3.7 Thermal and Hydraulic Design

3.7.1 Power Generation Objective

The objective of the thermal and hydraulic design of the core is to achieve power operation of the fuel over the life of the core without sustaining fuel damage.

3.7.2 Power Generation Design Basis

The thermal hydraulic design of the core shall provide the following characteristics:

a. The ability to achieve rated core power output throughout the design lifetime of the fuel without sustaining fuel damage.

b. The flexibility to adjust core power output over the range of plant load and load maneuvering requirements without sustaining fuel damage.

3.7.3 Safety Design Basis

1. The thermal hydraulic design of the core shall establish limits for use in setting devices of the nuclear safety systems so that no fuel damage occurs as a result of abnormal operational transients (see Chapter 14, “Plant Safety Analysis”).

2. The thermal hydraulic design of the core shall establish a thermal hydraulic safety limit for use in evaluating the safety margin relating the consequences of fuel barrier failure to public safety.

3.7.4 Thermal and Hydraulic Limits

3.7.4.1 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, the Average Planar Linear Heat Generation Rate (APLHGR) must be maintained below the required Maximum APLHGR limit (MAPLHGR) and the Linear Heat Generation Rate (LHGR) must be maintained below the required Maximum LHGR limit (MLHGR). The steady-state MCPR, MAPLHGR, and MLHGR limits are determined by analysis of the most severe moderate frequency Abnormal Operational Transients (AOTs) to accommodate uncertainties and provide reasonable assurance that no fuel damage results during moderate frequency AOTs at any time in life.

3.7.4.2 Requirements for Abnormal Operational Transients (AOTs)

The MCPR, MAPLHGR, and MLHGR limits are established such that no safety limit is expected to be exceeded during the most severe moderate frequency AOT event as defined in Chapter 14, "Plant Safety Analysis."

3.7.4.3 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 AOT. Demonstration that the steady-state MCPR, MAPLHGR, and MLHGR limits are not exceeded is sufficient to conclude that the design bases are satisfied.

3.7.5 Description of Thermal - Hydraulic Design of the Reactor Core

3.7.5.1 Critical Power Ratio

A description of the critical power ratio is provided in Subsection 3.7.7.1, "Critical Power." Criteria used to calculate the critical power ratio safety limit are given in References 32 and 42 for AREVA reload analyses.

3.7.5.2 Average Planar Linear Heat Generation Rate (APLHGR)

Models used to calculate the APLHGR limit are given in Subsection 3.2.5.1, "Evaluation Methods," as pertaining to the fuel mechanical design limits, and in Subsection 6.5.2.1, "Analysis Model," as pertaining to 10 CFR 50, Appendix K limits.

3.7.5.3 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 2, 3, and 4). The components of bundle pressure drop considered are friction, local, elevation, and acceleration (Subsections 3.7.5.4.1 through 3.7.5.4.4, 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 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 tieplate and through the lower tieplate holes into the bypass flow region. All 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 exits 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. The bypass flow paths considered in the analysis and typical values of the fraction of bypass flow through each flow path are given in Reference 5.

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. In evaluating fluid properties a constant pressure model is used.

For core design and monitoring, assembly-specific 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.

3.7.5.4 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.

3.7.5.4.1 Friction Pressure Drop

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

$$\Delta P_f = \frac{w^2}{2 g_c \rho} \frac{fL}{D_H A_{ch}^2} \Phi_{TPF}^2$$

where

ΔP_f = friction pressure drop, psi

w = mass flow rate

g_c = conversion factor

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

AREVA pressure drop methodology is described in References 33, 34, and 35.

3.7.5.4.2 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

$$\Delta P_L = \frac{w^2}{2 g_c \rho} \frac{K}{A^2} \Phi_{TPL}^2$$

where

ΔP_L = local pressure drop, psi
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. For AREVA analyses the Reference 33, 34, and 35 methodologies are used. For advanced spacer designs a quality modifier has been incorporated in the two-phase multiplier to better fit the data. Empirical constants were added to fit the results to data taken for the specific designs of the BWR fuel assembly. These data were obtained from tests performed in single-phase water to calibrate the orifice, the lower tieplate, and the holes in the lower tieplate, and in both single- and two-phase flow, to derive the best fit design values for spacer and upper tieplate pressure drop. The range of test variables was specified to include the range of interest for boiling water reactors. New test data are obtained whenever there is a significant design change to ensure the most applicable methods are used.

