ATTACHMENT 5

Retyped Proposed Technical Specification Bases Changes

BASES

APPLICABLE SAFETY ANALYSES (continued)

2.1.1.1 Fuel Cladding Integrity

The SPCB critical power correlation is used for both AREVA and coresident fuel and is valid at pressure $\geq$700 psia, and bundle mass fluxes $\geq$0.1 x 10^6 lbm/hr-ft^2 ($\geq$12,000 lb_m/hr, i.e., $\geq$10% core flow, on a per bundle basis) for ATRIUM-10 and GE14 fuel types. The thermal margin monitoring at 23% power and higher, the hot channel flow rate will be >28,000 lb_m/hr (core flow not less than natural circulation, i.e., ~25%-30% core flow for 23% power); therefore the fuel cladding integrity SL is conservative relative to the applicable range of the SPCB critical power correlation. For operation at low pressures or low flows, another basis is used, as follows:

The static head across the fuel bundles due only to elevation effects from liquid only in the channel, core bypass region, and annulus at zero power, zero flow is approximately 4.5 psi. At all operating conditions, this pressure differential is maintained by the bypass region of the core and the annulus region of the vessel. The elevation head provided by the annulus produces natural circulation flow conditions which have balancing pressure head and loss terms inside the core shroud. This natural circulation principle maintains a core plenum to plenum pressure drop of about 4.5 to 5 psid along the natural circulation flow line of the P/F operating map. In the range of power levels of interest, approaching 23% of rated power below which thermal margin monitoring is not required, the pressure drop and density head terms tradeoff for power changes such that natural circulation flow is nearly independent of reactor power. This characteristic is represented by the nearly vertical portion of the natural circulation line on the P/F operating map. Analysis has shown that the hot channel flow rate is >28,000 lb_m/hr (>0.23 x 10^6 lb_m/hr-ft^2) in the region of operation with power ~23% and core pressure drop of about 4.5 to 5 psid. Full scale ATLAS test data taken at pressures from 14.7 psia to 800 psia indicate that the fuel assembly critical power at 28,000 lb_m/hr is approximately 3 MW_t. With the design peaking factors, this corresponds to a core thermal power of more than 50%.

(continued)

BASES

APPLICABLE SAFETY ANALYSES

2.1.1.1 Fuel Cladding Integrity (continued)

Thus operation up to 23% of rated power with normal natural circulation available is conservatively acceptable even if reactor pressure is equal to or below 800 psia. If reactor power is significantly less than 23% of rated (e.g., below 10% of rated), the core flow and the channel flow supported by the available driving head may be less than 28,000 lb_m/hr (along the lower portion of the natural circulation flow characteristic on the P/F map). However, the critical power that can be supported by the core and hot channel flow with normal natural circulation paths available remains well above the actual power conditions. The inherent characteristics of BWR natural circulation make power and core flow follow the natural circulation line as long as normal water level is maintained.

Thus, operation with core thermal power below 23% of rated without thermal margin surveillance is conservatively acceptable even for reactor operations at natural circulation. Adequate fuel thermal margins are also maintained without further surveillance for the low power conditions that would be present if core natural circulation is below 10% of rated flow.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.1.7.4 and SR 3.1.7.6

SR 3.1.7.4 requires an examination of the sodium pentaborate solution by using chemical analysis to ensure that the proper concentration of boron exists in the storage tank. The concentration is dependent upon the volume of water and quantity of boron in the storage tank. SR 3.1.7.6 requires verification that the SLC system conditions satisfy the following equation:

$$\frac{(\ C\ )(\ Q\ )(\ E\ )}{(\ 8.7\ \mathrm{WT}\ \%\ )(\ 50\ \mathrm{GPM}\ )(\ 94\ \mathrm{ATOM}\ \%\ )} >= 1.0$$

C = sodium pentaborate solution weight percent concentration
Q = SLC system pump flow rate in gpm
E = Boron-10 atom percent enrichment in the sodium pentaborate solution

To meet 10 CFR 50.62, the SLC System must have a minimum flow capacity and boron content equivalent in control capacity to 86 gpm of 13 weight percent natural sodium pentaborate solution. The atom percentage of natural B-10 is 19.8%. This equivalency requirement is met when the equation given above is satisfied. The equation can be satisfied by adjusting the solution concentration, pump flow rate or Boron-10 enrichment. If the results of the equation are < 1, the SLC System is no longer capable of shutting down the reactor with the margin described in Reference 2. As described in Reference 2, the BFN analysis assumes a flow capacity and boron content equivalent to 50 gpm of 8.7 weight percent and 94 atom percent B-10 enriched sodium pentaborate solution. This exceeds the requirement of 10 CFR 50.62, and the equation is adjusted to reflect the BFN requirements. The quantity of stored boron includes an additional margin (25%) beyond the amount needed to shut down the reactor to allow for possible imperfect mixing of the chemical solution in the reactor water, leakage, and the volume in other piping connected to the reactor system.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.7.4 and SR 3.1.7.6 (continued)

The sodium pentaborate solution (SPB) concentration is allowed to be > 9.2 weight percent provided the concentration and temperature of the sodium pentaborate solution are verified to be within the limits of Figure 3.1.7-1. This ensures that unwanted precipitation of the sodium pentaborate does not occur.

SR 3.1.7.4 and SR 3.1.7.6 must be performed every 31 days or within 24 hours of when boron or water is added to the storage tank solution to determine that the boron solution concentration is within the specified limits. The 31 day Frequency of these Surveillances is appropriate because of the relatively slow variation of boron concentration between surveillances.

SR 3.1.7.4 must be performed within 8 hours of discovery that the concentration is > 9.2 weight percent and every 12 hours thereafter until the concentration is verified to be ≤ 9.2 weight percent. This Frequency is appropriate under these conditions taking into consideration the SLC System design capability still exists for vessel injection under these conditions and the low probability of the temperature and concentration limits of Figure 3.1.7-1 not being met.

SR 3.1.7.5

This Surveillance requires the amount of Boron-10 in the SLC solution tank to be determined every 31 days. The enriched sodium pentaborate solution is made by combining stoichiometric quantities of borax and boric acid in demineralized water. Since the chemicals used have known Boron-10 quantities, the Boron-10 quantity in the sodium pentaborate solution formed can be calculated. This parameter is used as input to determine the volume requirements for reactivity control encompassed by SR 3.1.7.1. The 31 day Frequency of this Surveillance is appropriate because of the relatively slow variation of boron concentration between surveillances.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.1.7.11

SR 3.1.7.11 verifies that each valve in the system is in its correct position, but does not apply to the squib (i.e., explosive) valves. Verifying the correct alignment for manual, power operated, and automatic valves in the SLC System Flowpath provides assurance that the proper flow paths will exist for system operation. A valve is also allowed to be in the nonaccident position provided it can be aligned to the accident position from the control room, or locally by a dedicated operator at the valve control. This is acceptable since the SLC System is a manually initiated system. This surveillance also does not apply to valves that are locked, sealed, or otherwise secured in position since they are verified to be in the correct position prior to locking, sealing or securing. This verification of valve alignment does not require any testing or valve manipulation; rather, it involves verification that those valves capable of being mispositioned are in the correct position. This SR does not apply to valves that cannot be inadvertently misaligned, such as check valves. The 31 day Frequency is based on engineering judgment and is consistent with the procedural controls governing valve operation that ensures correct valve positions.

REFERENCES

1. 10 CFR 50.62.
2. NEDC-33860P, “Safety Analysis Report for Browns Ferry Nuclear Plant Units 1, 2, and 3 Extended Power Uprate,” Section 2.8.
3. NRC No. 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.
4. FSAR, Section 14.6.

BASES

LCO

The APLHGR limits specified in the COLR are the result of the fuel design, DBA, and transient analyses. With only one recirculation loop in operation, in conformance with the requirements of LCO 3.4.1, "Recirculation Loops Operating," the limit is determined by multiplying the exposure dependent limit by an APLHGR correction factor (Ref. 5 and Ref. 10). Cycle specific APLHGR correction factors for single recirculation loop operation are documented in the COLR. APLHGR limits are selected such that no power or flow dependent corrections are required. Additional APLHGR operating limit adjustments may be provided in the COLR supporting other analyzed equipment out-of-service conditions.

APPLICABILITY

The APLHGR limits are primarily derived from fuel design evaluations and LOCA and transient analyses that are assumed to occur at high power levels. Design calculations (Ref. 4) and operating experience have shown that as power is reduced, the margin to the required APLHGR limits increases. This trend continues down to the power range of 5% to 15% RTP when entry into MODE 2 occurs. When in MODE 2, the intermediate range monitor scram function provides prompt scram initiation during any significant transient, thereby effectively removing any APLHGR limit compliance concern in MODE 2. Therefore, at THERMAL POWER levels ≤ 23% RTP, the reactor is

(continued)

BASES

APPLICABILITY (continued)

operating with substantial margin to the APLHGR limits; thus, this LCO is not required.

ACTIONS

A.1

If any APLHGR exceeds the required limits, an assumption regarding an initial condition of the DBA and transient analyses may not be met. Therefore, prompt action should be taken to restore the APLHGR(s) to within the required limits such that the plant operates within analyzed conditions and within design limits of the fuel rods. The 2 hour Completion Time is sufficient to restore the APLHGR(s) to within its limits and is acceptable based on the low probability of a transient or DBA occurring simultaneously with the APLHGR out of specification.

B.1

If the APLHGR cannot be restored to within its required limits within the associated Completion Time, the plant must be brought to a MODE or other specified condition in which the LCO does not apply. To achieve this status, THERMAL POWER must be reduced to < 23% RTP within 4 hours. The allowed Completion Time is reasonable, based on operating experience, to reduce THERMAL POWER to < 23% RTP in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.2.1.1

APLHGRs are required to be initially calculated within 12 hours after THERMAL POWER is ≥ 23% RTP and then every 24 hours thereafter. They are compared to the specified limits in the COLR to ensure that the reactor is operating within the assumptions of the safety analysis. The 24 hour Frequency is based on both engineering judgment and recognition of the slowness of changes in power distribution during normal

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.2.1.1 (continued)

operation. The 12 hour allowance after THERMAL POWER ≥ 23% RTP is achieved is acceptable given the large inherent margin to operating limits at low power levels.

REFERENCES

1. NEDE-24011-P-A, Rev. 16, "General Electric Standard Application for Reactor Fuel," October 2007.
2. FSAR, Chapter 3.
3. FSAR, Chapter 14.
4. FSAR, Appendix N.
5. NEDC-32484P, "Browns Ferry Nuclear Plant Units 1, 2, and 3, SAFER/GESTR-LOCA Loss-of-Coolant Accident Analysis," Revision 2, December 1997.
6. NRC No. 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.
7. NEDC-32433P, “Maximum Extended Load Line Limit and ARTS Improvement Program Analyses for Browns Ferry Nuclear Plant Units 1, 2, and 3,” April 1995.
8. NEDO-30130-A, “Steady State Nuclear Methods,” May 1985.
9. NEDO-24154, “Qualification of the One-Dimensional Core Transient Model for Boiling Water Reactors,” October 1978.
10. NEDO-24236, “Browns Ferry Nuclear Plant Units 1, 2, and 3, Single-Loop Operation,” May 1981.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

The analytical methods and assumptions used in evaluating the abnormal operational transients to establish the operating limit MCPR are presented in References 2, 3, 4, 5, 8, 10, 11, 12, 13, 14, and 15. To ensure that the MCPR SL is not exceeded during any transient event that occurs with moderate frequency, limiting transients have been analyzed to determine the largest reduction in critical power ratio (CPR). The types of transients evaluated are loss of flow, increase in pressure and power, positive reactivity insertion, and coolant temperature decrease. The limiting transient yields the largest change in CPR (ΔCPR). When the largest ΔCPR is added to the MCPR SL, the required operating limit MCPR is obtained.

The MCPR operating limits derived from the transient analysis are dependent on the operating core flow and power to ensure adherence to fuel design limits during the worst transient that occurs with moderate frequency (Reference 8). Flow dependent MCPR ($MCPR_f$) limits are determined by steady state thermal hydraulic methods using the three dimensional BWR simulator code (Ref. 12) and the multichannel thermal hydraulics code (Ref. 13). The operating limit is dependent on the maximum core flow limiter setting in the Recirculation Flow Control System.

Power dependent MCPR limits ($MCPR_p$) are determined by the three-dimensional BWR simulator code (Ref. 12) and the one-dimensional transient codes (Refs. 14 and 15). Due to the sensitivity of the transient response to initial core flow levels at power levels below those at which the turbine control valve fast closure scrams are bypassed, high and low flow $MCPR_p$ operating limits are provided for operating between 23% RTP and the previously mentioned bypass power level.

The MCPR satisfies Criterion 2 of the NRC Policy Statement (Ref. 7).

(continued)

BASES (continued)

LCO

The MCPR operating limits specified in the COLR are the result of the Design Basis Accident (DBA) and transient analysis. Additionally MCPR operating limits supporting analyzed equipment out-of-service conditions are provided in the COLR. The operating limit MCPR is determined by the larger of the $MCPR_f$ and $MCPR_p$ limits.

