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 minimum critical power ratio (MCPR) must not be less than the required MCPR operating limit, and the average planar linear heat generation rate (APLHGR) must be maintained below the required maximum average planar linear heat generation rate (MAPLHGR) limit. The steady-state MCPR limit is 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. The MAPLHGR limits are a composite of two independent sets of local power limits; the LOCA limits, intended to assure compliance with 10CFR50.46 during a LOCA, and the thermal mechanical limits, intended to preclude violation of any of the Specified Acceptable Fuel Design Limits (SAFDLs). The composite MAPLHGR is determined by selecting the most limiting of the two over the applicable exposure range.

The MCPR and MAPLHGR 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

The thermal-hydraulic parameters for the Lungmen Nuclear Power Station (NPS) are compared to those for a typical ABWR plant in Table 4.4-1.

4.4.2.2 Critical Power Ratio

The critical power ratio (CPR) is the figure of merit used to express a thermal margin to the onset of boiling transition. This is defined as the ratio of the critical power (bundle power at which some point within the bundle experiences onset of boiling transition) to the operating bundle power. The thermal margin is stated in terms of the minimum CPR (MCPR), which corresponds to the most limiting fuel assembly in the core. To assure that safety limit MCPR is not exceeded during the most limiting AOO, the MCPR should be maintained above the operating limit MCPR during a steady state plant operation. Further description of the CPR is provided in Section 6.0 of Reference 4.4-1. See also Subsection 4.4.5.1.

4.4.2.3 Average Planar Linear Heat Generation Rate (APLHGR)

The APLHGR is the average linear heat generation rate (expressed in kw/ft) in any plane of a fuel bundle and is obtained by averaging LHGR over each fuel rod in the plane. The limiting value of APLHGR in the core is the maximum APLHGR (MAPLHGR). There are two independent MAPLHGR limits; the LOCA limits and the fuel rod thermal mechanical limits. The SAFER/GESTR-LOCA ECCS evaluation methodology is used to calculate MAPLHGR limits pertaining to ECCS performance analysis requirements of 10CFR50.46 for the postulated design basis LOCA. A detailed description of the LOCA MAPLHGR calculational procedure and models is provided in Section 7.0 of Reference 4.4-1. The GESTR-MECHNICAL fuel rod thermal-mechanical performance model is used to perform the fuel rod thermal-mechanical design and safety evaluations to determine the fuel rod thermal mechanical MAPLHGR limits. A detailed description of the model is provided in Section 3.0 of Reference 4.4-1.

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 values are 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 4.4-2, 4.4-3, and 4.4-4). The components of bundle pressure drop considered are friction, local, elevation, and acceleration (Subsection 4.4.2.6). 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 exists through the fit-up between the fuel support and the lower tieplate and through the lower tieplate holes into the bypass flow region. All initial and reload core fuel bundles have lower tieplate holes. The majority of the flow continues through the lower tieplate (experiencing a pressure loss) where some flow exists through the flow path defined by the fuel channel and lower tieplate into the bypass region. This bypass flow is lower for those fuel assemblies with finger springs.

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 Table, American Society of Mechanical Engineers, New York, 1989. 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. These pressure drops are calculated with basic models similar to those used throughout the nuclear power industry. The models used for each of these pressure drop components are described in Subsections 5.1- through 5.4 in Reference 4.4-1. Pressure drop measurements made in operating reactors confirm that the total measured core pressure drop and calculated core pressure drop are in good agreement.

4.4.2.7 Correlation and Physical Data

General Electric Company has obtained substantial amounts of physical data 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

General Electric Company has taken significant amounts of friction pressure drop data in multi-rod geometry's representative of BWR plant fuel bundles and correlated both the friction factor and two-phase multipliers on a best fit basis using the pressure drop formulations described in Subsections 5.1 through 5.4 in Reference 4.4-1. Tests are performed in single-phase water to calibrate the orifice and the lower tie plate, and in both single-and two-phase flows to arrive at best fit design values for spacer and upper tie plate pressure drop. 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.

4.4.2.7.2 Void Fraction Correlation

The void fraction correlation (Reference 4.4-8) includes effects of pressure, flow direction, mass velocity, quality, and subcooled boiling.

4.4.2.7.3 Heat Transfer Correlation

The Jens-Lottes (Reference 4.4-5) 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 4.4.4.1.

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 (RCS) includes those systems and components which contain or transport fluids coming from or going to the reactor core. These systems form a major portion of the reactor coolant pressure boundary (RCPB). The RCS is described in detail in Sections 5.1 and 5.4.