3.7.5.4.3 Elevation Pressure Drop

The elevation pressure drop is based on the relation:

$$\Delta P_E = \bar{\rho} \Delta L \frac{g}{g_c} \; ;$$

$$\bar{\rho} = \rho_f (1 - \alpha) + \rho_g \alpha$$

where

ΔP_E = elevation pressure drop, psi

ΔL = incremental length

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

g = acceleration of gravity

α = nodal average void fraction

ρ_f, ρ_g = saturated water and vapor density, respectively

AREVA void fraction models are described in References 33, 34, and 35.

3.7.5.4.4 <u>Acceleration Pressure Drop</u>

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:

$$\Delta P_{ACC} = (1 - \sigma_A^2)\frac{w^2}{2 g_c \rho_f A_2^2}$$

$$\sigma_A = \frac{A_2}{A_1}$$

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:

$$\Delta P_{ACC} = (1 - \sigma_A^2)\frac{w^2 \rho_H}{2 g_c \rho_{KE}^2 A_2^2}$$

where

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

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

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:

$$\Delta P_{ACC} = \frac{w^2}{g_c A_{ch}^2}\left[\frac{1}{\rho_{OUT}} - \frac{1}{\rho_{IN}}\right]$$

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

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

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

3.7.5.5 Correlation and Physical Data

AREVA has obtained substantial amounts of physical data in support of the pressure drop and thermal-hydraulic loads discussed in Subsection 3.7.5.4, "Core Pressure Drop and Hydraulic Loads." Correlations have been developed to fit these data to the formulations discussed.

3.7.5.5.1 Pressure Drop Correlations

AREVA has taken significant amounts of friction pressure drop data in multi-rod geometries 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 reported in Subsections 3.7.5.4.1 and 3.7.5.4.2. Tests are performed in single-phase water to calibrate the orifice and the lower tie-plate, and in both single-phase and two-phase flow 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 boiling water reactors. 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 3.7.5.4.1 and 3.7.5.4.2 have been confirmed by full scale prototype flow testing.

3.7.5.5.2 Void Fraction Correlation

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

3.7.5.5.3 Heat Transfer Correlation

The fuel heat transfer correlations for AREVA reload analyses are described in Reference 37.

3.7.5.6 Thermal Effects of Abnormal Operational Transients

The evaluation of the core's capability to withstand the thermal effects resulting from abnormal operational transients is covered in Chapter 14, "Plant Safety Analysis".

3.7.5.7 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 3.7.7.1.1, "Fuel Cladding Integrity Safety Limit."

3.7.5.8 Flux Tilt Considerations

For flux tilt considerations, refer to Subsection 3.6.4.2, "Power Distribution."

3.7.6 Description of the Thermal-Hydraulic Design of the Reactor Coolant System

3.7.6.1 Power/Flow Operating Map

3.7.6.1.1 Performance Range for Normal Operations

A boiling water reactor must operate within certain restrictions due to pump net positive suction head (NPSH) requirements, overall plant control characteristics, core thermal power limits, etc. A typical operating power-flow map for BFN is shown in Figure 3.7-1. The boundaries on the maps are as follows:

Natural Circulation Line (Line A in Figure 3.7-1)

The operating state of the reactor moves along this line for the normal control rod withdrawal sequence in the absence of recirculation pump operation.

<u>20 Percent Pump Speed Line (Line B in Figure 3.7-1)</u>

The operating state for the reactor follows this line for the normal control rod withdrawal sequence with recirculation pumps operating at approximately 20 percent speed.

<u>100 Percent Rod Line (Line which runs through point E in Figure 3.7-1)</u>

The 100% rod line passes through 100 percent power at 100 percent flow. The operating state for the reactor follows this line (or one roughly parallel to it) for recirculation flow changes with a fixed control rod pattern. The line is based on constant xenon concentration.

<u>APRM Rod Block Line (Shown in Figure 3.7-1)</u>

The line shown on the graph limits control rod withdrawal to within the constraint of the control rod block line.

<u>Pump Constant Speed Line</u>

The lines passing through Points H and I and Points G and J show the change in flow associated with power reduction by control rod insertion from 3458 MWt, 105% core flow and 3458 MWt, 100% core flow, respectively, while maintaining constant recirculation pump speed.

<u>Minimum Expected Flow Control Line (Shown in Figure 3.7-1)</u>

This line approximates the expected flow control line that the plant would follow if core flow were rapidly reduced from 107.7% of rated core flow at 35.4% power by reducing recirculation pump speed to 20% with a constant control rod pattern.

<u>Minimum Power Line (Shown in Figure 3.7-1)</u>

Operation above the line passing through Points L and K prevents cavitation of the jet pumps and reactor recirculation pumps by ensuring adequate subcooling and therefore net positive suction head for all normal modes of jet pump and reactor recirculation pump operation.

