design are considered in view of operation of the reactor:

(a) Operation performance of reactor
Under normal operation, the reactor is able to be operated without exceeding the thermal limits and to make adjustments to power.

4.2.3. Permissible design limits of fuel in thermal hydraulic design

Fuel damage occurs where there is a perforation of cladding leading to the subsequent release of fission products. The permissible design limits of the fuel shall be established with the aim of preventing fuel damage. The following shall be considered as mechanisms of fuel damage.

a. Perforation of cladding by overheating caused by insufficient cooling.
b. Perforation of cladding by strain caused by relative expansion of cladding and fuel pellets.

For “a”, the permissible design limits of fuel (with regard to MCPR) shall, considering standard deviation of each monitoring parameter on reactor core conditions, be established based on the design basis that even if the frequent design basis faults occurs, fuel cladding integrity would be assured. As a result of evaluation by statistical analysis on thermal hydraulic characteristics of a reactor core considering the standard deviation of each parameter, the MCPR is determined. Here, the MCPR is defined in "4.4.5 Analytical Methods.”
For “b”, the linear power density at which an average plastic strain of 1% in the circumferential direction (hereinafter referred to as the “1% plastic strain”) is generated shall be the permissible design limit of fuel in the thermal hydraulic design.

4.3. Description of Thermal-Hydraulic Design of the Reactor Core

4.3.1. Thermal hydraulic characteristic data

The example of thermal-hydraulic parameters for the ABWR is 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 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*	14.6wt%
Core average void fraction*	43%

*Values are approximate.

4.3.2. Critical Power Ratio (CPR)

CPR is the ratio of the fuel assembly power at which boiling transition occurs (critical power) to actual fuel assembly power. Critical power is evaluated by the GEXL boiling transition correlation. A description of the CPR calculation is provided in subsection 4.4.4.

4.3.3. 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 analysis 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.

4.3.4. Core Coolant Flow Distribution and Orifice Pattern

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 (References (1), (2), and (3)). The components of bundle pressure drop considered are friction, local, elevation, and acceleration (Subsections 4.3.6.1 through 4.3.6.4, respectively). Pressure drop measurements made in operating reactors confirm that the total measured core pressure drop and calculated core pressure drop are in good agreement. There is reasonable assurance, therefore, that the calculated flow distribution throughout the core is in close 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. The total core flow less the control rod cooling flow enters the lower plenum. A fraction of this passes through various bypass flow paths. The remainder passes through the orifice in the fuel support plate (experiencing a pressure loss) where some of the flow exits through the fit-up between the fuel support and the lower tie plate and through the lower tie plate holes into the bypass flow region. All initial and reload core fuel bundles have lower tie plate holes. The majority of the flow continues through the lower tie plate (experiencing a pressure loss) where some flow exits through the flow path defined by the fuel channel and lower tie plate into the bypass region.

Within the fuel assembly, heat balances on the active coolant are performed nodally. Fluid properties are expressed as the bundle average at the particular node of interest and are based on ASME Steam Tables (American Society of Mechanical Engineers, New York, 1989). 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 fuel assembly type. When the equal pressure drop criterion described above is satisfied, the flow distributions are established.

4.3.5. Void Fraction Distribution

The coolant flows into the inlet of the fuel assembly as single phase water. It is heated as it passes through the fuel assembly, and becomes a two-phase flow mixture of steam and water.

4.3.6. 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.

4.3.6.1. Friction Pressure Drop

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

$$\Delta P_f = \frac{w^2}{2g\rho}\frac{fL}{D_H {A_{ch}}^2}\phi_{TPL}^2 \qquad (4\text{-}1)$$

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_{TPL} = 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.

4.3.6.2. 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 fuel assembly.

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

$$\Delta P_L = \frac{w^2}{2g\rho}\frac{K}{A^2}\phi_{TPL}^2 \qquad (4\text{-}2)$$

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, and ρ are as previously defined. The formulation for the two-phase multiplier is similar to that reported in Reference (5). The local loss component of the total pressure drop across a region inside the fuel assembly 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.

4.3.6.3. Elevation Pressure Drop

The elevation pressure drop is based on the relation:

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

$$\bar{\rho} = \rho_f(1-\alpha) + \rho_g\alpha \qquad (4\text{-}3)$$

where

ΔP_E	=	elevation pressure drop
ΔL	=	incremental length

$\overline{\rho}$	=	average mixture density
α	=	nodal average void fraction
ρ_f , ρ_g	=	liquid and saturated vapor density, respectively

The void fraction model used is an extension of the Zuber-Findlay model (Reference (6)), and uses an empirical fit constant to predict a large block of steam void fraction data.