4.4.3.1.2 Reactor Coolant System Thermal–Hydraulic Data

The steady-state distribution at rated operating conditions for the Lungmen ABWR of temperature, pressure and flow rate for each flow path in the RCS 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 and recirculation loops of the RCS.

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 reactor internal pumps (RIPs) are operating for normal operation. For operation with less than 9 RIPs in operation, the necessary supporting analyses will be documented in the FSAR.

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. The nuclear system equipment, nuclear instrumentation, and the Reactor Protection System (RPS), 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: The operating state of the reactor moves along this line for the normal control rod withdrawal sequence in the absence of recirculation pump operation.
- Maximum Rod Line: The maximum rod line passes through 100% power at 85% 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.

4.4.3.3.2 Other Performance Characteristics

Other performance characteristics shown on the power/flow operating map are:

- Constant Rod Lines A, B, C, D, E, F—These lines show the change in flow associated with power change while maintaining constant control rod position.

- Constant Pump Speed Lines 1, 2, 3, 4, 5, 6, 7, 8, 9—These lines show the change in flow associated with power changes while maintaining constant RIP speeds.

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 (30%). Power changes, during normal startup and shutdown, will be in this region. The normal operating procedure is to startup along Curve 1 shown in Figure 4.4-1.
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 selected control rods run-in (SCRRI).
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 RIPs are equipped with anti-rotation devices (ARD) which prevent a tripped RIP from rotating backwards. The ARD begins operating at 31.4 rad/s decreasing speed. In order to prevent mechanical wear in the ARD, minimum speed is specified at 31.4 rad/s. However, to provide a stable operation, the minimum pump speed is set at 47.1 rad/s (30% of rated).

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 (30% 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 (Curves 1 through 9). 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 (Curves A through F).

4.4.3.4 Temperature-Power Operating Map

Temperature-power operating maps are not applicable to ABWR systems.

4.4.3.5 Load-Following Characteristics

Large negative operating coefficients inherent in the ABWR provide the following important advantages:

- Load-following with damped behavior in the heat transfer response.
- Load-following with recirculation flow control.
- Strong damping of spatial power distribution.

The design of the ABWR includes the ability to follow load demands over a reasonable range without requiring operator action. The Recirculation Flow Control System (RFC) load demand response in automatic mode is capable of responding to step load changes of 10 percent in the power range between 65 to 100 percent of rated power.

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 RCS.

4.4.3.7 Thermal Hydraulic Stability Performance

In light of the BWR thermal hydraulic instability experienced at the Lasalle NPS, the ABWR design assures that stability performance in the normal operating region (Regions I and IV in Figure 4.4-1) is more stable than current operating BWRs by incorporating the following design features:

(1) Smaller inlet orifices, which increase the inlet single-phase pressure drop, and, consequently, improve the core and channel stability.

(2) Wider control rod pitch, which increases flow area, and, reduces the void reactivity coefficient and improves both core and channel stability.

(3) More steam separators, which reduce the two-phase pressure drop, and improve stability.

(4) Automatic logic which prevents plant operation in the region with the least stability margin. See Subsection 7.7.1.2 (1)(g) for description of logic.

In order to reconfirm this conclusion, a stability analysis based on the procedures developed by the BWROG committee on thermal hydraulic stability (Reference 4.4-7) was performed for the Lungmen NPS . In this analysis, conservative nuclear conditions, taking into consideration of future core design, were assumed. The results at the most limiting conditions in the normal operating region (i.e., the intercept of the maximum rod line with all operating RIPs at their minimum speeds, assuming only 9 out of 10 RIPs are in operation) are as follows:

- Core Decay Ratio 0.54
- Channel Decay Ratio 0.12

These results are shown in Figure 4.4-3 together with the criteria. From Figure 4.4-3, it is confirmed that the Lungmen NPS is stable in the normal operating region.

Furthermore, automatic logic (Figure 4.4-4) which prevent plant operation in the region with the least stability margin is also implemented. This design is similar to Option I-A, one of long-term solutions considered by the BWROG. In addition, in order to meet the stability design requirements specified in the ALWR Utility Requirements Document, Option III, LPRM based oscillation power range monitor (OPRM), which is also one of the long-term solutions considered by the BWROG, has been implemented in the ABWR design.

As for issues that relate to anticipated transient without scram (ATWS) stability, they are of no concern to the ABWR design, since the ABWR design has logic to automatically initiate the Standby Liquid Control System (SLC), including automatic initiation of feedwater run back. Furthermore, the ABWR emergency procedure guideline (EPG) will incorporate any changes recommended by the BWROG.