<u>Increased Core Flow (ICF) Region (The region bounded by Points E, G, J, I, H, and F in Figure 3.7-1)</u>

The plant is licensed for Increased Core Flow (ICF) operation up to a maximum of 105% of rated core flow at 100% rated power. At core thermal powers less than rated but greater than or equal to 3458 MWt, the maximum allowable core flow is

limited to 105% of rated core flow. At thermal powers less than 3458 MWt, the maximum core flow is set by the constant recirculation pump speed line that passes through Point H, up to a maximum core flow of 112.6% of rated flow at 35.4% rated power on the Power/Flow operating map. ICF can be used to extend full power operation beyond the point where all rods are out at rated power and flow conditions (End of Full Power Life - EOFPL). ICF may be used prior to reaching EOFPL for operating flexibility.

Maximum Extended Load Line Limit Analysis (MELLLA) Region (shown in Figure 3.7-1)

The plant is licensed for Maximum Extended Load Line Limit Analysis (MELLLA) which allows operation at 3458 MWt down to 81% rated flow conditions. As shown in Figure 3.7-1, with power uprate to 3952 MWt, MELLLA allows operation at full power down to 99% rated flow conditions. The plant is also licensed for operation in the MELLLA+ region, which restores much of the operating flexibility at rated power that previously existed in the MELLLA domain prior to implementation of power uprate to the licensed core thermal power level of 3952 MWt. This is accomplished by expanding the licensed operating domain to allow operation above the MELLLA upper boundary at core flows greater than or equal to 55% of the rated core flow with reactor power up to 100% of rated thermal power. The MELLLA+ region upper boundary passes through 100% power at 85% core flow (point N). The operating state of the reactor follows this line (or one similar) for recirculation flow changes with a fixed control rod pattern at a constant xenon condition. The MELLLA+ upper boundary line terminates at 55% core flow (point O).

3.7.6.1.2 Flow Control

The following simplified description of BWR operation summarizes the principle modes of normal power range operation. Prior to unit startup the recirculation pumps are started one at a time and typically held at a pump speed of 28 percent or less of rated speed. The first part of the startup sequence is achieved by withdrawing control rods with the recirculation pumps at a pump speed of 28 percent or less of rated speed. Core power, steam flow, and feedwater flow increase as control rods are withdrawn by the operator, until feedwater flow increases to a point above the feedwater flow interlock. The low feedwater flow interlock (approximately 19% feedwater flow) prevents low power-high recirculation flow combinations which may create recirculation system NPSH problems. The natural circulation characteristics of the BWR are still very influential in this part of the power flow region.

Once the feedwater interlock has been cleared the recirculation flow in each loop can be increased to increase power. The operator then can achieve full power by a combination of control rod withdrawals and pump speed increases, depending on operating and core management strategies. A typical strategy for plant startup is to increase core flow to a mid-range value (typically greater than 55% core flow). Then control rods are withdrawn to a point just below the upper boundary of the licensed operating domain. Core flow can then be increased until the desired high power condition is reached. The normal power range operation is bounded by the licensed operating domain maximum rod line and 100% power.

The large negative operating coefficients, which are inherent in the BWR, provide the following important advantages:

1. Stable load change response following with well damped behavior and little undershoot or overshoot in the heat transfer response.

2. Load changes with recirculation flow control.

3. Strong damping of spatial power disturbances.

To increase reactor power, it is necessary only to increase the recirculation flow rate which reduces core average void content, causing an increase in core reactivity. As the reactor power increases, more steam is formed and the reactor stabilizes at a new power level with the transient excess reactivity balanced by the new void formation. No control rods are moved to accomplish this power level change. Conversely, when a power reduction is required, it is necessary only to reduce the recirculation flow rate. When this is done, more voids are formed in the moderator, and the reactor power output automatically decreases to a new power level commensurate with the new recirculation flow rate. No control rods are moved to accomplish the power reduction.

Varying the power level by varying the recirculation flow rate (flow control) is more advantageous than using control rod positioning. Flow variations perturb the reactor uniformly in the horizontal planes and thus, allow operation with flatter power distribution and reduced transient allowances. As the flow is varied, the power and void distributions remain approximately constant at the steady-state end points for a wide range of flow variations. These constant distributions provide the important advantage that the operator can adjust the power distribution at a reduced power and flow by movement of control rods and then bring the reactor to rated conditions by increasing flow, with the assurance that the power distribution will remain approximately constant. Subsection 7.9, "Recirculation Flow Control System," describes the means by which recirculation flow is varied.

