4.4 Thermal–Hydraulic Design

4.4.1 Design Basis

4.4.1.1 Safety Design Bases

Thermal-hydraulic design of the core shall establish the thermal-hydraulic safety limits for use in evaluating the safety margin relating the consequences of fuel cladding failure to public safety.

4.4.1.2 Requirements for Steady-State Conditions

For purposes of maintaining adequate fuel performance margin during normal steady-state operation, the MCPR must not be less than the required MCPR operating limit, and the APLHGR must be maintained below the required APLHGR limit (MAPLHGR). The steady-state MCPR and MAPLHGR limits are determined by analyses of the most severe moderate frequency anticipated operational occurrences (AOOs) to accommodate uncertainties and provide reasonable assurance that no fuel damage results during moderate frequency AOOs at any time in life. These limits are provided in the Technical Specifications.

4.4.1.3 Requirements for Anticipated Operational Occurrences (AOOs)

The MCPR and MAPLHGR limits are established such that no safety limit is expected to be exceeded during the most severe moderate frequency AOO event as defined in Chapter 15.

4.4.1.4 Summary of Design Bases

In summary, the steady-state operating limits have been established to assure that the design bases are satisfied for the most severe moderate frequency AOO. Demonstration that the steady-state MCPR and MAPLHGR limits are not exceeded is sufficient to conclude that the design bases are satisfied.

4.4.2 Description of Thermal-Hydraulic Design of the Reactor Core

4.4.2.1 Summary Comparison

Typical thermal-hydraulic parameters for the ABWR are compared to those for a typical BWR/6 plant in Table 4.4-1. Data for BWR/6 plant are documented in Reference 4.4-19.

4.4.2.2 Critical Power Ratio

A description of the critical power ratio is provided in Subsection 4.4.5.1. Criteria used to calculate the critical power ratio safety limit are given in Appendix 4B.

4.4.2.3 Average Planar Linear Heat Generation Rate (APLHGR)

Models used to calculate the APLHGR limit are given in Subsection 4.2.3.1.1 as pertaining to the fuel mechanical design limits, and in Subsection 6.3.3.7 as pertaining to 10CFR50 Appendix K limits.

4.4.2.4 Void Fraction Distribution

The axial distribution of core void fractions for the average radial channel and the maximum radial channel (end of node value) for the reference core loading pattern (Figure 4.3-1) are given in Table 4.4-2. The core average and maximum exit value is also provided. Similar distributions for steam quality are given in Table 4.4-3. The core average axial power distribution used to produce these tables is given in Table 4.4-4.

4.4.2.5 Core Coolant Flow Distribution and Orificing Pattern

The flow distribution to the fuel assemblies and bypass flow paths is 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 4.4-1, 4.4-2, and 4.4-3). The components of bundle pressure drop considered are friction, local, elevation, and acceleration (Subsections 4.4.2.6.1 through 4.4.2.6.4, 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 minus the control rod cooling flow enters the lower plenum. A small fraction of this flow passes through various bypass flow paths. The remainder passes through the orifice in the fuel support pieces (experiencing a pressure loss) where a small fraction of the flow exits into the bypass flow region while the rest continues into the transition piece. All initial and reload core fuel bundles have holes in the transition piece through which some flow passes to the bypass flow region. Some flow passes through the hole in the bottom support into the central water channel. The majority of the flow from the transition piece continues through the bottom tie plate (experiencing a pressure loss). A small fraction of the flow above the bottom tie plate goes to the cross wings while the remainder goes to the active fuel.

Within the fuel assembly, heat balances on the active coolant are performed nodally. Fluid properties are expressed as the node average at the particular node of interest and are based on 1979 steam tables from Schmidt (Reference 4.4-17). In evaluating fluid properties a constant pressure model is used.

The relative radial and axial power distributions documented in the country-specific supplement 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.

4.4.2.6 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.

4.4.2.6.1 Friction Pressure Drop

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

$$\Delta P_f = \frac{w^2}{2g_c\rho}\frac{fL}{D_H A_{ch}^2}\phi_{TPF}^2 \tag{4.4-1}$$

where

ΔP_f	=	friction pressure drop
w	=	mass flow rate
g_c	=	conversion constant
ρ	=	average nodal liquid density
D_H	=	channel hydraulic diameter
A_{ch}	=	channel flow area
L	=	incremental length
f	=	friction factor
ϕ_{TPF}	=	two-phase friction multiplier

The formulation for the two-phase multiplier is similar to that presented in References 4.4-4 and 4.4-5.

4.4.2.6.2 Local Pressure Drop

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

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

$$\Delta P_L = \frac{w^2}{2g\rho} \frac{K}{A^2} \phi_{TPL}^2 \tag{4.4-2}$$

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 defined above. The formulation for the two-phase multiplier is similar to that reported in Reference 4.4-5.