APPLICABILITY

The MCPR operating limits are primarily derived from transient analyses that are assumed to occur at high power levels. Below 23% RTP, the reactor is operating at a minimum recirculation pump speed and the moderator void ratio is small. Surveillance of thermal limits below 23% RTP is unnecessary due to the large inherent margin that ensures that the MCPR SL is not exceeded even if a limiting transient occurs. Statistical analyses indicate that the nominal value of the initial MCPR expected at 23% RTP is > 3.5. Studies of the variation of limiting transient behavior have been performed over the range of power and flow conditions. These studies encompass the range of key actual plant parameter values important to typically limiting transients. The results of these studies demonstrate that a margin is expected between performance and the MCPR requirements, and that margins increase as power is reduced to 23% RTP. This trend is expected to continue to the 5% to 15% power range when entry into MODE 2 occurs. When in MODE 2, the intermediate range monitor provides rapid scram initiation for any significant power increase transient, which effectively eliminates any MCPR compliance concern. Therefore, at THERMAL POWER levels < 23% RTP, the reactor is operating with substantial margin to the MCPR limits and this LCO is not required.

ACTIONS

A.1

If any MCPR is outside the required limits, an assumption regarding an initial condition of the design basis transient

(continued)

BASES

ACTIONS

A.1 (continued)

analyses may not be met. Therefore, prompt action should be taken to restore the MCPR(s) to within the required limits such that the plant remains operating within analyzed conditions. The 2 hour Completion Time is normally sufficient to restore the MCPR(s) to within its limits and is acceptable based on the low probability of a transient or DBA occurring simultaneously with the MCPR out of specification.

B.1

If the MCPR cannot be restored to within its required limits within the associated Completion Time, the plant must be brought to a MODE or other specified condition in which the LCO does not apply. To achieve this status, THERMAL POWER must be reduced to < 23% RTP within 4 hours. The allowed Completion Time is reasonable, based on operating experience, to reduce THERMAL POWER to < 23% RTP in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.2.2.1

The MCPR is required to be initially calculated within 12 hours after THERMAL POWER is ≥ 23% RTP and then every 24 hours thereafter. It is compared to the specified limits in the COLR to ensure that the reactor is operating within the assumptions of the safety analysis. The 24 hour Frequency is based on both engineering judgment and recognition of the slowness of changes in power distribution during normal operation. The 12 hour allowance after THERMAL POWER ≥ 23% RTP is achieved is acceptable given the large inherent margin to operating limits at low power levels.

(continued)

BASES (continued)

LCO (continued)	Additional LHGR operating limits adjustments may be provided the COLR to support analyzed equipment out-of-service operation.
APPLICABILITY	The LHGR limits are derived from fuel design analysis that is limiting at high power level conditions. At core thermal power levels < 23% RTP, the reactor is operating with a substantial margin to the LHGR limits and, therefore, the Specification is only required when the reactor is operating at ≥ 23% RTP.

(continued)

BASES (continued)

ACTIONS

A.1

If any LHGR exceeds its required limit, an assumption regarding an initial condition of the fuel design analysis is not met. Therefore, prompt action should be taken to restore the LHGR(s) to within its required limits such that the plant is operating within analyzed conditions. The 2 hour Completion Time is normally sufficient to restore the LHGR(s) to within its limits and is acceptable based on the low probability of a transient or Design Basis Accident occurring simultaneously with the LHGR out of specification.

B.1

If the LHGR cannot be restored to within its required limits within the associated Completion Time, the plant must be brought to a MODE or other specified condition in which the LCO does not apply. To achieve this status, THERMAL POWER is reduced to < 23% RTP within 4 hours. The allowed Completion Time is reasonable, based on operating experience, to reduce THERMAL POWER to < 23% RTP in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.2.3.1

The LHGR is required to be initially calculated within 12 hours after THERMAL POWER is ≥ 23% RTP and then every 24 hours thereafter. It is compared to the specified limits in the COLR to ensure that the reactor is operating within the assumptions of the safety analysis. The 24 hour Frequency is based on both engineering judgment and recognition of the slow changes in power distribution during normal operation. The 12 hour allowance after THERMAL POWER ≥ 23% RTP is achieved is acceptable given the large inherent margin to operating limits at lower power levels.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)	The trip setpoints are then determined accounting for the remaining instrument errors (e.g., drift). The trip setpoints derived in this manner provide adequate protection because instrumentation uncertainties, process effects, calibration tolerances, instrument drift, and severe environmental effects (for channels that must function in harsh environments as defined by 10 CFR 50.49) are accounted for.

The OPERABILITY of scram pilot valves and associated solenoids, backup scram valves, and SDV valves, described in the Background section, are not addressed by this LCO.

The individual Functions are required to be OPERABLE in the MODES or other specified conditions in the Table, which may require an RPS trip to mitigate the consequences of a design basis accident or transient. To ensure a reliable scram function, a combination of Functions are required in each MODE to provide primary and diverse initiation signals.

The only MODES specified in Table 3.3.1.1-1 are MODES 1 (which encompasses $\geq$ 26% RTP) and 2, and MODE 5 with any control rod withdrawn from a core cell containing one or more fuel assemblies. No RPS Function is required in MODES 3 and 4 since all control rods are fully inserted and the Reactor Mode Switch Shutdown Position control rod withdrawal block (LCO 3.3.2.1) does not allow any control rod to be withdrawn. In MODE 5, control rods withdrawn from a core cell containing no fuel assemblies do not affect the reactivity of the core and, therefore, are not required to have the capability to scram. Provided all other control rods remain inserted, no RPS function is required. In this condition, the required SDM (LCO 3.1.1) and refuel position one-rod-out interlock (LCO 3.9.2) ensure that no event requiring RPS will occur.

The specific Applicable Safety Analyses, LCO, and Applicability discussions are listed below on a Function by Function basis.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

Average Power Range Monitor

2.a. Average Power Range Monitor Neutron Flux - High, (Setdown)

For operation at low power (i.e., MODE 2), the Average Power Range Monitor Neutron Flux - High, (Setdown) Function is capable of generating a trip signal that prevents fuel damage resulting from abnormal operating transients in this power range. For most operation at low power levels, the Average Power Range Monitor Neutron Flux - High, (Setdown) Function will provide a secondary scram to the Intermediate Range Monitor Neutron Flux - High Function because of the relative setpoints. With the IRMs at Range 9 or 10, it is possible that the Average Power Range Monitor Neutron Flux - High, (Setdown) Function will provide the primary trip signal for a corewide increase in power.

No specific safety analyses take direct credit for the Average Power Range Monitor Neutron Flux - High, (Setdown) Function. However, this Function indirectly ensures that before the reactor mode switch is placed in the run position, reactor power does not exceed 23% RTP (SL 2.1.1.1) when operating at low reactor pressure and low core flow. Therefore, it indirectly prevents fuel damage during significant reactivity increases with THERMAL POWER < 23% RTP.

The Allowable Value is based on preventing significant increases in power when THERMAL POWER is < 23% RTP.

The Average Power Range Monitor Neutron Flux - High, Setdown Function must be OPERABLE during MODE 2 when control rods may be withdrawn since the potential for criticality exists. In MODE 1, the Average Power Range Monitor Neutron Flux - High Function provides protection against reactivity transients and the RWM and rod block monitor protect against control rod withdrawal error events.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

2.f. Oscillation Power Range Monitor (OPRM) Upscale

The OPRM Upscale Function provides compliance with GDC 10 and GDC 12, thereby providing protection from exceeding the fuel MCPR safety limit (SL) due to anticipated thermal hydraulic power oscillations.

References 13, 14, and 15 describe three algorithms for detecting thermal hydraulic instability related neutron flux oscillations: the period based detection algorithm, the amplitude based algorithm, and the growth rate algorithm. All three are implemented in the OPRM Upscale Function, but the safety analysis takes credit only for the period based detection algorithm. The remaining algorithms provide defense in depth and additional protection against unanticipated oscillations. OPRM Upscale Function OPERABILITY for Technical Specification purposes is based only on the period based detection algorithm.

The OPRM Upscale Function receives input signals from the local power range monitors (LPRMs) within the reactor core, which are combined into "cells" for evaluation of the OPRM algorithms.

The OPRM Upscale Function is required to be OPERABLE when the plant is in a region of power flow operation where anticipated events could lead to thermal hydraulic instability and related neutron flux oscillations. Within this region, the automatic trip is enabled when THERMAL POWER, as indicated by the APRM Simulated Thermal Power, is ≥ 23% RTP and reactor core flow, as indicated by recirculation drive flow is < 60% of rated flow, the operating region where actual thermal hydraulic oscillations may occur. Requiring the OPRM Upscale Function to be OPERABLE in MODE 1 provides consistency with operability requirements for other APRM functions and assures that the OPRM Upscale Function is OPERABLE whenever reactor power could increase into the region of concern without operator action.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

8. Turbine Stop Valve - Closure (continued)

Turbine Stop Valve - Closure signals are initiated from position switches located on each of the four TSVs. Two independent position switches are associated with each stop valve. One of the two switches provides input to RPS trip system A; the other, to RPS trip system B. Thus, each RPS trip system receives an input from four Turbine Stop Valve - Closure channels, each consisting of one position switch. The logic for the Turbine Stop Valve - Closure Function is such that three or more TSVs must be closed to produce a scram. This Function must be enabled at THERMAL POWER ≥ 26% RTP. This is normally accomplished automatically by pressure transmitters sensing turbine first stage pressure; therefore, opening the turbine bypass valves may affect this function.

The Turbine Stop Valve - Closure Allowable Value is selected to be high enough to detect imminent TSV closure, thereby reducing the severity of the subsequent pressure transient.

Eight channels of Turbine Stop Valve - Closure Function, with four channels in each trip system, are required to be OPERABLE to ensure that no single instrument failure will preclude a scram from this Function if any three TSVs should close. This Function is required, consistent with analysis assumptions, whenever THERMAL POWER is ≥ 26% RTP. This Function is not required when THERMAL POWER is < 26% RTP since the Reactor Vessel Steam Dome Pressure - High and the Average Power Range Monitor Fixed Neutron Flux - High Functions are adequate to maintain the necessary safety margins.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

9. Turbine Control Valve Fast Closure, Trip Oil Pressure - Low (PS-47-142, PS-47-144, PS-47-146, and PS-47-148)

Fast closure of the TCVs results in the loss of a heat sink that produces reactor pressure, neutron flux, and heat flux transients that must be limited. Therefore, a reactor scram is initiated on TCV fast closure in anticipation of the transients that would result from the closure of these valves. The Turbine Control Valve Fast Closure, Trip Oil Pressure - Low Function is the primary scram signal for the generator load rejection event analyzed in Reference 7. For this event, the reactor scram reduces the amount of energy required to be absorbed and, along with the actions of the EOC-RPT System, ensures that the MCPR SL is not exceeded.

Turbine Control Valve Fast Closure, Trip Oil Pressure - Low signals are initiated by the electrohydraulic control (EHC) fluid pressure at each control valve. One pressure switch is associated with each control valve, and the signal from each switch is assigned to a separate RPS logic channel. This Function must be enabled at THERMAL POWER ≥ 26% RTP. This is normally accomplished automatically by pressure transmitters sensing turbine first stage pressure; therefore, opening the turbine bypass valves may affect this function.

The Turbine Control Valve Fast Closure, Trip Oil Pressure - Low Allowable Value is selected high enough to detect imminent TCV fast closure.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

9. Turbine Control Valve Fast Closure, Trip Oil Pressure - Low (PS-47-142, PS-47-144, PS-47-146, and PS-47-148) (continued)

Four channels of Turbine Control Valve Fast Closure, Trip Oil Pressure - Low Function with two channels in each trip system arranged in a one-out-of-two logic are required to be OPERABLE to ensure that no single instrument failure will preclude a scram from this Function on a valid signal. This Function is required, consistent with the analysis assumptions, whenever THERMAL POWER is ≥ 26% RTP. This Function is not required when THERMAL POWER is < 26% RTP, since the Reactor Vessel Steam Dome Pressure - High and the Average Power Range Monitor Fixed Neutron Flux - High Functions are adequate to maintain the necessary safety margins.

For this instrument function, the nominal trip setpoint including the as-left tolerances is defined as the LSSS. The acceptable as-found band is based on a statistical combination of possible measurable uncertainties (i.e., setting tolerance, drift, temperature effects, and measurement and test equipment). During instrument calibrations, if the as-found setpoint is found to be conservative with respect to the Allowable Value, but outside its acceptable as-found band (tolerance range), as defined by its associated Surveillance Requirement procedure, then there shall be an initial determination to ensure confidence that the channel can perform as required before returning the channel to service in accordance with the Surveillance. The technician performing the Surveillance will evaluate the instrument's ability to maintain a stable setpoint within the as-left tolerance. The technician's evaluation will be reviewed by on shift personnel during the approval of the Surveillance data prior to returning the channel back to service at the completion of the Surveillance. This shall constitute the initial determination of operability. If a channel is found to exceed the channel's Allowable Value or cannot be reset within the

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.1.1.2

To ensure that the APRMs are accurately indicating the true core average power, the APRMs are calibrated to the reactor power calculated from a heat balance. The Frequency of once per 7 days is based on minor changes in LPRM sensitivity, which could affect the APRM reading, between performances of SR 3.3.1.1.7.