3.7.6.2 Thermal-Hydraulic Stability Performance

The AREVA analytical methodology for demonstrating stability compliance for AREVA fuel designs is described in Subsection 3.6.4.6, “Stability.” To provide additional assurance that regional instabilities will not occur, Browns Ferry has implemented the long-term stability solution designated as Option III in NEDO-31960, Supplement 1, “BWR Owners’ Group Long-Term Stability Solution Licensing Methodology,” for Units 2 and 3 and DSS-CD in NEDC-33075P-A, “GE Hitachi Boiling Water Reactor Detect and Suppress Solution – Confirmation Density,” for Unit 1. As part of the implementation of the MELLLA+ operating domain expansion, the Option III stability solution is replaced as the licensing basis stability solution by the Detect and Suppress Solution – Confirmation Density (DSS-CD) long term stability solution (NEDC-33075P-A, Revision 8, “GE Hitachi Boiling Water Reactor Detect and Suppress Solution – Confirmation Density”). The existing Option III algorithms are retained, with generic setpoints, to provide defense-in-depth protection for unanticipated stability events.

For DSS-CD long-term stability solution, the Oscillation Power Range Monitor (OPRM) Upscale Trip function of the Power Range Neutron Monitoring (PRNM) system is enabled. [Note: See Section 7.5.7.3.5 for a detailed description of the OPRM system.] The OPRM Upscale Trip function provides protection from exceeding the fuel MCPR Safety Limit in the event of thermal-hydraulic power oscillations. The OPRM receives input signals from the Local Power Range Monitors (LPRMs) within the reactor core. An Upscale Trip is issued if oscillatory changes in the neutron flux are detected. The OPRM Upscale Trip function is required to be operable when the plant is in a region of power-flow operation where actual thermal-hydraulic oscillations might occur (Tech Spec enabled region - greater than 23% rated thermal power and less than 75% recirculation drive flow.

The DSS-CD setpoints that provide protection of the Safety Limit MCPR during instability events are established on a cycle independent basis and are confirmed to be applicable for each reload cycle via an NRC approved process using established applicability checklists. If necessary, cycle specific setpoint adjustments can be implemented to provide adequate Safety Limit MCPR protection.

For Unit 1, a cycle specific DSS-CD stability analysis is performed for each reload cycle to conform the plant-applicability checklist in NEDC-33075P-A remains satisfied and define the cycle specific BSP regions in the COLR. The confirmation of the checklist ensures that the DSS-CD setpoints are applicable for that cycle and provide protection of the fuel limits.

If the OPRM trip function should become inoperable, alternate methods of stability protection are implemented in accordance with the Technical Specifications.

3.7.7 Evaluation

The thermal-hydraulic design of the reactor core and reactor coolant system is based upon an objective of no fuel damage during normal operation or during abnormal operational transients. This design objective is demonstrated by analysis as described in the following sections.

3.7.7.1 Critical Power

The objective for normal operation and AOTs 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, 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 AOTs 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 32).

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

3.7.7.1.1 Fuel Cladding Integrity Safety Limit

The generation of the Minimum Critical Power Ratio (MCPR) limit requires a statistical analysis of each reload core near the limiting MCPR condition. The MCPR Fuel Cladding Integrity Safety Limit applies not only for core wide AOTs, but is also applied to the localized rod withdrawal error AOT. The cycle-specific Safety Limit MCPR is derived based on methodology documented in References 32 and 42 for AREVA reload analyses. For AREVA reload analyses, the Reference 43, 44, and 46-47 approved critical power correlations are used as appropriate to specific fuel types.

The resulting safety limit MCPR for each cycle is given in the AREVA Reload Analysis Report for each BFN unit (included in Appendix N of the FSAR).

Statistical Model

The statistical analysis utilizes a model of the BWR core which 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. Details of the procedure are documented in References 32 and 42 for AREVA reload analyses. 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. Uncertainties typical of AREVA cycle-specific analyses are provided in Reference 39.

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.

BWR Statistical Analysis

Statistical analyses are performed for each operating cycle that provide the fuel cladding integrity safety limit MCPR. This safety limit MCPR is derived based on methodology documented in Reference 42.

3.7.7.1.2 MCPR Operating Limit Calculational Procedure

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

Calculational Procedure for AOT Pressurization Events

Core-wide rapid pressurization events (turbine trip w/o bypass, load rejection w/o bypass, feedwater controller failure) are analyzed using the system model COTRANSA2 (Reference 40) for AREVA reload analyses.