4.4.2.6.3 Elevation Pressure Drop

The elevation pressure drop is based on the relation:

$$\Delta P_E = \bar{\rho}\Delta L; \quad \bar{\rho} = \rho_f(1-\alpha) + \rho_g\alpha \tag{4.4-3}$$

where

ΔP_E = elevation pressure drop

ΔL = incremental length

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

α = nodal average void fraction

ρ_f, ρ_g = saturated water and vapor density, respectively

The void fraction model used is an extension of the Zuber-Findlay model (Reference 4.4-6).

4.4.2.6.4 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:

$$\Delta P_{ACC} = (1-\sigma_A^2)\frac{W^2}{2g_c\rho_1 A_2^2}$$
$$\sigma_A = \frac{A_2}{A_1} = \frac{\text{final flow area}}{\text{initial flow area}} \tag{4.4-4}$$

where

ΔP_{ACC} = acceleration pressure drop

A_2 = final flow area

A_1 = initial flow area

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:

$$\Delta P_{ACC} = (1-\sigma_A^2)\frac{W^2\rho_H}{2g_c\rho^2{}_{KE}A_2^2} \tag{4.4-5}$$

where

$\frac{1}{\rho_H} = \frac{x}{\rho_g} + \frac{(1-x)}{\rho_l}$, homogeneous density

$$\frac{1}{\rho_{KE}^2} = \frac{x^3}{\rho_g^2\alpha^2} + \frac{(1-x)^3}{\rho_l^2(1-\alpha)^2}, \tag{4.4-6}$$

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:

$$\Delta P_{ACC} = \frac{W^2}{g_c A_{ch}^2}\left(\frac{1}{\rho_{out}} - \frac{1}{\rho_{in}}\right) \tag{4.4-7}$$

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

$$\frac{1}{\rho_M} = \frac{x^2}{\rho_g \alpha} + \frac{(1-x)^2}{\rho_l (1-\alpha)} \tag{4.4-8}$$

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.

4.4.2.7 Correlation and Physical Data

Substantial amounts of physical data have been obtained in support of the pressure drop and thermal-hydraulic loads discussed in Subsection 4.4.2.6. Correlations have been developed to fit these data to the formulations discussed.

4.4.2.7.1 Pressure Drop Correlations

Significant amounts of friction pressure drop data in multi-rod geometries representative of BWR plant fuel bundles has been collected. This data has been used to correlate both the friction factor and two-phase multipliers on a best fit basis using the pressure drop formulations reported in Subsection 4.4.2.6.1 and 4.4.2.6.2. Tests are 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 tieplate pressure drop. The range of test variables is specified to include the range of test variables is specified to include the range of interest to 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 4.4.2.6.1 and 4.4.2.6.2 for the fuel designs described in Subsection 4.2.2, was confirmed by full scale prototype flow tests.

4.4.2.7.2 Void Fraction Correlation

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

4.4.2.7.3 Heat Transfer Correlation

The Jens-Lottes (Reference 4.4-7) heat transfer correlation is used in fuel design to determine the cladding-to-coolant heat transfer coefficients for nucleate boiling.

4.4.2.8 Thermal Effects of Anticipated Operational Occurrences

The evaluation of the core's capability to withstand the thermal effects resulting from anticipated operational occurrences is covered in Chapter 15.

4.4.2.9 Uncertainties in Estimates

Uncertainties in thermal-hydraulic parameters are considered in the statistical analysis which is performed to establish the fuel cladding integrity safety limit documented in Subsection 4.4.5.5.

4.4.2.10 Flux Tilt Considerations

For flux tilt considerations, refer to Subsection 4.3.2.2.

4.4.3 Description of the Thermal–Hydraulic Design of the Reactor Coolant System

4.4.3.1 Plant Configuration Data

4.4.3.1.1 Reactor Coolant System Configuration

The Reactor Coolant System is described in Section 5.4.

4.4.3.1.2 Reactor Coolant System Thermal–Hydraulic Data

The steady-state distribution of temperature, pressure and flow rate for each flow path in the Reactor Coolant System is shown in Figure 5.1-1.

4.4.3.1.3 Reactor Coolant System Geometric Data

Volumes of regions and components within the reactor vessel are shown in Figure 5.1-2.

Table 4.4-5 provides the flow path length, height, liquid level, minimum elevations, and minimum flow areas for each major flow path volume within the reactor vessel.

4.4.3.2 Operating Restrictions on Pumps

Expected recirculation pump performance curves are shown in Figure 5.4-3. These curves are valid for all conditions with a normal operation range varying from approximately 20% to 115% of rated pump flow.

Subsection 4.4.3.3 gives the operating limits imposed on the recirculation pumps by cavitation, pump load, bearing design flow starvation, pump speed, and steam separator performance.