A restriction to satisfying this SR when < 23% RTP is provided that requires the SR to be met only at ≥ 23% RTP because it is difficult to accurately maintain APRM indication of core THERMAL POWER consistent with a heat balance when < 23% RTP. At low power levels, a high degree of accuracy is unnecessary because of the large, inherent margin to thermal limits (MCPR and APLHGR). At ≥ 23% RTP, the Surveillance is required to have been satisfactorily performed within the last 7 days, in accordance with SR 3.0.2. A Note is provided which allows an increase in THERMAL POWER above 23% if the 7 day Frequency is not met per SR 3.0.2. In this event, the SR must be performed within 12 hours after reaching or exceeding 23% RTP. Twelve hours is based on operating experience and in consideration of providing a reasonable time in which to complete the SR.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.1.1.15

This SR ensures that scrams initiated from the Turbine Stop Valve - Closure and Turbine Control Valve Fast Closure, Trip Oil Pressure - Low Functions will not be inadvertently bypassed when THERMAL POWER is ≥ 26% RTP. This involves calibration of the bypass channels (PIS-1-81A, PIS-1-81B, PIS-1-91A, and PIS-1-91B). Adequate margins for the instrument setpoint methodologies are incorporated into the actual setpoint.

If any bypass channel's setpoint is nonconservative (i.e., the Functions are bypassed at ≥ 26% RTP, either due to open main turbine bypass valve(s) or other reasons), then the affected Turbine Stop Valve - Closure and Turbine Control Valve Fast Closure, Trip Oil Pressure - Low Functions are considered inoperable. Alternatively, the bypass channel can be placed in the conservative condition (nonbypass). If placed in the nonbypass condition (Turbine Stop Valve - Closure and Turbine Control Valve Fast Closure, Trip Oil Pressure - Low Functions are enabled), this SR is met and the channel is considered OPERABLE.

The Frequency of 24 months is based upon the assumption of a 24 month calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.1.1.17

This SR ensures that scrams initiated from OPRM Upscale Function (Function 2.f) will not be inadvertently bypassed when THERMAL POWER, as indicated by the APRM Simulated Thermal Power, is ≥ 23% RTP and core flow, as indicated by recirculation drive flow, is < 60% rated core flow. This normally involves confirming the bypass setpoints. Adequate margins for the instrument setpoint methodologies are incorporated into the actual setpoint. The actual surveillance ensures that the OPRM Upscale Function is enabled (not bypassed) for the correct values of APRM Simulated Thermal Power and recirculation drive flow. Other surveillances ensure that the APRM Simulated Thermal Power and recirculation flow properly correlate with THERMAL POWER and core flow, respectively.

If any bypass setpoint is nonconservative (i.e., the OPRM Upscale Function is bypassed when APRM Simulated Thermal Power ≥ 23% RTP and recirculation drive flow < 60% rated), then the affected channel is considered inoperable for the OPRM Upscale Function. Alternatively, the bypass setpoint may be adjusted to place the channel in a conservative condition (unbypass). If placed in the unbypassed condition, this SR is met and the channel is considered OPERABLE.

The frequency of 24 months is based on engineering judgment and reliability of the components.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

The feedwater and main turbine high water level trip instrumentation is assumed to be capable of providing a turbine trip in the design basis transient analysis for a feedwater controller failure, maximum demand event (Ref. 1). The reactor vessel high water level trip indirectly initiates a reactor scram from the main turbine trip (above 26% RTP) and trips the feedwater pumps, thereby terminating the event. The reactor scram mitigates the reduction in MCPR.

Feedwater and main turbine high water level trip instrumentation satisfies Criterion 3 of the NRC Policy Statement (Ref. 3).

LCO

The LCO requires two channels of the Reactor Vessel Water Level - High instrumentation per trip system to be OPERABLE to ensure that no single instrument failure will prevent the feedwater pump turbines and main turbine trip on a valid Reactor Vessel Water Level - High signal. Both channels in either trip system are needed to provide trip signals in order for the feedwater and main turbine trips to occur. Each channel must have its setpoint set within the specified Allowable Value of SR 3.3.2.2.3. The Allowable Value is set to ensure that the thermal limits are not exceeded during the event. The actual setpoint is calibrated to be consistent with the applicable setpoint methodology assumptions. Nominal trip setpoints are specified in the setpoint calculations. The nominal setpoints are selected to ensure that the setpoints do not exceed the Allowable Value between successive CHANNEL CALIBRATIONS. Operation with a trip setpoint less conservative than the nominal trip setpoint, but within its Allowable Value, is acceptable.

(continued)

BASES

LCO (continued)

Trip setpoints are those predetermined values of output at which an action should take place. The setpoints are compared to the actual process parameter (e.g., reactor vessel water level), and when the measured output value of the process parameter exceeds the setpoint, the associated device (e.g., trip unit) changes state. The analytic limits are derived from the limiting values of the process parameters obtained from the safety analysis. The Allowable Values are derived from the analytic limits, corrected for calibration, process, and some of the instrument errors. A channel is inoperable if its actual trip setpoint is not within its required Allowable Value. The trip setpoints are then determined accounting for the remaining instrument errors (e.g., drift). The trip setpoints derived in this manner provide adequate protection because instrumentation uncertainties, process effects, calibration tolerances, instrument drift, and severe environmental effects (for channels that must function in harsh environments as defined by 10 CFR 50.49) are accounted for.

APPLICABILITY

The feedwater and main turbine high water level trip instrumentation is required to be OPERABLE at ≥ 23% RTP to ensure that the fuel cladding integrity Safety Limit and the cladding 1% plastic strain limit are not violated during the feedwater controller failure, maximum demand event. As discussed in the Bases for LCO 3.2.1, "Average Planar Linear Heat Generation Rate (APLHGR)," and LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)," sufficient margin to these limits exists below 23% RTP; therefore, these requirements are only necessary when operating at or above this power level.

(continued)

BASES

ACTIONS

B.1 (continued)

The 2 hour Completion Time is sufficient for the operator to take corrective action, and takes into account the likelihood of an event requiring actuation of feedwater and main turbine high water level trip instrumentation occurring during this period. It is also consistent with the 2 hour Completion Time provided in LCO 3.2.2 for Required Action A.1, since this instrumentation's purpose is to preclude a MCPR violation.

C.1

With the required channels not restored to OPERABLE status or placed in trip, THERMAL POWER must be reduced to < 23% RTP within 4 hours. As discussed in the Applicability section of the Bases, operation below 23% RTP results in sufficient margin to the required limits, and the feedwater and main turbine high water level trip instrumentation is not required to protect fuel integrity during the feedwater controller failure, maximum demand event. The allowed Completion Time of 4 hours is based on operating experience to reduce THERMAL POWER to < 23% RTP from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

The Surveillances are modified by a Note to indicate that when a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided the associated Function maintains feedwater and main turbine high water level trip capability. Upon completion of the Surveillance, or expiration of the 6 hour allowance, the channel must be returned to OPERABLE status

(continued)

BASES

BACKGROUND (continued)

Each EOC-RPT trip system is a two-out-of-two logic for each Function; thus, either two TSV - Closure or two TCV Fast Closure, Trip Oil Pressure - Low signals are required for a trip system to actuate. If either trip system actuates, both recirculation pumps will trip. There are two EOC-RPT breakers in series per recirculation pump. One trip system trips one of the two EOC-RPT breakers for each recirculation pump, and the second trip system trips the other EOC-RPT breaker for each recirculation pump.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

The TSV - Closure and the TCV Fast Closure, Trip Oil Pressure - Low Functions are designed to trip the recirculation pumps in the event of a turbine trip or generator load rejection to mitigate the increase in neutron flux, heat flux, and reactor pressure, and to increase the margin to the MCPR SL, and LHGR limits. The analytical methods and assumptions used in evaluating the turbine trip and generator load rejection are summarized in References 2, 3, and 4.

To mitigate pressurization transient effects, the EOC-RPT must trip the recirculation pumps after initiation of closure movement of either the TSVs or the TCVs. The combined effects of this trip and a scram reduce fuel bundle power more rapidly than a scram alone, resulting in an increased margin to the MCPR SL, and LHGR limits. Alternatively, MCPR limits for an inoperable EOC-RPT, as specified in the COLR, are sufficient to prevent violation of the MCPR Safety Limit, and fuel mechanical limits. The EOC-RPT function is automatically disabled when turbine first stage pressure is < 26% RTP.

EOC-RPT instrumentation satisfies Criterion 3 of the NRC Policy Statement (Ref. 6).

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

Turbine Stop Valve - Closure (continued)

Closure of the TSVs is determined by measuring the position of each valve. There are two separate position signals associated with each stop valve, the signal from each switch being assigned to a separate trip channel. The logic for the TSV - Closure Function is such that two or more TSVs must be closed to produce an EOC-RPT. This Function must be enabled at THERMAL POWER ≥ 26% RTP. This is normally accomplished automatically by pressure transmitters sensing turbine first stage pressure; therefore, opening the turbine bypass valves may affect this function. To consider this function OPERABLE, bypass of the function must not occur when bypass valves are open. Four channels of TSV - Closure, with two channels in each trip system, are available and required to be OPERABLE to ensure that no single instrument failure will preclude an EOC-RPT from this Function on a valid signal. The TSV - Closure Allowable Value is selected to detect imminent TSV closure.

This protection is required, consistent with the safety analysis assumptions, whenever THERMAL POWER is ≥ 26% RTP. Below 26% RTP, the Reactor Vessel Steam Dome Pressure - High and the Average Power Range Monitor (APRM) Fixed Neutron Flux - High Functions of the Reactor Protection System (RPS) are adequate to maintain the necessary margin to the MCPR SL, and LHGR limits.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)	Turbine Control Valve Fast Closure, Trip Oil Pressure - Low (PS-47-142, PS-47-144, PS-47-146, and PS-47-148)

Fast closure of the TCVs during a generator load rejection results in the loss of a heat sink that produces reactor pressure, neutron flux, and heat flux transients that must be limited. Therefore, an RPT is initiated on TCV Fast Closure, Trip Oil Pressure - Low in anticipation of the transients that would result from the closure of these valves. The EOC-RPT decreases reactor power and aids the reactor scram in ensuring that the MCPR SL, and LHGR limits are not exceeded during the worst case transient.

Fast closure of the TCVs is determined by measuring the electrohydraulic control fluid pressure at each control valve. There is one pressure switch associated with each control valve, and the signal from each switch is assigned to a separate trip channel. The logic for the TCV Fast Closure, Trip Oil Pressure - Low Function is such that two or more TCVs must be closed (pressure switch trips) to produce an EOC-RPT. This Function must be enabled at THERMAL POWER ≥ 26% RTP. This is normally accomplished automatically by pressure transmitters sensing turbine first stage pressure; therefore, opening the turbine bypass valves may affect this function. To consider this function OPERABLE, bypass of the function must not occur when bypass valves are open. Four channels of TCV Fast Closure, Trip Oil Pressure - Low, with two channels in each trip system, are available and required to be OPERABLE to ensure that no single instrument failure will preclude an EOC-RPT from this Function on a valid signal. The TCV Fast Closure, Trip Oil Pressure - Low Allowable Value is selected high enough to detect imminent TCV fast closure.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

Turbine Control Valve Fast Closure, Trip Oil Pressure - Low (PS-47-142, PS-47-144, PS-47-146, and PS-47-148) (continued)

This protection is required consistent with the safety analysis whenever THERMAL POWER is ≥ 26% RTP. Below 26% RTP, the Reactor Vessel Steam Dome Pressure - High and the APRM Fixed Neutron Flux - High Functions of the RPS are adequate to maintain the necessary safety margins.

ACTIONS

A Note has been provided to modify the ACTIONS related to EOC-RPT instrumentation channels. Section 1.3, Completion Times, specifies that once a Condition has been entered, subsequent divisions, subsystems, components, or variables expressed in the Condition, discovered to be inoperable or not within limits, will not result in separate entry into the Condition. Section 1.3 also specifies that Required Actions of the Condition continue to apply for each additional failure, with Completion Times based on initial entry into the Condition. However, the Required Actions for inoperable EOC-RPT instrumentation channels provide appropriate compensatory measures for separate inoperable channels. As such, a Note has been provided that allows separate Condition entry for each inoperable EOC-RPT instrumentation channel.

(continued)

BASES

ACTIONS (continued)

B.1 and B.2

Required Actions B.1 and B.2 are intended to ensure that appropriate actions are taken if multiple, inoperable, untripped channels within the same Function result in the Function not maintaining EOC-RPT trip capability. A Function is considered to be maintaining EOC-RPT trip capability when sufficient channels are OPERABLE or in trip, such that the EOC-RPT System will generate a trip signal from the given Function on a valid signal and both recirculation pumps can be tripped. Alternately, Required Action B.2 requires the MCPR and LHGR limits for inoperable EOC-RPT, as specified in the COLR, to be applied. This also restores the margin to MCPR and LHGR limits assumed in the safety analysis.