The Time Varying Axial Power Shape (TVAPS) calculation is performed by COTRANSA2 (Reference 40) for AREVA reload analyses. The TVAPS is a short time period phenomena that occurs during the control rod scram that terminates an AOT. The analytical procedures used to evaluate the AOTs account for TVAPS either in a bounding manner or explicitly, depending on the AOT and the fuel design.

Calculational Procedure for AOT Slow Events

For AREVA reload analyses, the MICROBURN-B2 (Reference 35) 3-D simulator code is used for quasi-steady-state loss of feedwater heating (LFWH) transients; for transient events that cannot be handled in a quasi-steady-state manner, COTRANSA2 (Reference 40) is used. Inadvertent HPCI startup is not analyzed due to the fact that the core enthalpy change for the event is similar to the loss of feedwater heating event and the severity of the inadvertent HPCI startup is generally bounded by the load reject no bypass and feedwater controller failure events. The loss of feedwater heating event is analyzed each cycle to demonstrate that the loss of feedwater heating event and inadvertent HPCI startup events remain non-limiting. When necessary, it is analyzed using COTRANSA2 for AREVA analyses.

Rod Withdrawal Error Calculational Procedure

The reactor core behavior during the rod withdrawal error transient is calculated by doing a series of steady-state three-dimensional coupled nuclear-thermal-hydraulic calculations using the 3-D BWR Simulator (Reference 35).

Event Descriptions

For AREVA analyses, the AOT descriptions in Chapter 14 of this UFSAR are utilized with appropriate cycle-specific input parameter updates.

MCPR Operating Limit Calculation

For AREVA reload analyses, the ΔCPR for rapid AOTs is calculated using XCOBRA (Reference 41) for the initial steady-state analysis and XCOBRA-T (Reference 37) for the transient thermal margin analysis of the limiting fuel assembly.

MCPR Uncertainty Considerations for AREVA Reload Analyses

For fast transient events, the one-dimensional kinetic thermal-hydraulic COTRANSA2 code is used for the reactor system analysis, with the XCOBRA/XCOBRA-T codes evaluating the initial and transient hot channel hydraulics and ΔCPR. The NRC approved application methodology of References 40, 41, and 42 provides adequate conservatism by accounting for uncertainties in the computed ΔCPR results. Therefore, for a given transient event, the required operating limit MCPR (OLMCPR) is simply the XCOBRA-T calculated ΔCPR added to the safety limit MCPR (SLMCPR).

The results of the system pressurization transients are sensitive to the control rod scram speed used in the calculations. To take advantage of average scram speeds

faster than those associated with the technical specifications surveillance times, scram speed-dependent MCPR limits are provided. If the control rod scram time performance is equal or better than the nominal scram speed (NSS) insertion times specified in the core operating limits report (COLR), the NSS-based MCPR operating limits apply. If the control rod scram time performance is equal or better than the optimal scram speed (OSS) insertion times specified in the COLR, the OSS-based MCPR operating limits apply. Otherwise, MCPR operating limits are applied that are based on the technical specification scram speed (TSSS) control rod insertion times. The plant technical specifications allow for operation with a certain number and arrangement of “slow” control rods as well as one stuck control rod. Conservative adjustments to the OSS, NSS, and TSSS scram speeds are input to the reload transient analyses to account for these slow and stuck rod effects on scram reactivity.

TSSS, NSS, and OSS-based power-dependent MCPR operating limits are reported in the reload licensing analysis report.

Low Flow and Low Power Effects on MCPR

The operating limit MCPR must be increased for low flow and, for plants with ARTS, low power conditions. This is because, in the BWR, power increases as core flow increases which results in a corresponding lower MCPR. If the MCPR at a reduced flow condition were at the 100% power and flow MCPR operating limit, a sufficiently large inadvertent flow increase could cause the MCPR to decrease below the Fuel Cladding Integrity Safety Limit MCPR.

The Average Power Range Monitor, Rod Block Monitor, and Technical Specification (ARTS) Improvement Program imposes both power- and flow-dependent limits imposed on the operating limit MCPR (OLMCPR). The flow-dependent required OLMCPR, $MCPR_f$, is defined as a function of the core flow rate and maximum rated power core flow capability. For powers between 100% of rated and the bypass point for the turbine stop valve/turbine control valve fast closure scram signal (about 26% of rated), the power-dependent OLMCPR, $MCPR_p$, is directly supplied. For powers between 23% rated and the bypass point, the $MCPR_p$ limits are absolute values and are defined separately for high core flows (>50% of rated flow) and for low core flows (≤50% of rated flow) conditions. There is no thermal limits monitoring required below 23% of rated power. The OLMCPR to be used at powers less than 100% becomes the most limiting value of either $MCPR_f$ or $MCPR_p$.