4.3.6.4. 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:

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

$$\sigma_A = \frac{A_2}{A_1} = \frac{final\ \ flow\, area}{initial\ \ flow\, area} \qquad (4\text{-}4)$$

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:

$$\Delta P_{ACC} = \left(1-\sigma_A^2\right)\frac{w^2 \rho_H}{2g\rho_{KE}^2 A_2^2} \qquad (4\text{-}5)$$

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 (4-6)

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:

$$\Delta P_{ACC} = \frac{w^2}{g A_{ch}^2} \left(\frac{1}{\rho_{out}} - \frac{1}{\rho_{in}} \right) \tag{4-7}$$

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

$$\frac{1}{\rho_M} = \frac{x^2}{\rho_g \alpha} + \frac{(1-x)^2}{\rho_f (1-\alpha)} \tag{4-8}$$

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 on the order of a few percent of the total pressure drop.

4.3.7. Correlation and Physical Data

Substantial amounts of physical data in support of the pressure drop and thermal-hydraulic loads discussed in Subsection 4.3.6 have been obtained. Correlations have been developed to fit these data to the formulations discussed.

4.3.7.1. 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 4.3.6.1 and 4.3.6.2. 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 discussed in Subsections 4.3.6.1 and 4.3.6.2 for the fuel designs described in Section 2, was confirmed by full scale prototype flow tests.

4.3.7.2. Void Fraction Correlation

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

4.3.7.3. Heat Transfer Correlation

The Jens-Lottes (Reference (7)) heat transfer correlation is used in fuel design to determine the cladding-to-coolant heat transfer coefficients for nucleate boiling.

4.3.8. Uncertainties in Estimates

Uncertainties in thermal-hydraulic parameters are considered in the statistical analysis that is performed to establish the fuel cladding integrity safety limit documented in Subsection 4.4.5.1.

4.4. Description of the Thermal–Hydraulic Design of the Reactor Coolant System

4.4.1. 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 Section 1.

4.4.2. Power/Flow Operating Map

4.4.2.1. 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. 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:

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

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

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

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

iv **Constant Pump Speed Lines:** These lines show the change in flow associated with power changes while maintaining constant RIP speeds.

v **Natural Circulation Line:** The operating state of the reactor moves along this line for the normal control rod 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% power rod line, i, iii and iv.

4.4.2.2. 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 pressurization 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 control rods.

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

4.4.3. Thermal Hydraulic Stability Performance

In general, BWRs have a large negative power coefficient and a self-controllability against disturbances of reactivity caused by various operations such as control rod 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 the previous BWR plant types.

- A 155.5 mm fuel lattice pitch that is wider than a BWR5 lattice pitch of 152.4mm lattice. This provides more non-boiling area in the core and less negative void coefficient.

- 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).

- The inner width of the channel box 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.

- A larger number of low ΔP 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 4.4.2.

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, selected control rods are automatically inserted at high core power / low core flow conditions.

As such, ABWR stability characteristics will be confirmed with analyses at the maximum power point (102% of rated power) with minimum RIP speed. The results of the analysis will determine the core decay ratio and channel decay ratio that satisfy the design criteria (decay ratio less than 1.0).

4.4.4. Evaluation

4.4.4.1. Critical Power

The objective for normal and frequent design basis faults is to maintain fuel cladding integrity. The figure of merit utilized 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 fuel assembly in the core. To ensure that adequate margin is maintained, a design requirement based on a statistical analysis was selected as follows:

Frequent design basis faults caused by a single operator error or equipment malfunction shall be limited such that, considering uncertainties in manufacturing and monitoring the core operating state, fuel cladding integrity would be assured.

4.4.4.2. Core Hydraulics

Core hydraulics models and correlations are discussed in Subsection 4.3.

4.4.5. Analytical Methods

4.4.5.1. 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 is also applied to the localized rod withdrawal error.

4.4.5.1.1. Statistical Model

The statistical analysis utilizes 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% of the rods are expected to avoid boiling transition by interpolation among the means of the distributions formed by all the trials.

4.4.5.1.2. Bounding BWR Statistical Analysis

Statistical analyses have been performed that provide fuel cladding integrity safety limit MCPRs applicable to all GNF fuel designs for the ABWR reload and initial core cycles. The results of the analyses show that at least 99.9% 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.

4.4.5.1.3. 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 exceeded for any frequent design basis faults.

4.4.5.1.4. Evaluation procedure for frequent design basis faults

The evaluation of frequent design basis faults are described in Reference (8).