The 2 hour Completion Time is sufficient time for the operator to take corrective action, and takes into account the likelihood of an event requiring actuation of the EOC-RPT instrumentation during this period. It is also consistent with the 2 hour Completion Time provided in LCO 3.2.2 for Required Action A.1, since this instrumentation's purpose is to preclude a MCPR or LHGR violation.

C.1

With any Required Action and associated Completion Time not met, THERMAL POWER must be reduced to < 26% RTP within 4 hours. The allowed Completion Time of 4 hours is reasonable, based on operating experience, to reduce THERMAL POWER to < 26% RTP from full power conditions in an orderly manner and without challenging plant systems.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.4.1.2

This SR ensures that an EOC-RPT initiated from the TSV - Closure and TCV Fast Closure, Trip Oil Pressure - Low Functions will not be inadvertently bypassed when THERMAL POWER is ≥ 26% RTP. This involves calibration of the bypass channels. Adequate margins for the instrument setpoint methodologies are incorporated into the actual setpoint. If any bypass channel's setpoint is nonconservative (i.e., the Functions are bypassed at ≥ 26% RTP, either due to open main turbine bypass valves or other reasons), the affected TSV - Closure and TCV Fast Closure, Trip Oil Pressure - Low Functions are considered inoperable. Alternatively, the bypass channel can be placed in the conservative condition (nonbypass). If placed in the nonbypass condition, this SR is met with the channel considered OPERABLE.

The Frequency of 24 months is based upon the assumption of a 24 month calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis.

SR 3.3.4.1.3

CHANNEL CALIBRATION is a complete check of the instrument loop and the sensor. This test verifies the channel responds to the measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drifts between successive calibrations consistent with the plant specific setpoint methodology. The Frequency is based upon the assumption of a 24 month calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis.

(continued)

BASES

SURVEILLANCE REQUIREMENTS	SR 3.4.2.1 (continued) Note 2 allows this SR not to be performed until 24 hours after THERMAL POWER exceeds 23% of RTP. During low flow conditions, jet pump noise approaches the threshold response of the associated flow instrumentation and precludes the collection of repeatable and meaningful data. The 24 hours is an acceptable time to establish conditions appropriate to perform this SR.
REFERENCES	1. FSAR, Section 14.6.3. 2. GE Service Information Letter No. 330, "Jet Pump Beam Cracks," June 9, 1980. 3. NUREG/CR-3052, "Closeout of IE Bulletin 80-07: BWR Jet Pump Assembly Failure," November 1984. 4. NRC No. 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.

BASES

APPLICABLE SAFETY ANALYSES (continued)

The maximum allowable leakage rate for the primary containment (L_a) is 2.0% by weight of the containment air per 24 hours at the design basis LOCA maximum peak containment pressure (P_a) of 49.1 psig (Ref. 1).

Primary containment satisfies Criterion 3 of the NRC Policy Statement (Ref. 6).

LCO

Primary containment OPERABILITY is maintained by limiting leakage to $\leq 1.0\ L_a$, except prior to the first startup after performing a required Primary Containment Leakage Rate Testing Program leakage test. At this time, applicable leakage limits must be met. Compliance with this LCO will ensure a primary containment configuration, including equipment hatches, that is structurally sound and that will limit leakage to those leakage rates assumed in the safety analyses.

Individual leakage rates specified for the primary containment air lock are addressed in LCO 3.6.1.2.

APPLICABILITY

In MODES 1, 2, and 3, a DBA could cause a release of radioactive material to primary containment. In MODES 4 and 5, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, primary containment is not required to be OPERABLE in MODES 4 and 5 to prevent leakage of radioactive material from primary containment.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.1.1.2 (continued)

Satisfactory performance of this SR can be achieved by establishing a known differential pressure between the drywell and the suppression chamber and verifying that the pressure in either the suppression chamber or the drywell does not change by more than 0.25 inch of water per minute over a 10 minute period. The leakage test is performed every 24 months. The 24 month Frequency was developed considering it is prudent that this Surveillance be performed during a unit outage and also in view of the fact that component failures that might have affected this test are identified by other primary containment SRs.

REFERENCES

1. NEDC-33860P, “Safety Analysis Report for Browns Ferry Nuclear Plant Units 1, 2, and 3 Extended Power Uprate,” Section 2.6.

2. FSAR, Section 14.6.

3. 10 CFR 50, Appendix J, Option B.

4. NEI 94-01, Revision O, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J."

5. ANSI/ANS-56.8-1994, "American National Standard for Containment System Leakage Testing Requirement."

6. NRC No. 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.

BASES

BACKGROUND (continued)

the air lock to remain open for extended periods when frequent primary containment entry is necessary. Under some conditions allowed by this LCO, the primary containment may be accessed through the air lock, when the interlock mechanism has failed, by manually performing the interlock function.

The primary containment air lock forms part of the primary containment pressure boundary. As such, air lock integrity and leak tightness are essential for maintaining primary containment leakage rate to within limits in the event of a DBA. Not maintaining air lock integrity or leak tightness may result in a leakage rate in excess of that assumed in the unit safety analysis.

APPLICABLE SAFETY ANALYSES

The DBA that postulates the maximum release of radioactive material within primary containment is a LOCA. In the analysis of this accident, it is assumed that primary containment is OPERABLE, such that release of fission products to the environment is controlled by the rate of primary containment leakage. The primary containment is designed with a maximum allowable leakage rate (L_a) of 2.0% by weight of the containment air per 24 hours at the calculated maximum peak containment pressure (P_a) of 49.1 psig (Ref. 3). This allowable leakage rate forms the basis for the acceptance criteria imposed on the SRs associated with the air lock.

Primary containment air lock OPERABILITY is also required to minimize the amount of fission product gases that may escape primary containment through the air lock and contaminate and pressurize the secondary containment.

The primary containment air lock satisfies Criterion 3 of the NRC Policy Statement (Ref. 4).

(continued)

B 3.6 CONTAINMENT SYSTEMS

B 3.6.3.1 Containment Atmosphere Dilution (CAD) System

BASES

BACKGROUND

The CAD System functions to maintain combustible gas concentrations within the primary containment at or below the flammability limits following a postulated loss of coolant accident (LOCA) by diluting hydrogen and oxygen with nitrogen. To ensure that a combustible gas mixture does not occur, oxygen concentration is kept < 5.0 volume percent (v/o), or hydrogen concentration is kept < 4.0 v/o.

The CAD System is manually initiated and consists of two independent, 100% capacity subsystems, each of which is capable of supplying nitrogen through separate piping systems to the drywell and suppression chamber of each unit. Each subsystem includes a liquid nitrogen supply tank, ambient vaporizer, electric heater, and a manifold with branches to each primary containment (for Units 1, 2, and 3). The nitrogen storage tanks each contain ≥ 2615 gal, which is adequate for 7 days of CAD subsystem operation (Ref. 4).

The CAD System operates in conjunction with emergency operating procedures that are used to reduce primary containment pressure periodically during CAD System operation. This combination results in a feed and bleed approach to maintaining hydrogen and oxygen concentrations below combustible levels.

(continued)

BASES

ACTIONS

B.1and B.2 (continued)

The Completion Time of 7 days is a reasonable time to allow continued reactor operation with two CAD subsystems inoperable because the hydrogen control function is maintained (via the Primary Containment Inerting System) and because of the low probability of the occurrence of a LOCA that would generate hydrogen in amounts capable of exceeding the flammability limit.

C.1

If any Required Action cannot be met within the associated Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours. The allowed Completion Time of 12 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.3.1.1

Verifying that there is ≥ 2615 gal of liquid nitrogen supply in each nitrogen storage tank will ensure at least 7 days of post-LOCA CAD operation. This minimum volume of liquid nitrogen represents the analytical limit assumed in the analysis of the primary containment atmosphere following a postulated LOCA and does not include allowance for potential nitrogen boiloff and tank level instrumentation inaccuracies. This minimum volume of liquid nitrogen allows sufficient time after an accident to replenish the nitrogen supply for long term inerting. This is verified every 31 days to ensure that the system is capable of performing its intended function when required.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.3.1.1 (continued)

The 31 day Frequency is based on operating experience, which has shown 31 days to be an acceptable period to verify the liquid nitrogen supply and on the availability of other hydrogen mitigating systems.

SR 3.6.3.1.2

Verifying the correct alignment for manual, power operated, and automatic valves in each of the CAD subsystem flow paths provides assurance that the proper flow paths exist for system operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since these valves were verified to be in the correct position prior to locking, sealing, or securing.

A valve is also allowed to be in the nonaccident position provided it can be aligned to the accident position within the time assumed in the accident analysis. This is acceptable because the CAD System is manually initiated. This SR does not apply to valves that cannot be inadvertently misaligned, such as check valves. This SR does not require any testing or valve manipulation; rather, it involves verification that those valves capable of being mispositioned are in the correct position.

The 31 day Frequency is appropriate because the valves are operated under procedural control, improper valve position would only affect a single subsystem, the probability of an event requiring initiation of the system is low, and the system is a manually initiated system.

(continued)

BASES (continued)

REFERENCES

1. AEC Safety Guide 7, Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident, March 10, 1971.

2. FSAR, Section 5.2.6.

3. NRC No. 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.

4. ANP-3403P, “Fuel Uprate Safety Analysis Report for Browns Ferry Nuclear Plant Units 1, 2, and 3,” Section 2.6.4, August 2015.

BASES

APPLICABLE SAFETY ANALYSES (continued)	With two and three units fueled, a worse case single failure could also include the loss of two RHRSW pumps caused by losing a 4 kV shutdown board since there are certain alignment configurations that allow two RHRSW pumps to be powered from the same 4 kV shutdown board. As discussed in the FSAR, Section 14.6.3.3.2 (Ref. 4) for these analyses, manual initiation of the OPERABLE RHRSW subsystems and the associated RHR System is assumed to occur 10 minutes after a DBA. The analyses assume that there are two RHRSW subsystems operating in each unit, with one RHRSW pump in each subsystem capable of producing 4000 gpm of flow. In this case, the maximum suppression chamber water temperature and pressure are 187.3°F and 49.1 psig, respectively, well below the design temperature of 281°F and maximum allowable pressure of 62 psig. The RHRSW System satisfies Criterion 3 of the NRC Policy Statement (Ref 5).
LCO	Four RHRSW subsystems are required to be OPERABLE to provide the required redundancy to ensure that the system functions to remove post accident heat loads, assuming the worst case single active failure occurs coincident with the loss of offsite power. Additionally, since the RHRSW pumps are shared between the three BFN units, the number of OPERABLE pumps required is also dependent on the number of units fueled. An RHRSW subsystem is considered OPERABLE when: a. At least one RHRSW pump (i.e., one required RHRSW pump) is OPERABLE; and b. An OPERABLE flow path is capable of taking suction from the intake structure and transferring the water to the associated RHR heat exchanger at the assumed flow rate.

(continued)

BASES (continued)

REFERENCES

1. FSAR, Section 10.9.

2. FSAR, Chapter 5.

3. FSAR, Chapter 14.

4. FSAR, Section 14.6.3.3.2.

5. NRC No. 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.

B 3.7 PLANT SYSTEMS

B 3.7.5 Main Turbine Bypass System

BASES

BACKGROUND	The Main Turbine Bypass System is designed to control steam pressure when reactor steam generation exceeds turbine requirements during unit startup, sudden load reduction, and cooldown. It allows excess steam flow from the reactor to the condenser without going through the turbine. The bypass capacity of the system is 23% of the Nuclear Steam Supply System rated steam flow. Sudden load reductions within the capacity of the steam bypass can be accommodated without reactor scram. The Main Turbine Bypass System consists of nine valves connected to the main steam lines between the main steam isolation valves and the turbine stop valve bypass valve chest. Each of these nine valves is operated by hydraulic cylinders. The bypass valves are controlled by the pressure regulation function of the Pressure Regulator and Turbine Generator Control System, as discussed in the FSAR, Section 7.11.3.3 (Ref. 1). The bypass valves are normally closed, and the pressure regulator controls the turbine control valves that direct all steam flow to the turbine. If the speed governor or the load limiter restricts steam flow to the turbine, the pressure regulator controls the system pressure by opening the bypass valves. When the bypass valves open, the steam flows from the bypass chest, through connecting piping, to the pressure breakdown assemblies, where a series of orifices are used to further reduce the steam pressure before the steam enters the condenser.

(continued)

BASES (continued)

APPLICABILITY

The Main Turbine Bypass System is required to be OPERABLE at ≥ 23% RTP to ensure that the fuel cladding integrity Safety Limit is not violated during abnormal operational transients. As discussed in the Bases for LCO 3.2.1 and LCO 3.2.2, sufficient margin to these limits exists at < 23% RTP. Therefore, these requirements are only necessary when operating at or above this power level.