End-of-Cycle Coastdown Considerations

AOT analyses are performed at the full power, EOC, all-rods-out condition. Once an individual plant reaches this condition, it may shutdown for refueling or it may be placed in a coastdown mode of operation. In the end-of-cycle coastdown type of operation, if coastdown is initiated from a reactor power level

greater than 3458 MWt, recirculation pump speed must be decreased as reactor power coasts down in order to maintain core flow less than or equal to 105% of rated. Once reactor power is less than or equal to 3458 MWt, the plant is allowed to coastdown to a lower percent of rated power while maintaining recirculation pump speed less than or equal to the constant pump speed line corresponding to 105% of rated core flow at 3458 MWt.

For GE methods, in Reference 29, evaluations were made at 90%, 80%, and 70% power level points on the linear curve. The results show that the pressure and MCPR from the limiting pressurization AOT exhibit a larger margin for each of these points than the EOC full power, full flow case. MLHGR limits for the full power, rated flow case are conservative for the coastdown period, since the power will be decreasing and rated core flow will be maintained. Therefore, it can be concluded that the coastdown operation beyond full power operation is conservatively bounded by the analysis at the EOC conditions. In Reference 30, this conclusion is confirmedfor coastdown operation down to 40% power and is shown to hold for analyses performed with ODYN.

For AREVA methods, the nominal start of coastdown cycle exposure is conservatively extended. Coastdown limits are then determined at the final cycle exposure bounding of anticipated operation, forming the licensing basis maximum core average exposure (CAVEX).

3.7.7.2 Analysis Options

3.7.7.2.1 MCPR Margin Improvement Options

Several MCPR margin improvement options have been developed for operating BWRs. The following options are utilized at Browns Ferry:

(1) Recirculation Pump Trip
(2) Thermal Power Monitor
(3) Exposure-Dependent Limits
(4) Improved Scram Times

The exposure-dependent limits option is used on an as-needed basis. AREVA Reload Analysis Report, for each unit indicates which options are currently analyzed.

Recirculation Pump Trip

Due to operator concerns associated with the difficulty of restarting the recirculation pumps following a reactor scram, the recirculation pump trip (RPT) feature is maintained in a bypass condition.

For many of the plant operating cycles, the limiting AOTs are the turbine trip, generator load rejection, or other AOTs which result in a turbine trip. A significant improvement in thermal margin can be realized if the severity of these transients is reduced. The RPT feature accomplishes this by cutting off power to the recirculation pump motors anytime that the turbine control valve or turbine stop valve fast closure occurs.

This rapid reduction in recirculation flow increases the core void content during the AOT, thereby reducing the peak AOT power and heat flux.

Basically, the RPT consists of switches installed in both the turbine control valves and the turbine stop valves. When these valves close, breakers are tripped which releases the recirculation pumps to coast down under their own inertia.

Thermal Power Monitor

The APRM simulated thermal power trip (APRM thermal power monitor) is a minor modification to the APRM system. The modified APRM system generates two upscale trips. On one, the APRM signal (which is proportional to the thermal neutron flux) is compared with a reference which is not dependent on flow rate.

During normal reactor operations, neutron flux spikes may occur due to conditions such as transients in the recirculation system, transients during large flow control load maneuvers, or transients during turbine stop valve tests. The neutron flux leads the heat flux during transients because of the fuel time constant. And the neutron flux for these transients trips upscale before the heat flux increases significantly. (High heat flux is the precursor of fuel damage.) Thus, increased availability can be achieved without fuel jeopardy by adding a trip dependent on heat flux (thermal power).

For this trip, the APRM signal is passed through a low pass RC filter. It is compared with a recirculation flow rate dependent reference which decreases approximately parallel to the flow control lines.

In addition to increased availability, another benefit is that with the minor operational spikes filtered out, the heat flux trip setpoint is lower than the neutron flux trip setpoint. For long, slow AOTs such as the loss-of-feedwater heater, the heat flux and neutron flux are almost in equilibrium. For these AOTs, the lower trip setpoint

results in an earlier scram with a smaller increase in heat flux and a corresponding reduction in the consequences.

The APRM Simulated Thermal Power Trip at Browns Ferry is non-safety grade and is not taken credit for in any of the licensing transient analyses.