It is required that at least 9 out of 10 RIPs are operating for normal operation. For operation with less than 9 RIPs in operation, the COL applicant will provide the necessary supporting analyses.

4.4.3.3 Power/Flow Operating Map

4.4.3.3.1 Limits for Normal Operation

The ABWR must operate with certain restrictions because of pump net positive suction head (NPSH), overall plant control characteristics, core thermal power limits, etc. The power-flow map for 10 RIP operation is shown in Figure 4.4-1, and for 9 RIP operation in Figure 4.4-2. Those power-flow maps for the power range of operation shown were used in the system

response analyses documented in Section 6.3 and Chapter 15. See Subsections 4.4.7.1 and 4.4.7.2 for COL license information. The nuclear system equipment, nuclear instrumentation, and the Reactor Protection System, in conjunction with operating procedures, maintain operations within the area of the operating map for normal operating conditions. The boundaries on this map are as follows:

- **Natural Circulation Line, 0:** The operating state of the reactor moves along this line for the normal control rod withdrawal sequence in the absence of recirculation pump operation.
- **102% 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.
- **Steam Separator Limit Line:** This line results from the requirements to have acceptable moisture carryover fraction from the steam separator.

The COL applicant will provide the specific power/flow operating map and thermal limits for the core loading to the NRC for information (Subsection 4.4.7.1).

4.4.3.3.2 Not Used

4.4.3.3.3 Regions of the Power/Flow Map

Region I	This region defines the system operational capability with the reactor internal pumps running at their minimum speed (31%). Power changes, during normal startup and shutdown, will be in this region.
Region II	This is the low power area of the operating map where the carryover through steam separators is expected to exceed the acceptable value. Operation within this region is precluded by system interlocks.
Region III	This is the high-power/low-flow area of the operating map in which the system is the least damped. Operation within this region is precluded by SCRRI (Selected Control Rods Run-In).
Region IV	This represents the normal operating zone of the map where power changes can be made, by either control rod movement or by core flow changes, through the change of the pump speeds.

4.4.3.3.4 Design Features for Power/Flow Control

The following limits and design features are employed to maintain power/flow conditions shown in Figure 4.4-1:

(1) **Minimum Power Limits at Intermediate and High Core Flows:** To prevent unacceptable separator performance, the recirculation system is provided with an interlock to reduce the RIP speed.

(2) **Pump Minimum Speed Limit:** The reactor internal pumps (RIPs) are equipped with anti-rotation devices (ARD) which prevent a tripped RIP from rotating backwards. The ARD begins operating at 42.4 rad/s decreasing speed. In order to prevent mechanical wear in the ARD, minimum speed is specified at 42.4 rad/s. However, to provide a stable operation, the minimum pump speed is set at 47.1 rad/s (31% of required).

4.4.3.3.5 Flow Control

The normal plant startup procedure requires the startup of all RIPs first and maintain at their minimum pump speed (31% of rated), at which point reactor heatup and pressurization can commence. When operating pressure has been established, reactor power can be increased. This power/flow increase will follow a line within Region I of the flow control map shown in Figure 4.4-1. The system is then brought to the desired power/flow level within the normal operating area of the map (Region IV) 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. Change of pump speeds with constant control rod position will result in power/flow changes along, or nearly parallel to, the rated flow control line.

4.4.3.4 Temperature-Power Operating Map

Not applicable.

4.4.3.5 Load-Following Characteristics

Not Used.

4.4.3.6 Thermal–Hydraulic Characteristics Summary Table

The thermal-hydraulic characteristics are provided in Table 4.4-1 for the core and tables of Section 5.4 for other portions of the Reactor Coolant System.

4.4.3.7 Thermal Hydraulic Stability Performance

Thermal Hydraulic Stability is discussed in Section 15.9.

4.4.4 Loose-Parts Monitoring System

The Loose-Parts Monitoring System (LPMS) is designed to provide detection of loose metallic parts within the reactor pressure vessel. The LPMS detects structure-borne sound that can indicate the presence of loose parts impacting against the reactor pressure vessel or its internals. The LPM detection system can evaluate some aspects of selected signals. However, the system, by itself, cannot diagnose the presence and location of a loose part.

4.4.4.1 Power Generation Design Bases

The LPMS is designed to provide detection and operator warning of loose parts in the RPV to avoid or mitigate damage to or malfunctions of reactor components.

Additional design considerations provide for the inclusion of electronic features to minimize operator interfacing requirements during normal LPMS operation. These electronic features also enhance the LPMS detection and analysis function when operator action is required to investigate potential loose parts.

4.4.4.2 System Description

The LPMS monitors the RPV for indications of loose parts. The alarm setting for each sensor is determined after system installation is complete. The alarm setting is set to discriminate between normal background noises and any actual loose part impact signal to minimize spurious alarms. Each sensor channel is isolated to reduce the possibility of signal ground loop problems and to minimize sensor signal background noise. Background noises are also minimized by use of tuned filters. A disable signal is provided during control rod movement and other plant maneuvers that may initiate a LPM alert-level alarm.