4.4.5.1.5. CPR calculation with channel bow

Channel bow influences the thickness of the water gap between bundles and therefore impacts the pin power distributions inside the bundle. The pin power distribution influences the bundle R-factor, thereby impacting the bundle CPR. The R-factor is a parameter that characterizes the radial power peaking in the bundle. Therefore, the R-factor used in cycle specific core design and core monitoring calculations is dependent on the amount of channel bow. The amount of channel bow applicable to a specific core design is determined using a channel bow prediction model. This amount is applied to the R-factor calculation and these adjusted R-factors are then used in the CPR calculation to account for the effect of channel bow on CPR.

4.4.5.2. 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:

Xc = f(LB, DQ, G, L, LA, P, R)

where,

Xc:	Critical quality
LB:	Boiling length
DQ:	Thermal diameter (i.e., 4A/(total rodded perimeter))
G:	Mass flow rate per unit area
L:	Heated length
LA:	Annular flow length
P:	System pressure
R:	Factor given in the function of local peaking

As shown in the above equation, critical quality has a typical feature that the influence of axial power distribution is expressed with the length of boiling and the influence of local power distribution with the R-factor. Critical quality is defined by a random point in axial direction of a fuel assembly, and when actual steam quality at that point is larger than the critical quality, boiling transition starts. The fuel assembly power at that time is defined as critical power.

4.5. References

(1) "Core Flow Distribution in a Modern Boiling Water Reactor as Measured in Monticello", NEDO-10299A, October 1976.

(2) H.T. Kim and H. S. Smith, "Core Flow Distribution in a General Electric Boiling Water Reactor as Measured in Quad Cities Unit 1", NEDO-10722A, August 1976.

(3) "Brunswick Steam Electric Plant Unit 1 Safety Analysis Report for Plant Modifications to Eliminate Significant In-Core Vibrations", NEDO-21215, March 1976.

(4) R.C. Marinelli and D.E. Nelson, "Prediction of Pressure Drops During Forced Convection Boiling of Water", ASME Trans., 70, 695-702, 1948.

(5) C.J. Baroozy, "A Systematic Correlation for Two-Phase Pressure Drop", Heat Transfer Conference (Los Angeles), AICLE, Preprint No. 37, 1966.

(6) N. Zuber and J.A. Findlay, "Average Volumetric Concentration in Two-Phase Flow Systems", Transactions of the ASME Journal of Heat Transfer, November 1965.

(7) W.H. Jens and P.A. Lottes, "Analysis of Heat Transfer, Burnout, Pressure Drop and Density Data for High Pressure Water", USAEC Report- 4627, 1972.

(8) "Fault Studies to Discuss Deterministic Analysis, PSA and Fault Schedule Development", XE-GD-0105.

(9) "Core computation with process computer - Boiling water type nuclear power station" (Hitachi, Ltd., HLR - 029, March 1985).

(10) "Thermal design method for reactors - Boiling water type nuclear power station" (Hitachi, Ltd., HLR - 008, April 1977).

(11) "General Electric BWR Thermal Analysis Basis", NEDO-10958-A, January 1977.

5. Functional design of reactivity control system

This section describes the reactivity control system basis and the models used to analyze the fuel.

5.1. Functional requirements

The safety functional requirements met by the neutronic core design are:

Control of core reactivity to enable the chain reaction to be stopped under all circumstances and to return the reactor to a safe state.

5.2. Design basis

5.2.1. Design basis for safety

The following items for the design are considered in view of the safety of the reactor:

(a) Independence of reactor shutdown system
The reactor shutdown system is able to bring the core to a subcritical state from a hot stand-by or power operation condition, and has two independent systems (the control rod and fine-motion control rod drive system; and the standby liquid control system) that can maintain the subcritical state at hot stand-by conditions.

(b) Hot shutdown capability of reactor shutdown system
For frequent design basis faults, the control rod and fine-motion control rod drive system of an independent system included in the reactor shutdown system prevent exceeding the allowable fuel design limit, and are able to bring the core to a subcritical state at hot conditions and maintain the subcritical state at hot conditions.

(c) Cold shutdown capability of reactor shutdown system
At least one of the independent systems included in the reactor shutdown system is able to bring the core to a subcritical state at cold conditions and maintain the subcritical state at cold conditions.

5.2.2. Specific items in reactivity control system

(a) Shutdown capability of Standby Liquid Control System
The Standby Liquid Control System is able to bring the core to a subcritical state from the hot stand-by or power operation condition and to achieve and maintain the subcritical state at the temperature of 20 degree-C while compensating for the reactivity insertion due to the decay of xenon and change of the coolant temperature.