In summary, the ABWR stability design is consistent with the licensing methodology proposed by the BWROG committee on thermal hydraulic stability. The ABWR will be stable in the normal operating region.

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 (RPV) or its internals. The LPMS can evaluate the characteristics of the sensed vibration signals and provide input to plant operators so they can diagnose the signal.

Units 1 & 2 each have identical LPMSs. But, they are not connected together in any way nor do they share any common sensors.

4.4.4.1 Power Generation Design Bases

The LPMS is designed to provide detection and operator alert to the presence of loose metallic parts in the RPV in time for the operator to take appropriate action 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 for indications of loose metallic parts within the RPV. 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 appropriate filtering of the signal. The capability to avoid false alert signals due to plant maneuvers such as control rod movement and other actions is required.

LPMS sensors are usually accelerometers. The array of LPMS accelerometers typically consist of a set of redundant sensors on separate channels 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 near the (1) main steam outlet nozzles, (2) feedwater inlet nozzles, (3) flooded nozzles, and (4) control rod drive housings. The sensors are mounted in such a fashion as to provide good response and sensitivity.

The on-line 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 J or more on the inside surface of the RPV within 0.91 m 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 and are filtered out by LPMS.

Physical separation is maintained between the signal cables of the redundant sensors as they are grouped and routed through cable penetrations to the LPMS panels. Each group of sensor signals is terminated at a separate LPMS panel. The two LPMS panels are located in an area that 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.

The manual mode is used for the following functions:

(1) Preoperational testing

- Establish the alert levels for these modes
- Perform channel checks
- Perform channel function tests
- Verify that the background noise measured is sufficiently small that the signal associated with a detectable loose-part impact would be clearly discernible

(2) Startup and power operation

- Establish alert levels for these modes
- Perform channel checks
- Manually listen to audio portion of signals from selected sensors to detect loose parts
- Perform channel function tests
- Verify that the background noise measured during normal plant operation is sufficiently small that the signal associated with a detectable loose-part impact would be clearly discernible

(3) Cold shutdown or refueling to verify channel calibration

- Perform channel checks
- Perform channel function tests

4.4.4.3 Normal System Operation

The LPMS will be set to alarm for detected signals having characteristics of metal-to-metal impacts that exceed the alarm setpoint and will simultaneously start recording at least four sensors at a time on the Data Recorder.

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 corresponding unique frequency spectrum. This signature and spectrum results from a combination of both internal and external sources due to normal and transient conditions.

Discrimination logic is 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.

4.4.4.4 Safety Evaluation

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. 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 and the LPMS will not initiate any automatic actions which affect the reactor operation. The plant operator makes the preliminary evaluation of what actions are required based on available information. If the presence of unusual metal impact sound is indicated, then the station engineers perform additional evaluations. Plant personnel with expertise in analyzing LPMS data 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.

4.4.4.5 Calibration

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., LPMS sensor signals disabled).

Provisions will be made to check the LPMS calibration at the LPMS data acquisition and analysis panels during 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.

4.4.4.6 Seismic and Environmental Conditions

The LPMS electronic components located inside the containment are designed and installed to perform their function following all seismic events up to and including an Operating Basis Earthquake.

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.

4.4.4.7 Installation, Testing, Operation and Maintenance Training

Provisions exist for periodic on-line sensor signal channel check and functional tests and for off-line 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 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.

4.4.4.8 Design Life

All LPMS electronic components within the containment are designed for a 40-year design life. In those instances where a 40-year design life is not practicable, a replacement program will be established for those parts that are anticipated to have limited service life.

4.4.4.9 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 operation 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, pressure, and axial power shape which exists at the specified reactor condition. Thermal margin is stated in terms of the 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 (Reference 4.4-6).

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 transient CPR margin loss (Delta CPR) and the peak cladding temperature (PCT) for the loss of coolant accident (LOCA) are influenced by the axial and local power (R factor) distribution of a fuel bundle. The effect of these power distributions on the transient delta CPR is discussed in Appendix V and the effect on the PCT in Appendix VI of Reference 4.4-6.

4.4.5.4 Core Thermal Response

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

4.4.5.5 Analytical Methods

4.4.5.5.1 Fuel Cladding Integrity Safety Limit

The generation of the 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 AOO. The Safety Limit MCPR (SLMCPR) is calculated every cycle using the actual core loadings for the plant and cycle to be analyzed.

The criteria used to calculate the safety limit include:

(1) Core radial power distribution selected to maximize the number of bundles at or near the MCPR limit.

(2) 99.9% of the fuel rods in the core expected to avoid boiling transition.