LPMS sensors are usually accelerometers. The array of LPMS accelerometers typically consist of a set of sensors strategically mounted on the external surface of the primary pressure vessel boundary at various elevations and azimuths at natural collection regions for potential loose parts. General mounting locations are at the (1) main steam outlet nozzle, (2) feedwater inlet nozzle, (3) flooded nozzles, and (4) control rod drive housings. The sensors are mounted in such a fashion as to provide good response and sensitivity, even at high frequency.

The online system sensitivity is such that the system can detect a metallic loose part that has a mass between 0.11 kg to 13.6 kg and impacts with kinetic energy of 0.68 joules or more on the inside surface of the RPV within 0.91m of a sensor. The loose parts impact frequency range of interest is typically from 1 to 10 kHz. Frequencies lower than 1 kHz are generally associated with flow-induced vibration (FIV) signals or flow noise.

Physical separation is maintained between the sensor signal cables at each natural collection region to an area where they are grouped and routed through several cable penetrations to a termination area. The termination area is accessible by maintenance personnel during full power operation.

The LPMS includes provisions for both automatic and manual startup of data acquisition equipment with automatic activation in the event that the preset alert level is reached or exceeded. The system also initiates an alarm to the control room personnel when an alert condition is reached. The Data Acquisition System will automatically select the alarmed signal sensor channel plus additional channels for simultaneous recording. The signal analysis equipment will allow immediate visual and audio monitoring of all signals.

Provisions exist for periodic online sensor signal channel check and functional tests and for offline channel calibration during periods of cold shutdown or refueling. The LPMS electronics are designed to facilitate the recognition, location, replacement, repair, and adjustment of malfunctioning LPMS electronic components. The LPMS electronic components located inside the containment have been designed and installed to perform their function following all seismic events that do not require plant shutdown. The plant will shutdown when the recorded motion exceeds the limits specified in Subsection 3.7.4.4. The LPMS electronic components selected for this application are rated to meet the normal operating radiation, vibration, temperature, and humidity environments in which the components are installed.

All LPMS electronic components within the containment are designed for a 60-year design life. In those instances where a 60-year design life is not practicable, a replacement program will be established for those parts that are anticipated to have limited service life.

4.4.4.3 Normal System Operation

The LPMS will be set to alarm for detected signals having characteristics of metal-to-metal impacts.

After installation of the sensor array, the LPMS overall and individual sensor signal channels can be characterized at plant startup before operation monitoring. Each accelerometer channel will exhibit its own particular signature and unique corresponding frequency spectrum. This signature and spectrum results from a combination of both internal and external sources due to normal and transient conditions.

Calibration is an important part of LPMS operation. The alarm level setpoint is determined by using a manual calibration device to simulate the presence of a loose part impact near each sensor. The setpoint is typically based on a percentage of the calibration signal magnitude, and is a function of actual background noise. Additionally, calibrated impacts at various locations near the sensors assist in diagnosing the source of the signal. (e.g LPM sensor signals disabled).

Discrimination logic is typically incorporated in the LPMS to avoid spurious alarms. Discrimination logic rejects events that do not have the characteristics of an impact signal of a loose part. Typical discrimination functions are based on the length of time the signal is above the setpoint, the number of channels alarming, the time between alarms, the repetition of the signal, and the signal waveform and frequency content. False alert signals due to plant maneuvers are avoided by the use of administrative procedures by control room personnel.

Usually, the plant operator makes the preliminary evaluation based on the available information. If the presence of unusual metal impact sound is indicated, then the station engineers perform additional evaluations. LPM experts are required to correctly diagnose the presence and location of a loose part. In order to reach proper conclusions, various factors must be considered such as: plant operating conditions, location of the sensors that alarmed, and comparison of the amplitude and frequency contents of these sensor signals with known normal LPMS operation sensor signal data.

4.4.4.4 Safety Evaluation

The LPMS is intended to be used for information purposes only by the plant operator. The plant operators do not rely on the information provided by the LPMS for the performance of any safety-related action. However, although the LPMS is not classified as a safety-related system, it is designed to meet the seismic and environmental operability recommendations of Regulatory Guide 1.133.

4.4.4.5 Test and Inspection

The LPMS will be calibrated to detect a metallic loose part that has a mass from 0.11 kg to 13.6 kg and impacts with kinetic energy of 0.68 joules within 0.91m of each sensor. Provisions will be made to check the LPM system calibration at the LPMS data acquisition and analysis console at each refueling. The system will be recalibrated as necessary when found to be out of calibration. A test and reset capability will be included for functional test capability.