(b) Shutdown margin with control rod
The control rod and the concentration of the burnable neutron absorber of the fuel assembly are designed so that the criteria in 3.2.1(e) are satisfied.

5.3. Description

5.3.1. Control Rod Drive (CRD)

The CRD System is composed of three major elements:

(1) Electro-hydraulic fine motion control rod drive (FMCRD) mechanisms,

(2) Hydraulic control units (HCU), and

(3) Control rod drive water pressure system.

The FMCRDs provide electric-motor-driven positioning for normal insertion and withdrawal of the control rods and hydraulic-powered rapid insertion (scram) of control rods during abnormal operating conditions. There are a total of 205 FMCRDs mounted in CRD housings welded into the reactor vessel bottom head.

The hydraulic power required for scram is provided by high pressure water stored in 103 individual HCUs. Each HCU contains a nitrogen-water accumulator charged to high pressure and the necessary valves and components to scram two CRs. Additionally, during normal operation, the HCUs provide a flow path for purge water to the associated FMCRDs.

The water pressure system supplies clean, demineralised water that is used to charge the HCU scram accumulators and to provide purge water to the FMCRDs during normal operation. The water pressure system is also the source of pressurized water for purging the Reactor Internal Pumps (RIPs) and the Clean Up Water (CUW) pumps.

The CRD is designed to perform the following functions for power generation and safety:

(1) The CRD through its electric motors position the CRs in the core depending on the control signal from the Rod Control and Information System (RC&IS) when performing normal insertion and withdrawal for control of core reactivity changes.

(2) The CRD through the FMCRDs implements reactor scram operation when receiving the scram signal from the Reactor Protection System (RPS). The FMCRD electric motors are actuated to back up the full insertion of the CRs with the scram follow-in signal from the RPS.

(3) The CRD opens the scram valves provided on the outlet of each HCU accumulator and thereby the pressurized water stored in the HCU accumulator is supplied to the piston section of the FMCRD in the event that the scram signal was initiated. As a result, the FMCRDs are hydraulically driven and each CR is rapidly inserted to shut down the reactor.

(4) Through the Alternative Rod Insertion (ARI) signal, from the Anticipated Transient without Scram System (ATWS), the CRD allows the CRs to be hydraulically inserted in case scram could not be performed upon a scram signal.

(5) At the same time, the CRD actuates the FMCRD electric motors to back up the full insertion of the CRs with the FMCRD run-in signal from the ATWS.

(6) The CRD supplies purge water from the discharge side of the CRD pumps in order to prevent deposition of crud from the reactor side in the FMCRDs during plant normal operation.

(7) The CRD supplies purge water from the discharge side of the CRD pumps in order to prevent leakage of reactor water into the Reactor Internal Pumps (RIPs) and the Clean Up Water (CUW) pumps during plant normal operation.

(8) The CRD is utilized to pressurize the Reactor Pressure Vessel (RPV) when the leakage and hydrostatic test is implemented.

5.3.2. Standby Liquid Control System

The Standby Liquid Control System (SLCS) is designed to provide the capability of bringing the reactor, at any time in a cycle, from a full power and minimum control rod inventory (which is defined to be at the peak of the xenon transient) to a subcritical condition with the reactor in the most reactive xenon-free state. The requirements of this system are dependent primarily on the reactor power level and on the reactivity effects of voids and temperature between full-power and cold, xenon-free conditions.

5.4. Safety evaluation of the CRD system

The safety evaluation of the control rod drives is given below.

5.4.1. Evaluation of scram time

The rod scram function of the CRD System provides the negative reactivity insertion required. The scram time is reflected in plant transient analyses.

5.4.2. Scram reliability

High scram reliability is the result of a number of features of the CRD System. For example:

(1) Each accumulator provides sufficient stored energy to scram two CRDs at any reactor pressure.

(2) Each pair of drive mechanisms has its own scram valve and dual solenoid scram pilot valve; therefore, only a single scram valve needs to open for scram to be initiated. Both pilot valve solenoids must be de-energized to initiate a scram.

(3) The RPS and the HCUs are designed so that the scram signal and mode of operation override all others.

(4) The FMCRD hollow piston and guide tube are designed so they will not restrain or prevent control rod insertion during scram.

(5) Each FMCRD mechanism initiates electric motor-driven insertion of its control rod simultaneous with the initiation of hydraulic scram. This provides a diverse means to assure control rod insertion.

5.4.3. Precluding excessive rate of reactivity addition

Excessive rates of reactivity addition are precluded by the design of the FMCRD. Prevention of rod ejection due to FMCRD pressure boundary failure is reflected in the structural design. Prevention of control rod drop is achieved by the C&I and mechanical design of the CR and FMCRD.