See additional information on the safety limit MCPR contained in Section 6.2 of Reference 4.4-1.

4.4.5.5.2 MCPR Operating Limit Calculation

A plant and cycle 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 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 one dimensional dynamic model (ODYN) as documented in Section 6.3 of Reference 4.4-1.

4.4.5.5.4 Calculational Procedure for AOO Slow Events

Core-wide non-pressurization AOOs are analyzed using the REDY transient model as documented in Section 6.3 of Reference 4.4-1.

The ABWR version of the REDY code incorporates minor improvements to model features unique to the ABWR, and also improvements have been made to the physical and numerical models in several areas. The core hydraulic model includes a bypass region in addition to the active core region. A more detailed vessel pressure distribution model has been added. Additional changes are the capability to model three separate groupings of internal recirculation pumps, steam quenching in the vessel dome, an option to include momentum balance in the steamline, and the ability to simulate cold water injection into the lower plenum.

Comparisons of results calculated using the ABWR version of REDY have been compared against results calculated using the ABWR version of ODYN, which has been qualified as discussed in Subsection 4.4.5.5.3. The comparisons indicate that the ABWR version of REDY conservatively predicts results relative to ODYN, and therefore it is concluded that the ABWR version of REDY is a satisfactory predictor of ABWR transient behavior.

4.4.6 Testing and Verification

The testing and verification techniques will 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.

4.4.7 References

4.4-1 *GESTAR III Republic of China*, General Electric Standard Application for Reactor Fuel, NEDE-24011-P-A-RC-7.

4.4-2 *Core Flow Distribution in a Modern Boiling Water Reactor as Measured in Monticello*, NEDO-10299A, October 1976.

4.4-3 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.

4.4-4 *Brunswick Steam Electric Plant Unit 1 Safety Analysis Report for Plant Modifications to Eliminate Significant In-Core Vibrations*, NEDO-21215, March 1976.

4.4-5 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.

4.4-6 *General Electric BWR Thermal Analysis Basis (GETAB): Data Correlation and Design Application*, NEDO-10958-A, January 1977.

4.4-7 Glen Watford, *BWR Owners' Group Long-term Stability Solutions Licensing Methodology*, NEDO-31960, June 1991.

4.4-8 PANACEA BWR Core Simulator, Volume I, NEDE-20884, March 1988.

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

General Operating Conditions	Lungmen NPS* 278-872	ABWR 278-872
Reference design thermal output (MWt)	3926	3926
Power level for engineered safety features (MWt)	4005	4005
Steam flow rate, at 215.6°C final feedwater temperature (Kg/h)	7.64×10^6	7.64×10^6
Core coolant flow rate (Kg/h)	52.2×10^6	52.2×10^6
Feedwater flow rate (Kg/h)	7.62×10^6	7.62×10^6
System pressure, nominal in steam dome (MPa)	7.2	7.2
System pressure, nominal core design (MPa)	7.3	7.3
Coolant saturation temperature at core design pressure (°C)	288.3	288.3
Average power density (kW/L)	49.2	50.6
Core total heat transfer area (m^2)	9283.8	7727.3
Design operating minimum critical power ratio (MCPR)	1.31	1.17
Core inlet enthalpy at 215.6°C FFWT (kJ/kg)	1227.2	1227.2
Core inlet temperature at 215.6°C FFWT (°C)	278.5	278.5
Core maximum exit voids within assemblies (%)	75.1	75.1
Core average void fraction, active coolant	0.440	0.408
Active coolant flow area per assembly (cm^2)	92.78	101.1
Core average inlet velocity (m/s)	1.96	1.96
Maximum inlet velocity (m/s)	2.27	2.27
Total core pressure drop (MPa)	0.168	0.168
Core support plate pressure drop (MPa)	0.138	0.138
Average orifice pressure drop, central region (MPa)	0.060	0.060
Average orifice pressure drop, peripheral region (MPa)	0.122	0.122
Maximum channel pressure loading (MPa)	0.090	0.075
Average-power assembly channel pressure loading (bottom) (MPa)	0.070	0.066
Shroud support ring and lower shroud pressure loading (MPa)	0.205	0.165
Upper shroud pressure loading (MPa)	0.043	0.024