The manufacturer will provide services of qualified personnel to provide technical guidance for installation, startup, and acceptance testing of the system. In addition, the manufacturer will provide the necessary training of plant personnel for proper system operation and maintenance and planned operating and record-keeping procedures.

4.4.4.6 Instrumentation Application

The LPMS consists of sensors, cables, signal conditioning equipment, alarming monitor, signal analysis and data acquisition equipment, and calibration equipment.

4.4.5 Evaluation

4.4.5.1 Critical Power

The objective for normal and AOOs is to maintain nucleate boiling and thus avoid a transition to film boiling. Operating limits are specified to maintain adequate margin to the onset of the boiling transition. 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 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 exists 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:

> Moderate frequency AOOs caused by a single operator error or equipment malfunction shall be limited such that, considering uncertainties in manufacturing and monitoring the core operating state, at least 99.9% of the fuel rods would be expected to avoid boiling transition (References 4.4-8 and 4.4-18).

Both the transient and normal operating thermal limits in terms of MCPR are derived from this basis.

4.4.5.2 Core Hydraulics

Core hydraulics models and correlations are discussed in Subsection 4.4.2.

4.4.5.3 Influence of Power Distributions

The influence of power distributions on the thermal-hydraulic design is discussed in References 4.4-8, 4.4-9, and 4.4-18.

4.4.5.4 Core Thermal Response

The thermal response of the core for accidents and expected AOO conditions is given Chapter 15.

4.4.5.5 Analytical Methods

4.4.5.5.1 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 AOOs, but is also applied to the localized rod withdrawal error AOO. The safety limit MCPR is derived based on the criteria of Appendix 4B.

4.4.5.5.1.1 Statistical Model

The statistical analysis utilizes a model of the ABWR core which simulates the relevant core conditions. This code produces a critical power ratio (CPR) map of the core based on inputs of power distribution, flow and heat balance information. Details of the procedure are documented in References 4.4-8 and 4.4-18. Random Monte Carlo selections of all operating parameters based on the uncertainty ranges of manufacturing tolerances, uncertainties in measurement of core operating parameters, calculational 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.

4.4.5.5.1.2 Bounding BWR Statistical Analysis

Statistical analyses have been performed which provide fuel cladding integrity safety limit MCPRs applicable to the Westinghouse fuel designs in the ABWR reload and initial core cycles. These safety limit MCPRs were derived based on the criteria in Appendix 4B. 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 1.11 for the initial core, and 1.09 for reload cores.

4.4.5.5.2 MCPR Operating Limit Calculational

A plant-unique MCPR operating limit is established to provide adequate assurance that the fuel cladding integrity safety limit for that plant is not exceeded for any moderate frequency AOO. This operating requirement is obtained by addition of the maximum ΔCPR value for the most limiting AOO (including any imposed adjustment factors) from conditions postulated to occur at the plant to the fuel cladding integrity safety limit.

4.4.5.5.3 Calculational Procedure for AOO Pressurization Events

Core-wide rapid pressurization events are analyzed using the evaluation models described in Section 15.0.4.4.1.

4.4.5.5.4 Calculational Procedure for AOO Slow Events

Core-wide non-pressurization AOOs are analyzed using the evaluation models described in Section 15.0.4.4.1.

4.4.6 Testing and Verification

The testing and verification techniques to be used to assure that the planned thermal and hydraulic design characteristics of the core have been provided, and will remain within required limits throughout core lifetime, are discussed in Chapter 14.

An analysis is performed to determine the required cooling for a fuel assembly post-LOCA. This analysis is discussed in Appendix 6C and is used to develop acceptance criteria for a downstream fuel effects test performed prior to initial cycle operation.

4.4.7 COL License Information

4.4.7.1 Power/Flow Operating Map

The COL applicant will provide the specific power/flow operating map to be used at the plant to the USNRC for information (Subsection 4.4.3.3.1).

4.4.7.2 Thermal Limits

The COL applicant will provide the thermal limits for the core loading at the plant to the USNRC for information (Subsection 4.4.3.3.1).

4.4.8 References

4.4-1 “Core Flow Distribution in a Modern Boiling Water Reactor as Measured in Monticello”, NEDO-10299A, October 1976.

4.4-2 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.

4.4-3 “Brunswick Steam Electric Plant Unit 1 Safety Analysis Report for Plant Modifications to Eliminate Significant In-Core Vibrations”, NEDO-21215, March 1976.

4.4-4 R.C. Martinelli and D.E. Nelson, “Prediction of Pressure Drops During Forced Convection Boiling of Water”, ASME Trans., 70, 695-702, 1948.

4.4-5 C.J. Baroozy, “A Systematic Correlation for Two-Phase Pressure Drop”, Heat Transfer Conference (Los Angeles), AICLE, Preprint No. 37, 1966.

4.4-6 N. Zuber and J.A. Findlay, “Average Volumetric Concentration in Two-Phase Flow Systems”, Transactions of the ASME Journal of Heat Transfer, November 1965.