ACTIONS

A.1

If the Main Turbine Bypass System is inoperable (one or more bypass valves inoperable), or the APLHGR, MCPR, and LHGR limits for an inoperable Main Turbine Bypass System, as specified in the COLR, are not applied, the assumptions of the design basis transient analysis may not be met. Under such circumstances, prompt action should be taken to restore the Main Turbine Bypass System to OPERABLE status or adjust the APLHGR, MCPR, and LHGR limits accordingly. The 2 hour Completion Time is reasonable, based on the time to complete the Required Action and the low probability of an event occurring during this period requiring the Main Turbine Bypass System.

B.1

If the Main Turbine Bypass System cannot be restored to OPERABLE status or the APLHGR, MCPR, and LHGR limits for an inoperable Main Turbine Bypass System are not applied, THERMAL POWER must be reduced to < 23% RTP. As discussed in the Applicability section, operation at < 23% RTP results in sufficient margin to the required limits, and the Main

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

2.1.1.1 Fuel Cladding Integrity

Critical power correlations are valid over a wide range of conditions per References 2 and 5, extending to expected conditions below 23% THERMAL POWER. For core thermal power levels at, or above 23% rated, the hot channel flow rate is expected to be >28,000 lbm/hr, (core flow not less than natural circulation i.e., ~25%-30 % core flow for 23% power); therefore, the fuel cladding integrity SL is conservative relative to the applicable range of the critical power correlations. For operation at low pressure/flow conditions, consistent with the low power region of the Power/Flow operating map, another basis is used as follows:

The static head across the fuel bundles is due to elevation effects from water solid channel, core bypass, and annulus regions, is approximately 4.5 psid. The pressure differential is maintained by the water solid bypass region of the core, along with the annulus region of the vessel. Elevation head provided by the bypass and annulus regions produces natural circulation flow conditions balancing pressure head with loss terms inside the core shroud.

Natural circulation principles maintain a core plenum to plenum pressure drop of approximately 4.5 to 5 psid along the natural circulation flow line of the Power/Flow operating map. When power levels approach 23% rated, pressure drop and density head terms are closely balanced as power changes, such that natural circulation flow is nearly independent of reactor power.

The flow characteristic is represented by the nearly vertical portion of the natural circulation line on the Power/Flow operating map. For a core pressure drop of approximately 4.5 to 5 psid, the hot channel flow rate is expected to be >28,000 lbm/hr in the region of operation when core power is ≤ 23% with a corresponding core pressure drop of about 4.5 to 5 psid.

(continued)

BASES

APPLICABLE SAFETY ANALYSES

2.1.1.1 Fuel Cladding Integrity (continued)

For example, Reference 5 test data, taken at low pressures and flow rates, indicate assembly critical power in excess of 4 MWt, for flow rates indicative of natural circulation conditions. At 23% rated power, assembly average power is $\leq$ 1.2 MWt. When considering design peaking factors, hot channel power could be expected to be on the order of 2 MWt. Consequently, operation up to 23% rated core power, with normal natural circulation available, is conservative even if reactor pressure is less than the lower pressure limit of the critical power correlation.

When reactor power is significantly less than 23% of rated (e.g., below 10% of rated), hot channel flow supported by the available driving head may fall below 28,000 lbm/hr (along the lower portion of the natural circulation flow characteristic on the Power/Flow map). However, the critical power supported by the flow, remains above actual hot channel power conditions. The inherent characteristics of BWR natural circulation make core power/flow follow the natural circulation line as long as normal annulus water level is maintained.

Operation below 23% rated core thermal power is conservatively acceptable, even for reactor operations at natural circulation. Adequate fuel thermal margins are maintained for low power conditions present during core natural circulation, even though the flow may be less than the critical power correlation applicability range.

(continued)

BASES

BACKGROUND (continued)

The worst case sodium pentaborate solution concentration required to shutdown the reactor with sufficient margin to account for 0.05 $\Delta k/k$ and Xenon poisoning effects is 9.2 weight percent. This corresponds to a 40°F saturation temperature. The worst case SLCS equipment area temperature is not predicted to fall below 50°F. This provides a 10°F thermal margin to unwanted precipitation of the sodium pentaborate. Tank heating components provide backup assurance that the sodium pentaborate solution temperature will never fall below 50°F but are not required for TS operability considerations.

APPLICABLE SAFETY ANALYSES

The SLC System is manually initiated from the main control room, as directed by the emergency operating instructions, if the operator believes the reactor cannot be shut down, or kept shut down, with the control rods. The SLC System is used in the event that enough control rods cannot be inserted to accomplish shutdown and cooldown in the normal manner. The SLC System injects borated water into the reactor core to add negative reactivity to compensate for all of the various reactivity effects that could occur during plant operations. To meet this objective, it is necessary to inject a quantity of boron, which produces a concentration of 720 ppm of natural boron, in the reactor coolant at 70°F. To allow for imperfect mixing, leakage and the volume in other piping connected to the reactor system, an amount of boron equal to 25% of the amount cited above is added (Ref. 2). This volume versus concentration limit and the temperature versus concentration limits in Figure 3.1.7-1 are calculated such that the required concentration is achieved accounting for dilution in the RPV with normal water level and including the water volume in the entire residual heat removal shutdown cooling piping and in the recirculation loop piping. This quantity of borated solution is the amount that is above the pump suction shutoff level in the boron solution storage tank. No credit is taken for the portion of the tank volume that cannot be injected.

(continued)

SURVEILLANCE REQUIREMENTS

SR 3.1.7.1 (continued)

pentaborate solution concentration requirements (≤ 9.2% by weight) and the required quantity of Boron-10 (≥ 203 lbs) establish the tank volume requirement. The 24 hour Frequency is based on operating experience that has shown there are relatively slow variations in the solution volume.

SR 3.1.7.2

SR 3.1.7.2 verifies the continuity of the explosive charges in the injection valves to ensure that proper operation will occur if required. An automatic continuity monitor may be used to continuously satisfy this requirement. Other administrative controls, such as those that limit the shelf life of the explosive charges, must be followed. The 31 day Frequency is based on operating experience and has demonstrated the reliability of the explosive charge continuity.

SR 3.1.7.3

SR 3.1.7.3 requires an examination of sodium pentaborate solution by using chemical analysis to ensure that the proper concentration of boron exists in the storage tank for post-LOCA suppression pool pH control. This parameter is used as input to determine the volume requirements for SR 3.1.7.1. The concentration is dependent upon the volume of water and quantity of boron in the storage tank.

SR 3.1.7.3 must be performed every 31 days or within 24 hours of when boron or water is added to the storage tank solution to determine that the boron solution concentration is within the specified limits. The 31 day Frequency of this Surveillance is appropriate because of the relatively slow variation of boron concentration between surveillances.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.1.7.4 and SR 3.1.7.6

SR 3.1.7.4 requires an examination of the sodium pentaborate solution by using chemical analysis to ensure that the proper concentration of boron exists in the storage tank. The concentration is dependent upon the volume of water and quantity of boron in the storage tank. SR 3.1.7.6 requires verification that the SLC system conditions satisfy the following equation:

$$\frac{(\ C\ )(\ Q\ )(\ E\ )}{(8.7 \text{ WT \%})(50 \text{ GPM})(94 \text{ ATOM \%})} >= 1.0$$

C = sodium pentaborate solution weight percent concentration
Q = SLC system pump flow rate in gpm
E = Boron-10 atom percent enrichment in the sodium pentaborate solution

To meet 10 CFR 50.62, the SLC System must have a minimum flow capacity and boron content equivalent in control capacity to 86 gpm of 13 weight percent natural sodium pentaborate solution. The atom percentage of natural B-10 is 19.8%. This equivalency requirement is met when the equation given above is satisfied. The equation can be satisfied by adjusting the solution concentration, pump flow rate or Boron-10 enrichment. If the results of the equation are < 1, the SLC System is no longer capable of shutting down the reactor with the margin described in Reference 2. As described in Reference 2, the BFN analysis assumes a flow capacity and boron content equivalent to 50 gpm of 8.7 weight percent and 94 atom percent B-10 enriched sodium pentaborate solution. This exceeds the requirement of 10 CFR 50.62, and the equation is adjusted to reflect the BFN requirements. The quantity of stored boron includes an additional margin (25%) beyond the amount needed to shut down the reactor to allow for possible imperfect mixing of the chemical solution in the reactor water, leakage, and the volume in other piping connected to the reactor system.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.1.7.4 and SR 3.1.7.6

The sodium pentaborate solution (SPB) concentration is allowed to be > 9.2 weight percent provided the concentration and temperature of the sodium pentaborate solution are verified

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.1.7.11

SR 3.1.7.11 verifies that each valve in the system is in its correct position, but does not apply to the squib (i.e., explosive) valves. Verifying the correct alignment for manual, power operated, and automatic valves in the SLC System Flowpath provides assurance that the proper flow paths will exist for system operation. A valve is also allowed to be in the nonaccident position provided it can be aligned to the accident position from the control room, or locally by a dedicated operator at the valve control. This is acceptable since the SLC System is a manually initiated system. This surveillance also does not apply to valves that are locked, sealed, or otherwise secured in position since they are verified to be in the correct position prior to locking, sealing or securing. This verification of valve alignment does not require any testing or valve manipulation; rather, it involves verification that those valves capable of being mispositioned are in the correct position. This SR does not apply to valves that cannot be inadvertently misaligned, such as check valves. The 31 day Frequency is based on engineering judgment and is consistent with the procedural controls governing valve operation that ensures correct valve positions.

REFERENCES

1. 10 CFR 50.62.
2. NEDC-33860P, “Safety Analysis Report for Browns Ferry Nuclear Plant Units 1, 2, and 3 Extended Power Uprate,” Section 2.8.
3. NRC No. 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.
4. FSAR, Section 14.6.

BASES (continued)

APPLICABILITY

APLHGR limits are primarily derived from fuel design evaluations and LOCA and transient analyses assumed to occur at high power levels. Design calculations and operating experience have shown that as power is reduced, the margin to the required APLHGR limits increases. This trend continues down to the power range of 5% to 15% RTP when entry into MODE 2 occurs. When in MODE 2, the intermediate range monitor scram function provides prompt scram initiation during any significant transient, thereby effectively removing any APLHGR limit compliance concern in MODE 2. Therefore, at THERMAL POWER levels ≤ 23% RTP, the reactor is operating with substantial margin to the APLHGR limits; thus, this LCO is not required.

(continued)

BASES (continued)

ACTIONS

A.1

If any APLHGR exceeds the required limits, an assumption regarding an initial condition of the DBA and transient analyses may not be met. Therefore, prompt action should be taken to restore the APLHGR(s) to within the required limits such that the plant operates within analyzed conditions and within design limits of the fuel rods. The 2 hour Completion Time is sufficient to restore the APLHGR(s) to within its limits and is acceptable based on the low probability of a transient or DBA occurring simultaneously with the APLHGR out of specification.

B.1

If the APLHGR cannot be restored to within its required limits within the associated Completion Time, the plant must be brought to a MODE or other specified condition in which the LCO does not apply. To achieve this status, THERMAL POWER must be reduced to < 23% RTP within 4 hours. The allowed Completion Time is reasonable, based on operating experience, to reduce THERMAL POWER to < 23% RTP in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.2.1.1

APLHGRs are required to be initially calculated within 12 hours after THERMAL POWER is ≥ 23% RTP and then every 24 hours thereafter. They are compared to the specified limits in the COLR to ensure that the reactor is operating within the assumptions of the safety analysis. The 24 hour Frequency is based on both engineering judgment and recognition of the slowness of changes in power distribution during normal operation. The 12 hour allowance after THERMAL POWER ≥ 23% RTP is achieved is acceptable given the large inherent margin to operating limits at low power levels.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

evaluated are loss of flow, increase in pressure and power, positive reactivity insertion, and coolant temperature decrease. The limiting transient yields the largest change in CPR (ΔCPR). When the largest ΔCPR is added to the MCPR SL, the required operating limit MCPR is obtained.

The MCPR operating limits derived from the transient analysis are dependent on the operating core flow and power to ensure adherence to fuel design limits during the worst transient that occurs with moderate frequency. Flow dependent MCPR ($MCPR_f$) limits are determined by steady-state thermal hydraulic methods using the three-dimensional BWR simulator code (Reference 12) and the multichannel thermal hydraulic code (Reference 13). The operating limit is dependent on the maximum core flow limiter setting in the Recirculation Flow Control System.

Power-dependent MCPR limits ($MCPR_p$) are determined by the three-dimensional BWR simulator code (Ref. 12) and the one-dimensional transient codes (Refs. 14 and 15). Due to the sensitivity of the transient response to initial core flow levels at power levels below those at which the turbine stop valve closure and turbine control valve fast closure scrams are bypassed, high and low flow $MCPR_p$ operating limits are provided for operating between 23% RTP and the previously mentioned bypass power level.

The MCPR satisfies Criterion 2 of the NRC Policy Statement (Ref. 7).

(continued)

BASES (continued)

LCO

The MCPR operating limits specified in the COLR are the result of the Design Basis Accident (DBA) and transient analysis. Additional MCPR operating limits may be provided in the COLR to support analyzed equipment out-of-service operation. The operating limit MCPR is determined by the larger of the $MCPR_f$ and $MCPR_p$ limits.