* Based on the core loading in Figure 4.3-1

Table 4.4-2 Void Distribution for Analyzed Core*

Core Average Value - 0.408 **Maximum Exit Value - 0.751** **Active Fuel Length - 3.71 m**			
	Node	**Core Average (Average Node Value)**	**Maximum Channel (End of Node Value)**
Bottom of Core	1	0	0
	2	0	0.011
	3	0.010	0.069
	4	0.045	0.162
	5	0.104	0.260
	6	0.174	0.346
	7	0.247	0.421
	8	0.315	0.481
	9	0372	0.529
	10	0.419	0.567
	11	0.458	0.597
	12	0.489	0.623
	13	0.515	0.643
	14	0.536	0.660
	15	0.554	0.675
	16	0.570	0.688
	17	0.585	0.700
	18	0.599	0.712
	19	0.611	0.723
	20	0.623	0.733
	21	0.633	0.740
	22	0.641	0.746
	23	0.646	0.750
Top of Core	24	0.649	0.751

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* For standard ABWR fuel design. Updated table for Lungmen GE12 fuel design will be provided in the FSAR.

Table 4.4-3 Flow Quality Distribution for Analyzed Core*

Core Average Value - 0.145
Maximum Exit Value - 0.258
Active Fuel Length - 3.71 m

	Node	Core Average (Average Node Value)	Maximum Channel (End of Node Value)
Bottom of Core	1	0	0
	2	0	0
	3	0	0.002
	4	0	0.008
	5	0	0.019
	6	0.005	0.033
	7	0.016	0.051
	8	0.028	0.069
	9	0.040	0.087
	10	0.052	0.105
	11	0.064	0.122
	12	0.074	0.137
	13	0.085	0.152
	14	0.094	0.165
	15	0.103	0.177
	16	0.110	0.189
	17	0.118	0.200
	18	0.126	0.212
	19	0.134	0.224
	20	0.142	0.235
	21	0.148	0.245
	22	0.154	0.252
	23	0.158	0.257
Top of Core	24	0.159	0.258

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* For standard ABWR fuel design. Updated table for Lungmen GE12 fuel design will be provided in the FSAR

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

	Node	Axial Power Factor
Bottom of Core	1	0.38
	2	0.69
	3	0.93
	4	1.10
	5	1.21
	6	1.30
	7	1.47
	8	1.51
	9	1.49
	10	1.44
	11	1.36
	12	1.28
	13	1.16
	14	1.06
	15	1.01
	16	0.97
	17	0.94
	18	0.97
	19	0.96
	20	0.91
	21	0.77
	22	0.59
	23	0.38
Top of Core	24	0.12

* For standard ABWR fuel design. Updated table for Lungmen GE12 fuel design will be provided in the FSAR

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 (Not including the CRD housing volume)	4.65	4.65 4.65	0.0	19.5
B. Core	4.36	4.36 4.36	4.65	†
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 (Not including the RIPs)	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 including bypass, this information will be provided in FSAR.