4.4-7 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.

4.4-8 “Reference Safety Report for Boiling Water Reactor Reload Fuel”, (CENPD-300-P-A, July 1996).

4.4-9 “10x10 SVEA Fuel Critical Power Experiments and CPR Correlations: SVEA-96 Optima2”, WCAP-16081-P-A, March 2005.

4.4-10 Not Used

4.4-11 Not Used

4.4-12 Not Used

4.4-13 Not Used

4.4-14 Not Used

4.4-15 Not Used

4.4-16 Not Used

4.4-17 Steam tables from Schmidt, 1979, in cooperation between the American Society of Mechanical Engineers, New York, the Japan Society of Mechanical Engineers, Tokyo, and the Verein Deutscher Ingenieure, Düsseldorf, 3-540-09601-9 and 0-387-09601-9.

4.4-18 "Reference Safety Report for Boiling Water Reactor Fuel and Core Analyses Supplement 1 to CENPD-300-P-A," WCAP 17322-P, September 2010.

4.4-19 P. Jourdain, E. Liman, et al., "DCDRA Tier 2 Table 4.3-2 and 4.4-1 - 4.4-4", BTC 10-1757, December 2010.

Table 4.4-1 Typical Thermal–Hydraulic Design Characteristics of the Reactor Core

General Operating Conditions	BWR/6 218-624	ABWR* 278-872
Reference design thermal output (MWt)	3250	3926
Power level for engineered safety features (MWt)	3315	4005
Steam flow rate, at final feedwater temperature (Mlb/hr)	14.280	16.857
Core coolant flow rate (Mlb/hr)	88.7**	115.1***
Feedwater flow rate (Mlb/hr)	14.250	16.819
Feedwater temperature (°F)	432	420
System pressure, nominal in steam dome (psia)	1070	1040
Coolant saturation temperature at core design pressure (°F)	554.0	549.5
Average power density (kW/L)	58.9	49.2
Core total heat transfer area (ft^2)	71,731	100,244
Core inlet enthalpy (Btu/lb)	531.6	527.5
Core inlet temperature (°F)	536	533
Core maximum exit voids within assemblies (%)	76.0	86.0
Core average void fraction, active coolant	0.50	0.35
Active coolant flow area per assembly ($in.^2$)	14.634	14.634
Total core pressure drop (psi)	25.5	22.9
Core support plate pressure drop (psi)	20.7	18.3
Average orifice pressure drop, central region (psi)	9.57	8.99
Average orifice pressure drop, peripheral region (psi)	19.44	16.68
Maximum channel pressure loading (psi)	11.12	9.94
Average-power assembly channel pressure loading (bottom) (psi)	9.11	6.92

* Based on the core loading in Figure 4.3-1.

** 105% core flow.

*** 100% core flow (EOC conditions).

Table 4.4-2 Void Distribution for Analyzed Core

Core Average Value - 0.352
Maximum Exit Value - 0.860
Active Fuel Length - 3.81 m

	Node	Core Average (Average Node Value)	Maximum Channel (Average Node Value)
Bottom of Core	1	0	0
	2	0	0
	3	0.003	0.031
	4	0.019	0.118
	5	0.055	0.232
	6	0.105	0.340
	7	0.160	0.431
	8	0.216	0.505
	9	0.265	0.563
	10	0.309	0.609
	11	0.347	0.647
	12	0.380	0.678
	13	0.411	0.705
	14	0.438	0.731
	15	0.463	0.752
	16	0.485	0.771
	17	0.506	0.788
	18	0.524	0.802
	19	0.541	0.816
	20	0.558	0.827
	21	0.574	0.837
	22	0.590	0.845
	23	0.604	0.852
	24	0.617	0.857
Top of Core	25	0.625	0.860

Table 4.4-3 Flow Quality Distribution for Analyzed Core

Core Average Value - 0.064
Maximum Exit Value - 0.397
Active Fuel Length - 3.81 m

	Node	Core Average (Average Node Value)	Maximum Channel (Average Node Value)
Bottom of Core	1	0	0
	2	0	0
	3	0	0
	4	0	0
	5	0	0.013
	6	0	0.030
	7	0.003	0.047
	8	0.012	0.065
	9	0.021	0.083
	10	0.030	0.102
	11	0.040	0.121
	12	0.050	0.141
	13	0.060	0.161
	14	0.070	0.182
	15	0.081	0.203
	16	0.091	0.225
	17	0.102	0.247
	18	0.112	0.269
	19	0.122	0.291
	20	0.133	0.313
	21	0.143	0.335
	22	0.154	0.356
	23	0.163	0.374
	24	0.172	0.389
Top of Core	25	0.177	0.397

Table 4.4-4 Axial Power Distribution Used to Generate Void and Quality Distributions for Analyzed Core