6. Engineering Computer Programme

6.1. PRIME

The PRIME code addresses the effects of fuel/cladding thermal expansion, fuel phase change, volume change, fuel irradiation swelling, densification, relocation and fission gas release, fuel-cladding axial slip, cladding creepdown, irradiation hardening and thermal annealing of irradiation hardening, pellet and cladding plasticity and creep, pellet hot pressing and plastic collapse, and development of a porous pellet rim at high exposure.

PRIME performs coupled thermal and mechanical interaction analyses. The incremental finite element mechanics model performs an axisymmetric radial mechanical interaction analysis to determine pellet and cladding stresses and strains at the pellet mid-height location. The thermal solution is obtained by numerical evaluation of the thermal conductivity integral.

PRIME is applicable to BWR fuel design and licensing analyses. Table 2.1 of NEDC-33256P specifies the range of applicability for various dimensional and performance parameters such as pellet diameter, enrichment, density, linear power and exposure.

References

(1) The PRIME Model for Analysis of Fuel Rod Thermal – Mechanical Performance Part 1 – Technical Bases, Global Nuclear Fuels-Americas, NEDO-33256-A, September 2010.

(2) The PRIME Model for Analysis of Fuel Rod Thermal – Mechanical Performance Part 2 – Qualification, Global Nuclear Fuels-Americas, NEDO-33257-A, September 2010.

(3) The PRIME Model for Analysis of Fuel Rod Thermal – Mechanical Performance Part 3 – Application Methodology, Global Nuclear Fuels-Americas, NEDO-33258-A, September 2010.

6.2. TGBLA

TGBLA is a lattice design computer programme for conventional BWRs that can model the following lattices: 7x7, 8x8, 9x9, or 10x10. Water rods, including large central water rods, may be introduced into cells of the 2-D mesh that TGBLA solves. The 10x10 lattice can have up to 4 cells per water rod. Lattices with vanishing rods, thick-thin channels, or some water-cross designs such as 10x10 cross lattice, are qualified.

TGBLA 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 homogenized cross-sections. TGBLA also performs burn up calculations for generating input to the BWR 3-D simulator. In addition, TGBLA generates rod-by-rod neutron cross-sections, gamma smeared power distributions and flux discontinuity factors.

References

(1) STEADY-STATE NUCLEAR METHOD, General Electric, NEDO-30130-A, May 1985.

6.3. PANACEA

The BWR Core Simulator (PANACEA) is a static, three-dimensional coupled nuclear-thermal-hydraulic computer programme representing the BWR core exclusive of the external flow loop. Provisions are made for fuel cycle and thermal limits calculations. 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. A special power-exposure iteration option is available for target exposure distribution and cycle length predictions. PANACEA includes the effect of Doppler broadening as a function of moderator density, exposure, control and moderator density history for a given fuel type.

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.

Neutronic parameters used by PANACEA are obtained from the 2-D lattice physics code (TGBLA) and parametrically fitted as a function of moderator density, exposure, control and moderator density history for a given fuel type.

PANACEA contains an improved 1-1/2 group (quasi-two group) physics model which accounts for spectral history and control blade history reactivity corrections. Control blade history local peaking effects are also considered in the new model. A pin power reconstruction model is incorporated to account for the effect of flux gradients across the nodes on the local peaking distribution. The improved 1.5 group model has been benchmarked against 3 groups, fine-mesh diffusion. Both eigenvalue and power distribution results from the improved 1.5 group physics model compare very well against the more detailed models. The model has also been qualified by examination of hot eigenvalues, cold eigenvalues, and Transverse In-core Probe (TIP) predictions for over 50 plant cycles for a wide variety of plant and loading pattern configurations.

References

(1) STEADY-STATE NUCLEAR METHOD, General Electric, NEDO-30130-A, May 1985.

6.4. ODYSY

ODYSY (One-Dimensional Dynamic Code for Stability) is a best-estimate Engineering Computer Programme (ECP) that incorporates a linearized, small perturbation, frequency domain model of the reactor core and associated coolant circulation system. The programme may be used to predict hydrodynamic stability for both a single channel and a full reactor core. It will predict both core-wide mode coupled thermal-hydraulic and reactor kinetic instabilities and single channel thermal-hydraulic instabilities. ODYSY is based on transient model, including an axial one-dimensional (1-D) kinetics model extended to multiple channels. It has axially varying void and Doppler reactivity feedback with improved flexibility in the fuel rod modelling to accommodate axial variations in fuel bundle geometry. The axial variation capability makes it ideal for evaluating the stability performance of advanced fuel designs that have axially varying geometry.