APPLICABILITY

The MCPR operating limits are primarily derived from transient analyses that are assumed to occur at high power levels. Below 23% RTP, the reactor is operating at a minimum recirculation pump speed and the moderator void ratio is small. Surveillance of thermal limits below 23% RTP is unnecessary due to the large inherent margin that ensures that the MCPR SL is not exceeded even if a limiting transient occurs. Statistical analyses indicate that the nominal value of the initial MCPR expected at 23% RTP is > 3.5. Studies of the variation of limiting transient behavior have been performed over the range of power and flow conditions. These studies encompass the range of key actual plant parameter values important to typically limiting transients. The results of these studies demonstrate that a margin is expected between performance and the MCPR requirements, and that margins increase as power is reduced to 23% RTP. This trend is expected to continue to the 5% to 15% power range when entry into MODE 2 occurs. When in MODE 2, the intermediate range monitor provides rapid scram initiation for any significant power increase transient, which effectively eliminates any MCPR compliance concern. Therefore, at THERMAL POWER levels < 23% RTP, the reactor is operating with substantial margin to the MCPR limits and this LCO is not required.

(continued)

BASES (continued)

ACTIONS

A.1

If any MCPR is outside the required limits, an assumption regarding an initial condition of the design basis transient analyses may not be met. Therefore, prompt action should be taken to restore the MCPR(s) to within the required limits such that the plant remains operating within analyzed conditions. The 2 hour Completion Time is normally sufficient to restore the MCPR(s) to within its limits and is acceptable based on the low probability of a transient or DBA occurring simultaneously with the MCPR out of specification.

B.1

If the MCPR cannot be restored to within its required limits within the associated Completion Time, the plant must be brought to a MODE or other specified condition in which the LCO does not apply. To achieve this status, THERMAL POWER must be reduced to < 23% RTP within 4 hours. The allowed Completion Time is reasonable, based on operating experience, to reduce THERMAL POWER to < 23% RTP in an orderly manner and without challenging plant systems.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.2.2.1

The MCPR is required to be initially calculated within 12 hours after THERMAL POWER is ≥ 23% RTP and then every 24 hours thereafter. It is compared to the specified limits in the COLR to ensure that the reactor is operating within the assumptions of the safety analysis. The 24 hour Frequency is based on both engineering judgment and recognition of the slowness of changes in power distribution during normal operation. The 12 hour allowance after THERMAL POWER ≥ 23% RTP is achieved is acceptable given the large inherent margin to operating limits at low power levels.

SR 3.2.2.2

Because the transient analysis takes credit for conservatism in the scram speed performance, it must be demonstrated that the specific scram speed distribution is consistent with that used in the transient analysis. SR 3.2.2.2 determines the actual scram speed distribution and compares it with the assumed distribution. The MCPR operating limit is determined based either on the applicable limit associated with scram times of LCO 3.1.4, “Control Rod Scram Times,” or the nominal scram times. The scram speed-dependent MCPR limits are contained in the COLR. This determination must be performed within 72 hours after each set of control rod scram time tests required by SR 3.1.4.1 and SR 3.1.4.2 because the effective scram speed distribution may change during the cycle. The 72-hour Completion Time is acceptable due to the relatively minor changes in the actual control rod scram speed distribution expected during the fuel cycle.

(continued)

BASES (continued)

LCO

The LHGR is a basic assumption in the fuel design analysis. The fuel has been designed to operate at rated core power with sufficient design margin to the LHGR calculated to cause a 1% fuel cladding plastic strain. The operating limit to accomplish this objective is specified in the COLR. Additional LHGR operating limits adjustments may be provided in the COLR to support analyzed equipment out-of-service operation.

Additional LHGR operating limits adjustments may be provided in the COLR to support analyzed equipment out-of-service operation.

APPLICABILITY

The LHGR limits are derived from fuel design analysis that is limiting at high power level conditions. At core thermal power levels < 23% RTP, the reactor is operating with a substantial margin to the LHGR limits and, therefore, the Specification is only required when the reactor is operating at ≥ 23% RTP.

(continued)

BASES (continued)

ACTIONS

A.1

If any LHGR exceeds its required limit, an assumption regarding an initial condition of the fuel design analysis is not met. Therefore, prompt action should be taken to restore the LHGR(s) to within its required limits such that the plant is operating within analyzed conditions. The 2 hour Completion Time is normally sufficient to restore the LHGR(s) to within its limits and is acceptable based on the low probability of a transient or Design Basis Accident occurring simultaneously with the LHGR out of specification.

B.1

If the LHGR cannot be restored to within its required limits within the associated Completion Time, the plant must be brought to a MODE or other specified condition in which the LCO does not apply. To achieve this status, THERMAL POWER is reduced to < 23% RTP within 4 hours. The allowed Completion Time is reasonable, based on operating experience, to reduce THERMAL POWER to < 23% RTP in an orderly manner and without challenging plant systems.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.2.3.1

The LHGR is required to be initially calculated within 12 hours after THERMAL POWER is ≥ 23% RTP and then every 24 hours thereafter. It is compared to the specified limits in the COLR to ensure that the reactor is operating within the assumptions of the safety analysis. The 24 hour Frequency is based on both engineering judgment and recognition of the slow changes in power distribution during normal operation. The 12 hour allowance after THERMAL POWER ≥ 23% RTP is achieved is acceptable given the large inherent margin to operating limits at lower power levels.

REFERENCES

1. FSAR, Chapter 14.
2. FSAR, Chapter 3.
3. NUREG-0800, Standard Review Plan 4.2, Section II.A.2(g), Revision 2, July 1981.
4. NRC No. 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

The trip setpoints are then determined accounting for the remaining instrument errors (e.g., drift). The trip setpoints derived in this manner provide adequate protection because instrumentation uncertainties, process effects, calibration tolerances, instrument drift, and severe environmental effects (for channels that must function in harsh environments as defined by 10 CFR 50.49) are accounted for.

The OPERABILITY of scram pilot valves and associated solenoids, backup scram valves, and SDV valves, described in the Background section, are not addressed by this LCO.

The individual Functions are required to be OPERABLE in the MODES or other specified conditions in the Table, which may require an RPS trip to mitigate the consequences of a design basis accident or transient. To ensure a reliable scram function, a combination of Functions are required in each MODE to provide primary and diverse initiation signals.

The only MODES specified in Table 3.3.1.1-1 are MODES 1 (which encompasses $\geq$ 26% RTP) and 2, and MODE 5 with any control rod withdrawn from a core cell containing one or more fuel assemblies. No RPS Function is required in MODES 3 and 4 since all control rods are fully inserted and the Reactor Mode Switch Shutdown Position control rod withdrawal block (LCO 3.3.2.1) does not allow any control rod to be withdrawn. In MODE 5, control rods withdrawn from a core cell containing no fuel assemblies do not affect the reactivity of the core and, therefore, are not required to have the capability to scram. Provided all other control rods remain inserted, no RPS function is required. In this condition, the required SDM (LCO 3.1.1) and refuel position one-rod-out interlock (LCO 3.9.2) ensure that no event requiring RPS will occur.

The specific Applicable Safety Analyses, LCO, and Applicability discussions are listed below on a Function by Function basis.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

2.a. Average Power Range Monitor Neutron Flux - High, (Setdown)

For operation at low power (i.e., MODE 2), the Average Power Range Monitor Neutron Flux - High, (Setdown) Function is capable of generating a trip signal that prevents fuel damage resulting from abnormal operating transients in this power range. For most operation at low power levels, the Average Power Range Monitor Neutron Flux - High, (Setdown) Function will provide a secondary scram to the Intermediate Range Monitor Neutron Flux - High Function because of the relative setpoints. With the IRMs at Range 9 or 10, it is possible that the Average Power Range Monitor Neutron Flux - High, (Setdown) Function will provide the primary trip signal for a corewide increase in power.

No specific safety analyses take direct credit for the Average Power Range Monitor Neutron Flux - High, (Setdown) Function. However, this Function indirectly ensures that before the reactor mode switch is placed in the run position, reactor power does not exceed 23% RTP (SL 2.1.1.1) when operating at low reactor pressure and low core flow. Therefore, it indirectly prevents fuel damage during significant reactivity increases with THERMAL POWER < 23% RTP.

The Allowable Value is based on preventing significant increases in power when THERMAL POWER is < 23% RTP.

The Average Power Range Monitor Neutron Flux - High, (Setdown) Function must be OPERABLE during MODE 2 when control rods may be withdrawn since the potential for criticality exists. In MODE 1, the Average Power Range Monitor Neutron Flux - High Function provides protection against reactivity transients and the RWM and rod block monitor protect against control rod withdrawal error events.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

2.f. Oscillation Power Range Monitor (OPRM) Upscale

The OPRM Upscale Function provides compliance with GDC 10 and GDC 12, thereby providing protection from exceeding the fuel MCPR safety limit (SL) due to anticipated thermal hydraulic power oscillations.

References 13, 14, and 15 describe three algorithms for detecting thermal hydraulic instability related neutron flux oscillations: the period based detection algorithm, the amplitude based algorithm, and the growth rate algorithm. All three are implemented in the OPRM Upscale Function, but the safety analysis takes credit only for the period based detection algorithm. The remaining algorithms provide defense in depth and additional protection against unanticipated oscillations. OPRM Upscale Function OPERABILITY for Technical Specification purposes is based only on the period based detection algorithm.

The OPRM Upscale Function receives input signals from the local power range monitors (LPRMs) within the reactor core, which are combined into "cells" for evaluation of the OPRM algorithms.

The OPRM Upscale Function is required to be OPERABLE when the plant is in a region of power flow operation where anticipated events could lead to thermal hydraulic instability and related neutron flux oscillations. Within this region, the automatic trip is enabled when THERMAL POWER, as indicated by the APRM Simulated Thermal Power, is ≥ 23% RTP and reactor core flow, as indicated by recirculation drive flow is < 60% of rated flow, the operating region where actual thermal hydraulic oscillations may occur. Requiring the OPRM Upscale Function to be OPERABLE in MODE 1 provides consistency with operability requirements for other APRM functions and assures that the OPRM Upscale Function is OPERABLE whenever reactor power could increase into the region of concern without operator action.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

8. Turbine Stop Valve - Closure (continued)

Turbine Stop Valve - Closure signals are initiated from position switches located on each of the four TSVs. Two independent position switches are associated with each stop valve. One of the two switches provides input to RPS trip system A; the other, to RPS trip system B. Thus, each RPS trip system receives an input from four Turbine Stop Valve - Closure channels, each consisting of one position switch. The logic for the Turbine Stop Valve - Closure Function is such that three or more TSVs must be closed to produce a scram. This Function must be enabled at THERMAL POWER ≥ 26% RTP. This is normally accomplished automatically by pressure transmitters sensing turbine first stage pressure; therefore, opening the turbine bypass valves may affect this function.

The Turbine Stop Valve - Closure Allowable Value is selected to be high enough to detect imminent TSV closure, thereby reducing the severity of the subsequent pressure transient.

Eight channels of Turbine Stop Valve - Closure Function, with four channels in each trip system, are required to be OPERABLE to ensure that no single instrument failure will preclude a scram from this Function if any three TSVs should close. This Function is required, consistent with analysis assumptions, whenever THERMAL POWER is ≥ 26% RTP. This Function is not required when THERMAL POWER is < 26% RTP since the Reactor Vessel Steam Dome Pressure - High and the Average Power Range Monitor Fixed Neutron Flux - High Functions are adequate to maintain the necessary safety margins.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

9. Turbine Control Valve Fast Closure, Trip Oil Pressure - Low (PS-47-142, PS-47-144, PS-47-146, and PS-47-148)

Fast closure of the TCVs results in the loss of a heat sink that produces reactor pressure, neutron flux, and heat flux transients that must be limited. Therefore, a reactor scram is initiated on TCV fast closure in anticipation of the transients that would result from the closure of these valves. The Turbine Control Valve Fast Closure, Trip Oil Pressure - Low Function is the primary scram signal for the generator load rejection event analyzed in Reference 7. For this event, the reactor scram reduces the amount of energy required to be absorbed and, along with the actions of the EOC-RPT System, ensures that the MCPR SL is not exceeded.

Turbine Control Valve Fast Closure, Trip Oil Pressure - Low signals are initiated by the electrohydraulic control (EHC) fluid pressure at each control valve. One pressure switch is associated with each control valve, and the signal from each switch is assigned to a separate RPS logic channel. This Function must be enabled at THERMAL POWER ≥ 26% RTP. This is normally accomplished automatically by pressure transmitters sensing turbine first stage pressure; therefore, opening the turbine bypass valves may affect this function.

The Turbine Control Valve Fast Closure, Trip Oil Pressure - Low Allowable Value is selected high enough to detect imminent TCV fast closure.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

9. Turbine Control Valve Fast Closure, Trip Oil Pressure - Low (PS-47-142, PS-47-144, PS-47-146, and PS-47-148) (continued)

Four channels of Turbine Control Valve Fast Closure, Trip Oil Pressure - Low Function with two channels in each trip system arranged in a one-out-of-two logic are required to be OPERABLE to ensure that no single instrument failure will preclude a scram from this Function on a valid signal. This Function is required, consistent with the analysis assumptions, whenever THERMAL POWER is ≥ 26% RTP. This Function is not required when THERMAL POWER is < 26% RTP, since the Reactor Vessel Steam Dome Pressure - High and the Average Power Range Monitor Fixed Neutron Flux - High Functions are adequate to maintain the necessary safety margins.