	Node	Axial Power Factor Core Average	Axial Power Factor Maximum Channel
Bottom of Core	1	0.17	0.29
	2	0.56	1.02
	3	0.73	1.45
	4	0.81	1.62
	5	0.88	1.71
	6	0.95	1.77
	7	1.02	1.84
	8	1.08	1.90
	9	1.08	1.91
	10	1.10	1.95
	11	1.13	2.00
	12	1.16	2.05
	13	1.19	2.13
	14	1.22	2.19
	15	1.22	2.22
	16	1.24	2.27
	17	1.22	2.29
	18	1.18	2.26
	19	1.21	2.29
	20	1.22	2.28
	21	1.22	2.20
	22	1.20	2.07
	23	1.08	1.74
	24	0.83	1.26
Top of Core	25	0.31	0.49

Table 4.4-5 Reactor Coolant System Geometric Data

		Flow Path Length (m)	Height and Liquid Level (m)	Elevation of Bottom of Each Volume* (m)	Average Flow Areas (m^2)
A.	Lower Plenum	4.65	4.65 4.65	0.0	19.5
B.	Core	4.36	4.36 4.36	4.65	16.1† includes bypass
C.	Upper Plenum and Separators	3.64	3.64 3.64	9.00	16.5
D.	Dome (Above Normal Water Level)	7.80	7.80 0	13.2	30.2
E.	Downcomer Area	12.6	12.6 12.6	1.84	16.2

* Reference point is vessel bottom zero.

† For the core loading given in Figure 4.3-1.