Exposure-Dependent Limits

The severity of any plant AOT pressurization event is worst at the end of the cycle primarily because the EOC all-rods-out scram curve gives the worst possible scram response. It follows that some limits relief may be obtained by analyzing the AOTs at other interim points in the cycle and administering the resulting limits on an “exposure dependent” basis.

This technique is straightforward and consists merely of repeating certain elements of the AOT analyses for selected midcycle exposures. Because the scram reactivity function monotonically deteriorates with exposure (after the reactivity peak), the limit determined for an exposure E_i is administered for all exposures in the interval $E_{i-1} < E \leq E_i$ where E_{i-1} is the next lower exposure point for which a limit was determined. This results in a table of MCPR limits to be applied through different exposure intervals of the cycle.

Improved Scram Times

As described in Section 3.7.7.1.2, subsection titled “MCPR Uncertainty Considerations for AREVA Reload Analyses”, for AREVA reload analyses power-dependent MCPR limits are provided for OSS, NSS, and TSSS bases. OSS and NSS bases limits may be used depending on scram speed measurements. Otherwise, the TSSS MCPR limits are applied. The TSSS and NSS-based power-dependent MCPR operating limits are reported in the reload licensing analysis report.

3.7.7.2.2 Operating Flexibility Options

A number of operating flexibility options have been developed for BWRs. The following options are utilized at Browns Ferry:

(1) Maximum Extended Load Line Limit
(2) Increased Core Flow
(3) Feedwater Temperature Reduction
(4) Turbine Bypass Out of Service
(5) ARTS Program
(6) Recirculation Coolant Pump Out of Service (Single-Loop Operation)
(7) End-of-Cycle Recirculation Pump Trip Out of Service

(8) Power Load Unbalance Out of Service
(9) Maximum Extended Load Line Limit Plus

The AREVA Reload Licensing Report for each unit indicates which options are currently analyzed (included in Appendix N of the FSAR).

Maximum Extended Load Line Limit

The Maximum Extended Load Line Limit Analysis (MELLLA) expands the operating domain to allow operation at 3458 MWt down to 81% rated flow conditions and at 3952 MWt down to 99% rated flow conditions. Addition of the MELLLA region provides improved power ascension capability to full power and additional flow range at rated power. Evaluations performed for MELLLA conditions include normal and AOTs, LOCA analysis, containment responses, and stability. The reload fuel dependent results of these analyses are re-evaluated each cycle.

Increased Core Flow Operation

Analyses are performed in order to justify operation at core flow rates in excess of the 100% rated flow condition. The limiting AOTs that are analyzed at rated flow as part of a standard supplemental reload licensing report are reanalyzed for increased core flow operation. In addition, the loss-of-coolant accident (LOCA), fuel loading error, rod drop accident, and rod withdrawal error are also re-evaluated for increased flow operation.

The effects of the increased pressure differences on the reactor internal components, fuel channels, and fuel bundles as a result of the increased flow are analyzed in order to ensure that the design limits will not be exceeded.

The thermal-hydraulic stability is re-evaluated for increased core flow operation, and the effects of flow-induced vibration are also evaluated to assure that the vibration criteria will not be exceeded.

Feedwater Temperature Reduction

Analyses are performed in order to justify operation at a reduced feedwater temperature at rated thermal power within the MELLLA domain. Usually, the analyses are performed for end-of-cycle operation with the last-stage feedwater heaters valved out of service. However, throughout cycle operation, an additional feedwater temperature reduction can be justified by analyses at the appropriate operating conditions.

The limiting AOTs are reanalyzed for operation at a reduced feedwater temperature. In addition, the loss-of-coolant accident (LOCA), fuel loading error, rod drop

accident, and rod withdrawal error are also re-evaluated for operation at a reduced feedwater temperature.

Turbine Bypass Out of Service

Operation of the turbine bypass system is assumed in the analysis of the Feedwater Controller Failure (FWCF)-maximum demand event. If this event is limiting or near limiting, the operating limit MCPR basis may be invalid if the bypass system cannot be demonstrated as fully functional. Reload specific evaluations may incorporate a FWCF without credit for bypass operation calculation as a provision when temporary factors render the system unavailable. Additionally, for extended operation with degraded bypass system operation, evaluations in support of this condition are augmented with the appropriate limiting events, such as the FWCF, for the applicable cycle.

ARTS Program

The ARTS program is a comprehensive project involving the Average Power Range Monitor (APRM), the Rod Block Monitor (RBM), and Technical Specification improvements.

Implementing the ARTS program provides for the following improvements which enhance the flexibility of the BWR during power level monitoring.