For this instrument function, the nominal trip setpoint including the as-left tolerances is defined as the LSSS. The acceptable as-found band is based on a statistical combination of possible measurable uncertainties (i.e., setting tolerance, drift, temperature effects, and measurement and test equipment). During instrument calibrations, if the as-found setpoint is found to be conservative with respect to the Allowable Value, but outside its acceptable as-found band (tolerance range), as defined by its associated Surveillance Requirement procedure, then there shall be an initial determination to ensure confidence that the channel can perform as required before returning the channel to service in accordance with the Surveillance. The technician performing the Surveillance will evaluate the instrument's ability to maintain a stable setpoint within the as-left tolerance. The technician's evaluation will be reviewed by on shift personnel during the approval of the Surveillance data prior to returning the channel back to service at the completion of the Surveillance. This shall constitute the initial determination of operability. If a channel is found to exceed the channel's Allowable Value or cannot be reset within the

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.1.1.2

To ensure that the APRMs are accurately indicating the true core average power, the APRMs are calibrated to the reactor power calculated from a heat balance. The Frequency of once per 7 days is based on minor changes in LPRM sensitivity, which could affect the APRM reading, between performances of SR 3.3.1.1.7.

A restriction to satisfying this SR when < 23% RTP is provided that requires the SR to be met only at ≥ 23% RTP because it is difficult to accurately maintain APRM indication of core THERMAL POWER consistent with a heat balance when < 23% RTP. At low power levels, a high degree of accuracy is unnecessary because of the large, inherent margin to thermal limits (MCPR and APLHGR). At ≥ 23% RTP, the Surveillance is required to have been satisfactorily performed within the last 7 days, in accordance with SR 3.0.2. A Note is provided which allows an increase in THERMAL POWER above 23% if the 7 day Frequency is not met per SR 3.0.2. In this event, the SR must be performed within 12 hours after reaching or exceeding 23% RTP. Twelve hours is based on operating experience and in consideration of providing a reasonable time in which to complete the SR.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.1.1.15

This SR ensures that scrams initiated from the Turbine Stop Valve - Closure and Turbine Control Valve Fast Closure, Trip Oil Pressure - Low Functions will not be inadvertently bypassed when THERMAL POWER is ≥ 26% RTP. This involves calibration of the bypass channels (PIS-1-81A, PIS-1-81B, PIS-1-91A, and PIS-1-91B). Adequate margins for the instrument setpoint methodologies are incorporated into the actual setpoint.

If any bypass channel's setpoint is nonconservative (i.e., the Functions are bypassed at ≥ 26% RTP, either due to open main turbine bypass valve(s) or other reasons), then the affected Turbine Stop Valve - Closure and Turbine Control Valve Fast Closure, Trip Oil Pressure - Low Functions are considered inoperable. Alternatively, the bypass channel can be placed in the conservative condition (nonbypass). If placed in the nonbypass condition (Turbine Stop Valve - Closure and Turbine Control Valve Fast Closure, Trip Oil Pressure - Low Functions are enabled), this SR is met and the channel is considered OPERABLE.

The Frequency of 24 months is based upon the assumption of a 24 month calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.1.1.17

This SR ensures that scrams initiated from OPRM Upscale Function (Function 2.f) will not be inadvertently bypassed when THERMAL POWER, as indicated by the APRM Simulated Thermal Power, is ≥ 23% RTP and core flow, as indicated by recirculation drive flow, is < 60% rated core flow. This normally involves confirming the bypass setpoints. Adequate margins for the instrument setpoint methodologies are incorporated into the actual setpoint. The actual surveillance ensures that the OPRM Upscale Function is enabled (not bypassed) for the correct values of APRM Simulated Thermal Power and recirculation drive flow. Other surveillances ensure that the APRM Simulated Thermal Power and recirculation flow properly correlate with THERMAL POWER and core flow, respectively.

If any bypass setpoint is nonconservative (i.e., the OPRM Upscale Function is bypassed when APRM Simulated Thermal Power ≥ 23% RTP and recirculation drive flow < 60% rated), then the affected channel is considered inoperable for the OPRM Upscale Function. Alternatively, the bypass setpoint may be adjusted to place the channel in a conservative condition (unbypass). If placed in the unbypassed condition, this SR is met and the channel is considered OPERABLE.

The frequency of 24 months is based on engineering judgment and reliability of the components.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

The feedwater and main turbine high water level trip instrumentation is assumed to be capable of providing a turbine trip in the design basis transient analysis for a feedwater controller failure, maximum demand event (Ref. 1). The reactor vessel high water level trip indirectly initiates a reactor scram from the main turbine trip (above 26% RTP) and trips the feedwater pumps, thereby terminating the event. The reactor scram mitigates the reduction in MCPR.

Feedwater and main turbine high water level trip instrumentation satisfies Criterion 3 of the NRC Policy Statement (Ref. 3).

LCO

The LCO requires two channels of the Reactor Vessel Water Level - High instrumentation per trip system to be OPERABLE to ensure that no single instrument failure will prevent the feedwater pump turbines and main turbine trip on a valid Reactor Vessel Water Level - High signal. Both channels in either trip system are needed to provide trip signals in order for the feedwater and main turbine trips to occur. Each channel must have its setpoint set within the specified Allowable Value of SR 3.3.2.2.3. The Allowable Value is set to ensure that the thermal limits are not exceeded during the event. The actual setpoint is calibrated to be consistent with the applicable setpoint methodology assumptions. Nominal trip setpoints are specified in the setpoint calculations. The nominal setpoints are selected to ensure that the setpoints do not exceed the Allowable Value between successive CHANNEL CALIBRATIONS. Operation with a trip setpoint less conservative than the nominal trip setpoint, but within its Allowable Value, is acceptable.

(continued)

BASES

LCO (continued)

Trip setpoints are those predetermined values of output at which an action should take place. The setpoints are compared to the actual process parameter (e.g., reactor vessel water level), and when the measured output value of the process parameter exceeds the setpoint, the associated device (e.g., trip unit) changes state. The analytic limits are derived from the limiting values of the process parameters obtained from the safety analysis. The Allowable Values are derived from the analytic limits, corrected for calibration, process, and some of the instrument errors. A channel is inoperable if its actual trip setpoint is not within its required Allowable Value. The trip setpoints are then determined accounting for the remaining instrument errors (e.g., drift). The trip setpoints derived in this manner provide adequate protection because instrumentation uncertainties, process effects, calibration tolerances, instrument drift, and severe environmental effects (for channels that must function in harsh environments as defined by 10 CFR 50.49) are accounted for.

APPLICABILITY

The feedwater and main turbine high water level trip instrumentation is required to be OPERABLE at ≥ 23% RTP to ensure that the fuel cladding integrity Safety Limit and the cladding 1% plastic strain limit are not violated during the feedwater controller failure, maximum demand event. As discussed in the Bases for LCO 3.2.1, "Average Planar Linear Heat Generation Rate (APLHGR)," LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)," and LCO 3.2.3, “LINEAR HEAT GENERATION RATE (LHGR),” sufficient margin to these limits exists below 23% RTP; therefore, these requirements are only necessary when operating at or above this power level.

(continued)

BASES

ACTIONS

B.1 (continued)

The 2 hour Completion Time is sufficient for the operator to take corrective action, and takes into account the likelihood of an event requiring actuation of feedwater and main turbine high water level trip instrumentation occurring during this period. It is also consistent with the 2 hour Completion Time provided in LCO 3.2.2 for Required Action A.1, since this instrumentation's purpose is to preclude a MCPR violation.

C.1

With the required channels not restored to OPERABLE status or placed in trip, THERMAL POWER must be reduced to < 23% RTP within 4 hours. As discussed in the Applicability section of the Bases, operation below 23% RTP results in sufficient margin to the required limits, and the feedwater and main turbine high water level trip instrumentation is not required to protect fuel integrity during the feedwater controller failure, maximum demand event. The allowed Completion Time of 4 hours is based on operating experience to reduce THERMAL POWER to < 23% RTP from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

The Surveillances are modified by a Note to indicate that when a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided the associated Function maintains feedwater and main turbine high water level trip capability. Upon completion of the Surveillance, or expiration of the 6 hour allowance, the channel must be returned to OPERABLE status

(continued)

BASES

BACKGROUND (continued)

Each EOC-RPT trip system is a two-out-of-two logic for each Function; thus, either two TSV - Closure or two TCV Fast Closure, Trip Oil Pressure - Low signals are required for a trip system to actuate. If either trip system actuates, both recirculation pumps will trip. There are two EOC-RPT breakers in series per recirculation pump. One trip system trips one of the two EOC-RPT breakers for each recirculation pump, and the second trip system trips the other EOC-RPT breaker for each recirculation pump.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

The TSV - Closure and the TCV Fast Closure, Trip Oil Pressure - Low Functions are designed to trip the recirculation pumps in the event of a turbine trip or generator load rejection to mitigate the increase in neutron flux, heat flux, and reactor pressure, and to increase the margin to the MCPR SL and LHGR limits. The analytical methods and assumptions used in evaluating the turbine trip and generator load rejection are summarized in References 2, 3, and 4.

To mitigate pressurization transient effects, the EOC-RPT must trip the recirculation pumps after initiation of closure movement of either the TSVs or the TCVs. The combined effects of this trip and a scram reduce fuel bundle power more rapidly than a scram alone, resulting in an increased margin to the MCPR SL and LHGR limits. Alternatively, MCPR limits for an inoperable EOC-RPT, as specified in the COLR, are sufficient to prevent violation of the MCPR Safety Limit and fuel mechanical limits. The EOC-RPT function is automatically disabled when turbine first stage pressure is < 26% RTP.

EOC-RPT instrumentation satisfies Criterion 3 of the NRC Policy Statement (Ref. 6).

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

Turbine Stop Valve - Closure (continued)

Closure of the TSVs is determined by measuring the position of each valve. There are two separate position signals associated with each stop valve, the signal from each switch being assigned to a separate trip channel. The logic for the TSV - Closure Function is such that two or more TSVs must be closed to produce an EOC-RPT. This Function must be enabled at THERMAL POWER ≥ 26% RTP. This is normally accomplished automatically by pressure transmitters sensing turbine first stage pressure; therefore, opening the turbine bypass valves may affect this function. To consider this function OPERABLE, bypass of the function must not occur when bypass valves are open. Four channels of TSV - Closure, with two channels in each trip system, are available and required to be OPERABLE to ensure that no single instrument failure will preclude an EOC-RPT from this Function on a valid signal. The TSV - Closure Allowable Value is selected to detect imminent TSV closure.

This protection is required, consistent with the safety analysis assumptions, whenever THERMAL POWER is ≥ 26% RTP. Below 26% RTP, the Reactor Vessel Steam Dome Pressure - High and the Average Power Range Monitor (APRM) Fixed Neutron Flux - High Functions of the Reactor Protection System (RPS) are adequate to maintain the necessary margin to the MCPR SL and LHGR limits.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

Turbine Control Valve Fast Closure, Trip Oil Pressure - Low (PS-47-142, PS-47-144, PS-47-146, and PS-47-148)

Fast closure of the TCVs during a generator load rejection results in the loss of a heat sink that produces reactor pressure, neutron flux, and heat flux transients that must be limited. Therefore, an RPT is initiated on TCV Fast Closure, Trip Oil Pressure - Low in anticipation of the transients that would result from the closure of these valves. The EOC-RPT decreases reactor power and aids the reactor scram in ensuring that the MCPR SL and LHGR limits are not exceeded during the worst case transient.

Fast closure of the TCVs is determined by measuring the electrohydraulic control fluid pressure at each control valve. There is one pressure switch associated with each control valve, and the signal from each switch is assigned to a separate trip channel. The logic for the TCV Fast Closure, Trip Oil Pressure - Low Function is such that two or more TCVs must be closed (pressure switch trips) to produce an EOC-RPT. This Function must be enabled at THERMAL POWER ≥ 26% RTP. This is normally accomplished automatically by pressure transmitters sensing turbine first stage pressure; therefore, opening the turbine bypass valves may affect this function. To consider this function OPERABLE, bypass of the function must not occur when bypass valves are open. Four channels of TCV Fast Closure, Trip Oil Pressure - Low, with two channels in each trip system, are available and required to be OPERABLE to ensure that no single instrument failure will preclude an EOC-RPT from this Function on a valid signal. The TCV Fast Closure, Trip Oil Pressure - Low Allowable Value is selected high enough to detect imminent TCV fast closure.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY	Turbine Control Valve Fast Closure, Trip Oil Pressure - Low (PS-47-142, PS-47-144, PS-47-146, and PS-47-148) (continued) This protection is required consistent with the safety analysis whenever THERMAL POWER is ≥ 26% RTP. Below 26% RTP, the Reactor Vessel Steam Dome Pressure - High and the APRM Fixed Neutron Flux - High Functions of the RPS are adequate to maintain the necessary safety margins.
ACTIONS	A Note has been provided to modify the ACTIONS related to EOC-RPT instrumentation channels. Section 1.3, Completion Times, specifies that once a Condition has been entered, subsequent divisions, subsystems, components, or variables expressed in the Condition, discovered to be inoperable or not within limits, will not result in separate entry into the Condition. Section 1.3 also specifies that Required Actions of the Condition continue to apply for each additional failure, with Completion Times based on initial entry into the Condition. However, the Required Actions for inoperable EOC-RPT instrumentation channels provide appropriate compensatory measures for separate inoperable channels. As such, a Note has been provided that allows separate Condition entry for each inoperable EOC-RPT instrumentation channel.