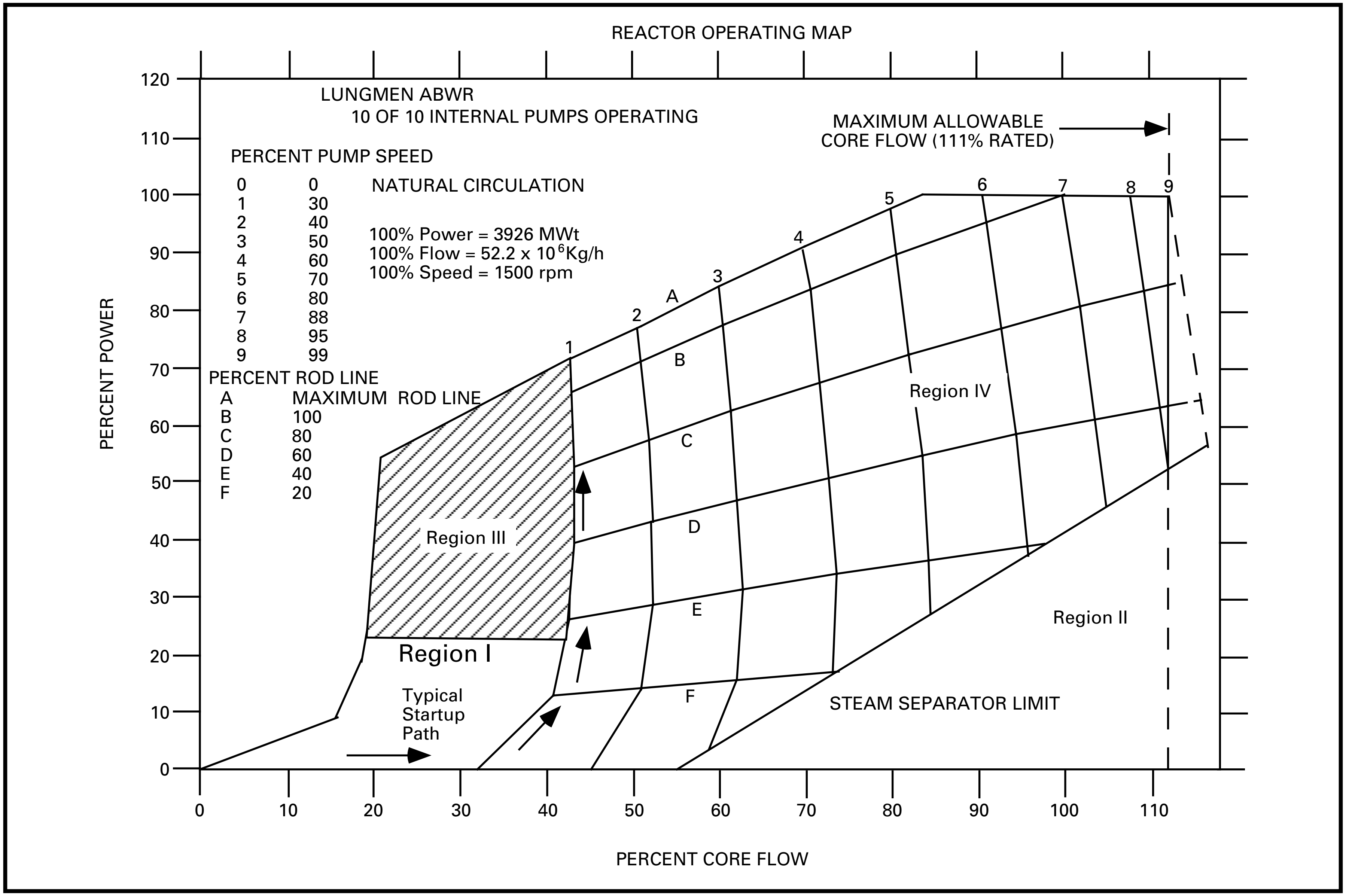


Figure 4.4-1 Power-Flow Operating Map Used for System Response Study

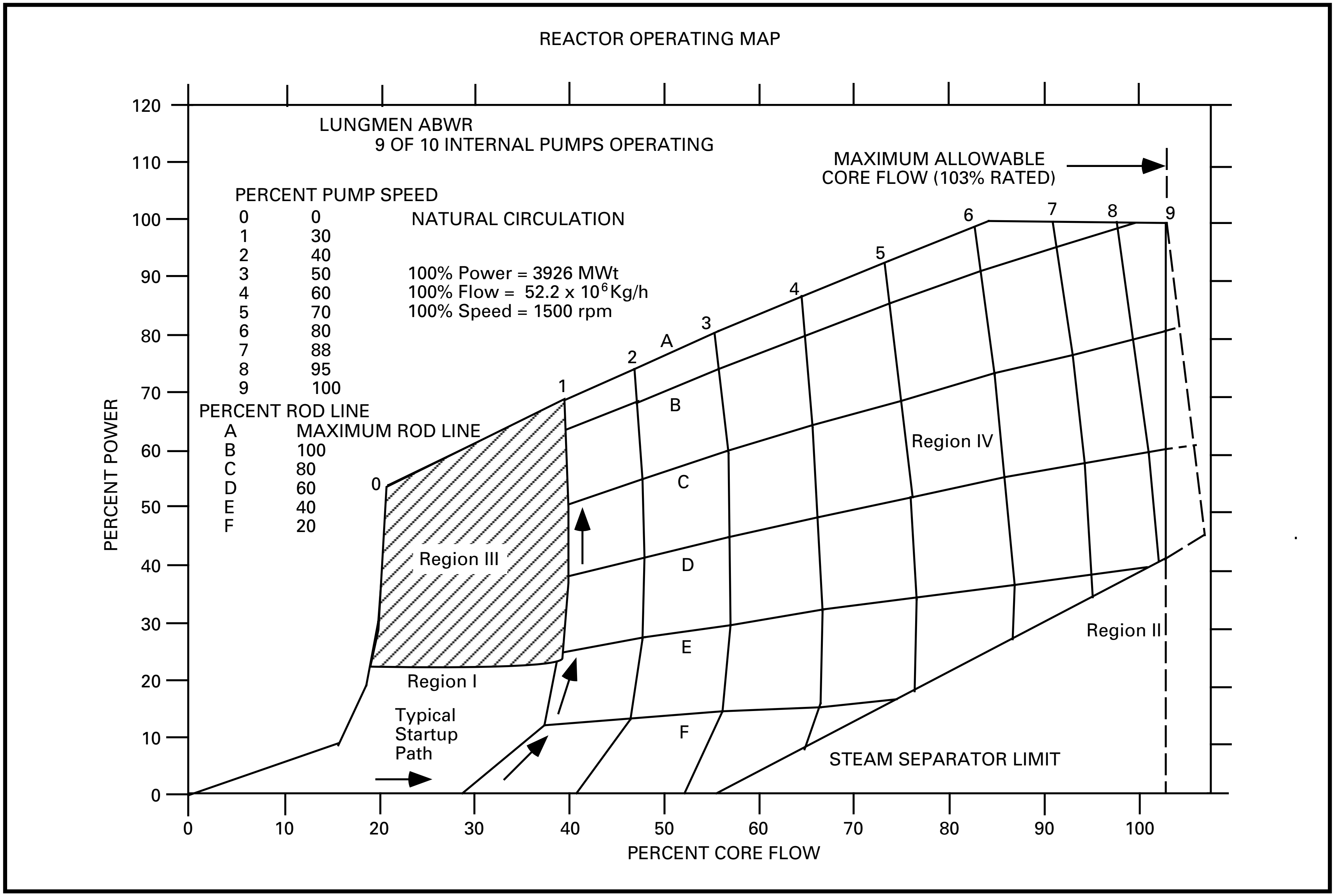
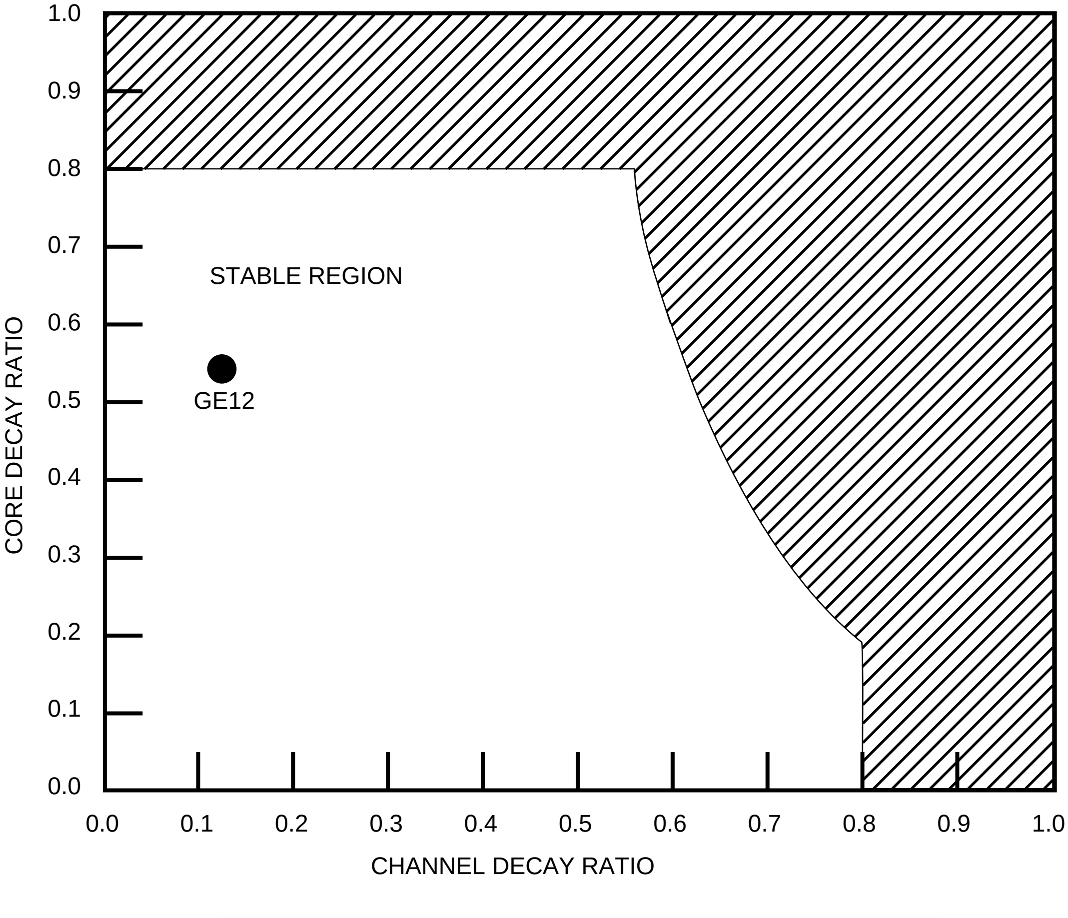
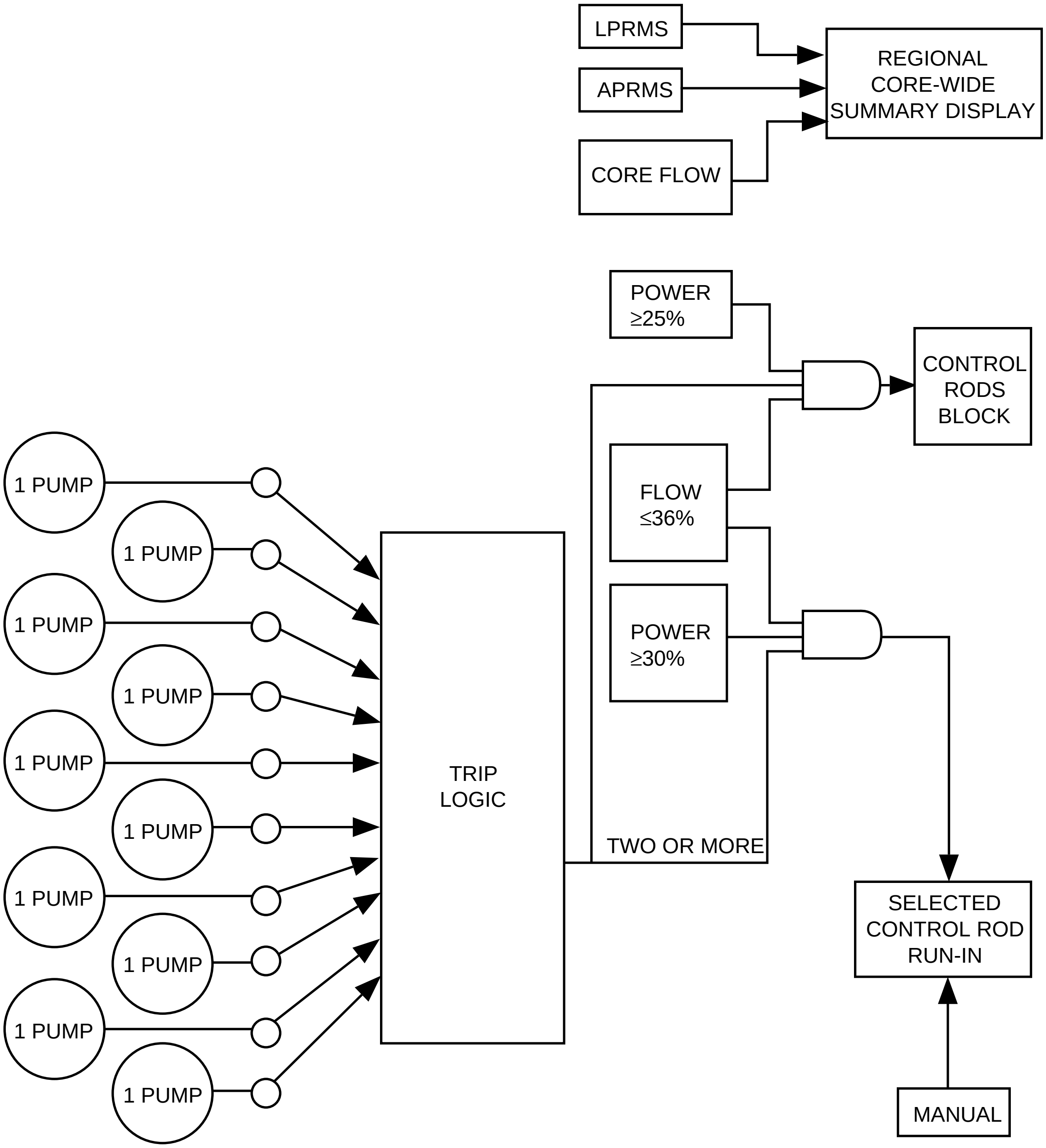


Figure 4.4-2 Power-Flow Operating Map Used for System Response Study (9 RIPs Operation)



LUNGMEN NPS IS STABLE IN NORMAL OPERATING DOMAIN.

Figure 4.4-3 Lungmen NPS Stability



NOTES: POWER ≥30%: TO ASSURE POWER LEVEL BELOW 80% ROD LINE AT NATURAL CIRCULATION.

FLOW ≤36%: TO ASSURE FLOW RATE IS HIGHER THAN THAT OF 8 RIPS OPERATIONS WITH THE MINIMUM PUMP SPEED.

Figure 4.4-4 Stability Controls and Protection Logic