(1) The Average Power Range Monitor (APRM) trip setdown requirement is replaced by power-dependent and flow-dependent MCPR operating limits to reduce the need for manual setpoint adjustments. In addition, another set of power- and flow-dependent limits (LHGR and/or MAPLHGR) are also specified for more rigorous fuel thermal protection during postulated transients at off-rated conditions. These power- and flow-dependent limits are verified for plant-specific application during the initial ARTS licensing implementation. For AREVA reload analyses, the power-and flow-dependent limits are reviewed and updated as needed each cycle.

(2) The RBM system is modified from flow-biased to power-dependent trips to allow the use of a new generic non-limiting analysis for the Rod Withdrawal Error (RWE) and to improve response predictability to reduce the frequency of nonessential alarms. For AREVA reload analyses, the RWE analysis with ARTS is re-evaluated each cycle.

The resulting improvements in the flexibility of the BWR provided by ARTS are designed to significantly minimize the time to achieve full power from startup conditions.

Recirculation Coolant Pump Out of Service

The plant is licensed to allow extended Single-Loop Operation (SLO) in the MELLLA domain. The capability of operating at reduced power with a single recirculation loop is highly desirable, from a plant availability/outage planning standpoint, in the event maintenance of a recirculation pump or other components renders one loop inoperative. SLO analyses evaluate the plant for continuous operation at a maximum expected power output.

To justify SLO, safety analyses have to be reviewed for one-pump operation. The MCPR fuel cladding integrity safety limit, AOT analyses, operating limit MCPR, and non-LOCA accidents are evaluated. Increased uncertainties in the total core flow and traversing incore probe (TIP) readings result in a small increase in the fuel cladding integrity safety limit MCPR.

SLO can also result in changes to plant response during a LOCA. These changes are accommodated by the application of reduction factors to the two-loop operation LHGR and/or MAPLHGR limits, if required. Reduction factors are evaluated on a plant and fuel type dependent basis. In each subsequent reload, reduction factors are checked for validity and, if new fuel types are added, new reduction factors may be needed in order to maintain the validity of the SLO analysis.

End-of-Cycle Recirculation Pump Trip Out of Service

The EOC-RPT-OOS contingency mode of operation eliminates the automatic recirculation pump trip signal when turbine trip or load rejection occurs. As such, the core flow decreases at a slower rate following the recirculation pump trip due to the anticipated transient without scram (ATWS) high pressure recirculation system trip, thus, increasing the severity of the transient responses.

Power Load Unbalance Out of Service

The load rejection event scenario depends on whether the initial power level is sufficient for the PLU feature to operate. The PLU causes fast closure of the TCV. If the PLU does not operate as the result of a load rejection, the TCV closes at the maximum demand rate for speed control (servo mode). For the PLUOOS analysis, the PLU is assumed inoperable at any power level and does not initiate fast TCV closure.

Maximum Extended Load Line Limit Plus (MELLLA+)

MELLLA+ expands the licensed operating domain to allow operation above the MELLLA upper boundary at core flows greater than or equal to 55% of rated core flow with reactor power up to 100% of rated thermal power. The MELLLA+ region upper boundary passes through 100% power (3952 MWt) at 85% core flow. The

MELLLA+ upper boundary line terminates at 55% core flow. Final Feedwater Temperature Reduction (greater than 10ºF from design feedwater temperature) and Single Loop Operation are not allowed in the MELLLA+ domain. Addition of the MELLLA+ region provides improved power ascension capability to full power and additional flow range at rated power. Evaluations performed for MELLLA+ conditions include normal and AOTs, LOCA analysis, containment responses, Anticipate Transient without SCRAM (ATWS) analysis, reactor and reactor internals structural analyses, fluence analyses, and stability. The fuel dependent results of these analyses are re-evaluated each cycle.

3.7.7.3 Core Hydraulics

Core hydraulics models and correlations are discussed in Subsection 3.7.5, “Description of Thermal-Hydraulic Design of the Reactor Core”.

3.7.7.4 Influence of Power Distributions

The influence of power distributions on the thermal-hydraulic design is discussed in Reference 41.

3.7.7.5 Core Thermal Response

The thermal response of the core for accidents and expected AOT conditions is given in Chapter 14, “Plant Safety Analysis”.

3.7.7.6 Analytical Methods

The analytical methods, thermodynamic data, and hydrodynamic data used in determining the thermal and hydraulic characteristics of the core are documented in Subsection 3.7.7.1.2, “MCPR Operating Limit Calculational Procedure.”

3.7.8 References

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