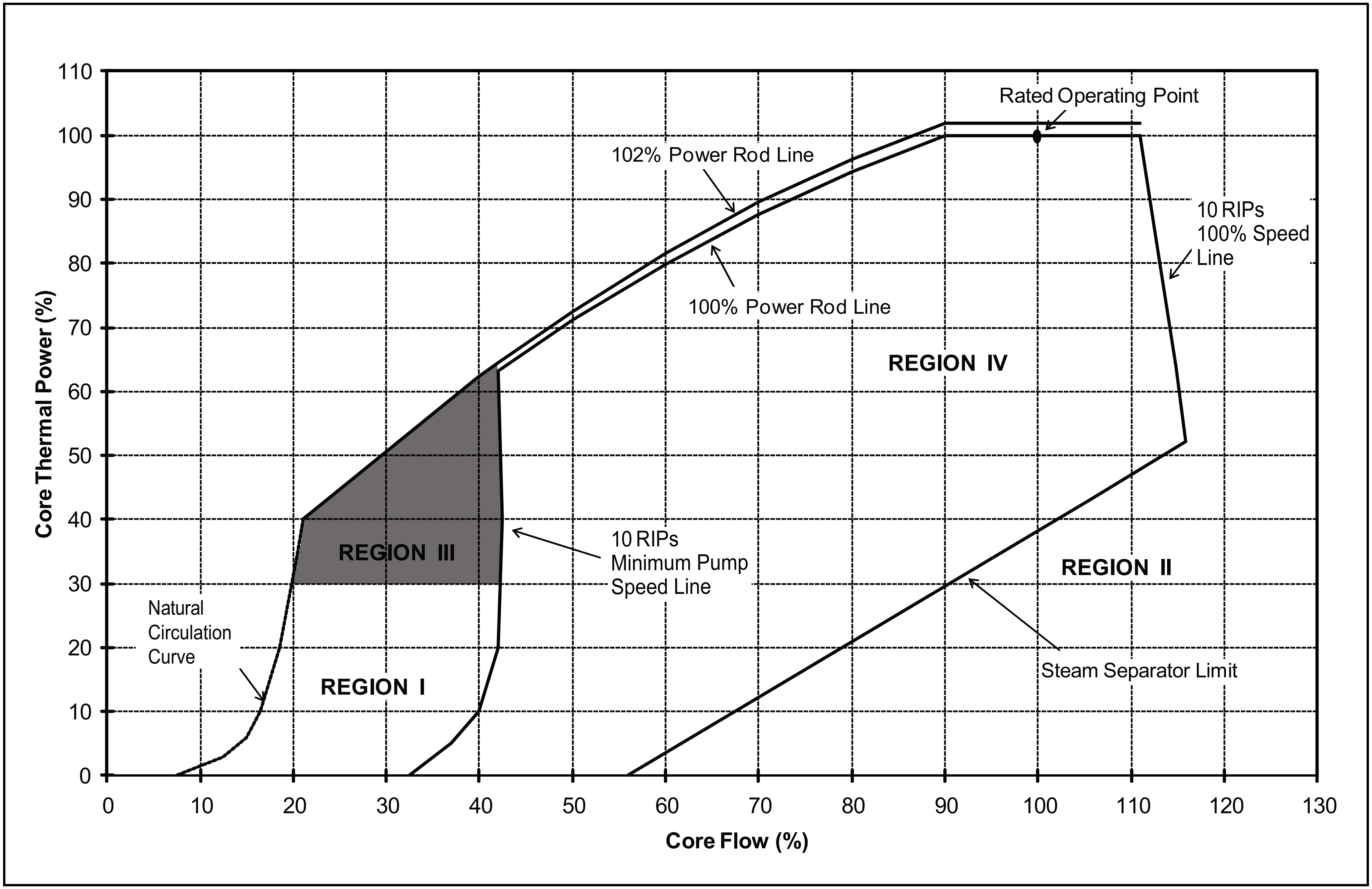

Figure 4.4-1 Power-Flow Operating Map Used for System Response Study

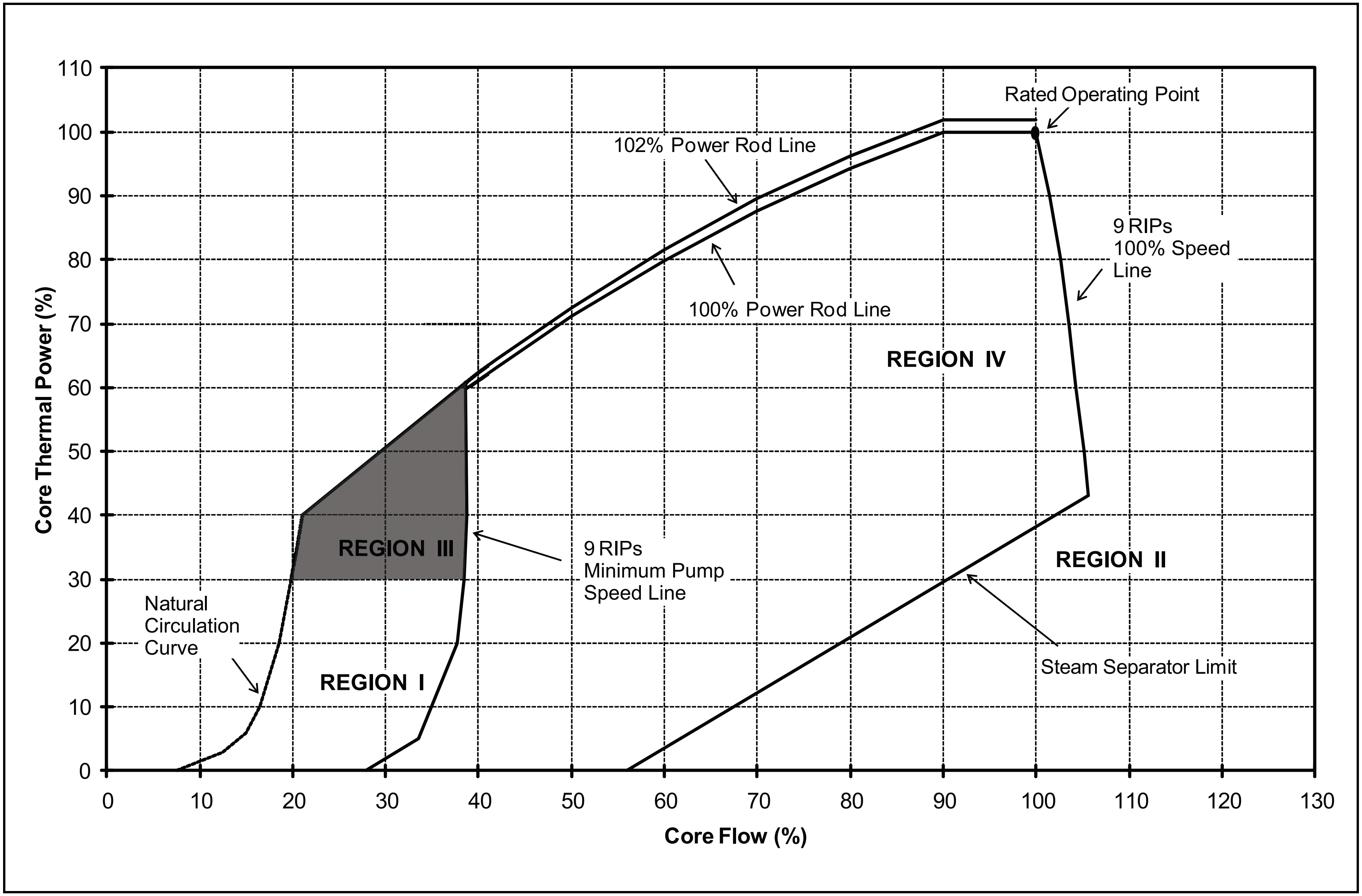

Figure 4.4-2 Power-Flow Operating Map Used for System Response Study (9 RIPs Operation)

Figure 4.4-3 Not Used

Figure 4.4-4 Not Used