(continued)

BASES

ACTIONS (continued)

B.1 and B.2

Required Actions B.1 and B.2 are intended to ensure that appropriate actions are taken if multiple, inoperable, untripped channels within the same Function result in the Function not maintaining EOC-RPT trip capability. A Function is considered to be maintaining EOC-RPT trip capability when sufficient channels are OPERABLE or in trip, such that the EOC-RPT System will generate a trip signal from the given Function on a valid signal and both recirculation pumps can be tripped. Alternately, Required Action B.2 requires the MCPR and LHGR limits for inoperable EOC-RPT, as specified in the COLR, to be applied. This also restores the margin to MCPR and LHGR limits assumed in the safety analysis.

The 2 hour Completion Time is sufficient time for the operator to take corrective action, and takes into account the likelihood of an event requiring actuation of the EOC-RPT instrumentation during this period. It is also consistent with the 2 hour Completion Time provided in LCO 3.2.2 for Required Action A.1, since this instrumentation's purpose is to preclude a MCPR or LHGR violation.

C.1

With any Required Action and associated Completion Time not met, THERMAL POWER must be reduced to < 26% RTP within 4 hours. The allowed Completion Time of 4 hours is reasonable, based on operating experience, to reduce THERMAL POWER to < 26% RTP from full power conditions in an orderly manner and without challenging plant systems.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.4.1.2

This SR ensures that an EOC-RPT initiated from the TSV - Closure and TCV Fast Closure, Trip Oil Pressure - Low Functions will not be inadvertently bypassed when THERMAL POWER is ≥ 26% RTP. This involves calibration of the bypass channels. Adequate margins for the instrument setpoint methodologies are incorporated into the actual setpoint. If any bypass channel's setpoint is nonconservative (i.e., the Functions are bypassed at ≥ 26% RTP, either due to open main turbine bypass valves or other reasons), the affected TSV - Closure and TCV Fast Closure, Trip Oil Pressure - Low Functions are considered inoperable. Alternatively, the bypass channel can be placed in the conservative condition (nonbypass). If placed in the nonbypass condition, this SR is met with the channel considered OPERABLE.

The Frequency of 24 months is based upon the assumption of a 24 month calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis.

SR 3.3.4.1.3

CHANNEL CALIBRATION is a complete check of the instrument loop and the sensor. This test verifies the channel responds to the measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drifts between successive calibrations consistent with the plant specific setpoint methodology. The Frequency is based upon the assumption of an 24 month calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.2.1 (continued)

Note 2 allows this SR not to be performed until 24 hours after THERMAL POWER exceeds 23% of RTP. During low flow conditions, jet pump noise approaches the threshold response of the associated flow instrumentation and precludes the collection of repeatable and meaningful data. The 24 hours is an acceptable time to establish conditions appropriate to perform this SR.

REFERENCES

1. FSAR, Section 14.6.3.
2. GE Service Information Letter No. 330, "Jet Pump Beam Cracks," June 9, 1980.
3. NUREG/CR-3052, "Closeout of IE Bulletin 80-07: BWR Jet Pump Assembly Failure," November 1984.
4. NRC No. 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.

BASES

APPLICABLE SAFETY ANALYSES (continued)

The maximum allowable leakage rate for the primary containment (L_a) is 2.0% by weight of the containment air per 24 hours at the design basis LOCA maximum peak containment pressure (P_a) of 49.1 psig (Ref. 1).

Primary containment satisfies Criterion 3 of the NRC Policy Statement (Ref. 6).

LCO

Primary containment OPERABILITY is maintained by limiting leakage to $\leq$ 1.0 L_a, except prior to the first startup after performing a required Primary Containment Leakage Rate Testing Program leakage test. At this time, applicable leakage limits must be met. Compliance with this LCO will ensure a primary containment configuration, including equipment hatches, that is structurally sound and that will limit leakage to those leakage rates assumed in the safety analyses.

Individual leakage rates specified for the primary containment air lock are addressed in LCO 3.6.1.2.

APPLICABILITY

In MODES 1, 2, and 3, a DBA could cause a release of radioactive material to primary containment. In MODES 4 and 5, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, primary containment is not required to be OPERABLE in MODES 4 and 5 to prevent leakage of radioactive material from primary containment.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.1.1.2 (continued)

Satisfactory performance of this SR can be achieved by establishing a known differential pressure between the drywell and the suppression chamber and verifying that the pressure in either the suppression chamber or the drywell does not change by more than 0.25 inch of water per minute over a 10 minute period. The leakage test is performed every 24 months. The 24 month Frequency was developed considering it is prudent that this Surveillance be performed during a unit outage and also in view of the fact that component failures that might have affected this test are identified by other primary containment SRs.

REFERENCES

1. NEDC-33860P, “Safety Analysis Report for Browns Ferry Nuclear Plant Units 1, 2, and 3 Extended Power Uprate,” Section 2.6.
2. FSAR, Section 14.6.
3. 10 CFR 50, Appendix J, Option B.
4. NEI 94-01, Revision O, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J."
5. ANSI/ANS-56.8-1994, "American National Standard for Containment System Leakage Testing Requirement."
6. NRC No. 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.

BASES

BACKGROUND (continued)	the air lock to remain open for extended periods when frequent primary containment entry is necessary. Under some conditions allowed by this LCO, the primary containment may be accessed through the air lock, when the interlock mechanism has failed, by manually performing the interlock function. The primary containment air lock forms part of the primary containment pressure boundary. As such, air lock integrity and leak tightness are essential for maintaining primary containment leakage rate to within limits in the event of a DBA. Not maintaining air lock integrity or leak tightness may result in a leakage rate in excess of that assumed in the unit safety analysis.
APPLICABLE SAFETY ANALYSES	The DBA that postulates the maximum release of radioactive material within primary containment is a LOCA. In the analysis of this accident, it is assumed that primary containment is OPERABLE, such that release of fission products to the environment is controlled by the rate of primary containment leakage. The primary containment is designed with a maximum allowable leakage rate (L_a) of 2.0% by weight of the containment air per 24 hours at the calculated maximum peak containment pressure (P_a) of 49.1 psig (Ref. 3). This allowable leakage rate forms the basis for the acceptance criteria imposed on the SRs associated with the air lock. Primary containment air lock OPERABILITY is also required to minimize the amount of fission product gases that may escape primary containment through the air lock and contaminate and pressurize the secondary containment. The primary containment air lock satisfies Criterion 3 of the NRC Policy Statement (Ref. 4).

(continued)

B 3.6 CONTAINMENT SYSTEMS

B 3.6.3.1 Containment Atmosphere Dilution (CAD) System

BASES

BACKGROUND

The CAD System functions to maintain combustible gas concentrations within the primary containment at or below the flammability limits following a postulated loss of coolant accident (LOCA) by diluting hydrogen and oxygen with nitrogen. To ensure that a combustible gas mixture does not occur, oxygen concentration is kept < 5.0 volume percent (v/o), or hydrogen concentration is kept < 4.0 v/o.

The CAD System is manually initiated and consists of two independent, 100% capacity subsystems, each of which is capable of supplying nitrogen through separate piping systems to the drywell and suppression chamber of each unit. Each subsystem includes a liquid nitrogen supply tank, ambient vaporizer, electric heater, and a manifold with branches to each primary containment (for Units 1, 2, and 3). The nitrogen storage tanks each contain ≥ 2615 gal, which is adequate for 7 days of CAD subsystem operation (Ref. 4).

The CAD System operates in conjunction with emergency operating procedures that are used to reduce primary containment pressure periodically during CAD System operation. This combination results in a feed and bleed approach to maintaining hydrogen and oxygen concentrations below combustible levels.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.6.3.1.1

Verifying that there is ≥ 2615 gal of liquid nitrogen supply in each nitrogen storage tank will ensure at least 7 days of post-LOCA CAD operation. This minimum volume of liquid nitrogen represents the analytical limit assumed in the analysis of the primary containment atmosphere following a postulated LOCA and does not include allowance for potential nitrogen boiloff and tank level instrumentation inaccuracies. This minimum volume of liquid nitrogen allows sufficient time after an accident to replenish the nitrogen supply for long term inerting. This is verified every 31 days to ensure that the system is capable of performing its intended function when required. The 31 day Frequency is based on operating experience, which has shown 31 days to be an acceptable period to verify the liquid nitrogen supply and on the availability of other hydrogen mitigating systems.

SR 3.6.3.1.2

Verifying the correct alignment for manual, power operated, and automatic valves in each of the CAD subsystem flow paths provides assurance that the proper flow paths exist for system operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since these valves were verified to be in the correct position prior to locking, sealing, or securing.

A valve is also allowed to be in the nonaccident position provided it can be aligned to the accident position within the time assumed in the accident analysis. This is acceptable because the CAD System is manually initiated. This SR does not apply to valves that cannot be inadvertently misaligned, such as check valves. This SR does not require any testing or valve manipulation; rather, it involves verification that those valves capable of being mispositioned are in the correct position.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.3.1.2 (continued)

The 31 day Frequency is appropriate because the valves are operated under procedural control, improper valve position would only affect a single subsystem, the probability of an event requiring initiation of the system is low, and the system is a manually initiated system.

REFERENCES

1. AEC Safety Guide 7, Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident, March 10, 1971.

2. FSAR, Section 5.2.6.

3. NRC No. 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.

4. ANP-3403P, "Fuel Uprate Safety Analysis Report for Browns Ferry Nuclear Plant Units 1, 2, and 3," Section 2.6.4, August 2015.

B 3.7 PLANT SYSTEMS

B 3.7.1 Residual Heat Removal Service Water (RHRSW) System

BASES

BACKGROUND

The RHRSW System is designed to provide cooling water for the Residual Heat Removal (RHR) System heat exchangers, required for a safe reactor shutdown following a Design Basis Accident (DBA) or transient. The RHRSW System is operated whenever the RHR heat exchangers are required to operate in the shutdown cooling mode or in the suppression pool cooling or spray mode of the RHR System.

The RHRSW System is common to the three BFN units and consists of four independent and redundant subsystems, each of which feeds one RHR heat exchanger in each unit. Each subsystem is made up of a header, two 4500 gpm pumps, a suction source, valves, piping, and associated instrumentation. Two subsystems, with one pump operating in each subsystem, are capable of providing 100% of the required cooling capacity to maintain safe shutdown conditions for one unit. The RHRSW System is designed with sufficient redundancy so that no single active component failure can prevent it from achieving its design function. The RHRSW System is described in the FSAR, Section 10.9 (Ref. 1).

Cooling water is pumped by the RHRSW pumps from the Wheeler Reservoir through the tube side of the RHR heat exchangers, and discharged back to the Wheeler Reservoir.

(continued)

BASES

BACKGROUND (continued)	The system is initiated manually from each of the three units control rooms. If operating during a loss of coolant accident (LOCA), the system is automatically tripped on degraded bus voltage to allow the diesel generators to automatically power only that equipment necessary to reflood the core. The system can be manually started any time the degraded bus voltage signal clears, and is assumed to be manually started within 10 minutes after the LOCA.
APPLICABLE SAFETY ANALYSES	The RHRSW System removes heat from the suppression pool to limit the suppression pool temperature and primary containment pressure following a LOCA. This ensures that the primary containment can perform its function of limiting the release of radioactive materials to the environment following a LOCA. The ability of the RHRSW System to support long term cooling of the reactor or primary containment is discussed in the FSAR, Chapters 5 and 14 (Refs. 2 and 3, respectively). These analyses explicitly assume that the RHRSW System will provide adequate cooling support to the equipment required for safe shutdown. These analyses include the evaluation of the long term primary containment response after a design basis LOCA. The safety analyses for long term cooling were performed for various combinations of RHR System failures and considers the number of units fueled. With one unit fueled, the worst case single failure that would affect the performance of the RHRSW System is any failure that would disable two subsystems of the RHRSW System.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

With two and three units fueled, a worst case single failure could also include the loss of two RHRSW pumps caused by losing a 4 kV shutdown board since there are certain alignment configurations that allow two RHRSW pumps to be powered from the same 4 kV shutdown board. As discussed in the FSAR, Section 14.6.3.3.2 (Ref. 4) for these analyses, manual initiation of the OPERABLE RHRSW subsystems and the associated RHR System is assumed to occur 10 minutes after a DBA. The analyses assume that there are two RHRSW subsystems operating in each unit, with one RHRSW pump in each subsystem capable of producing 4000 gpm of flow. In this case, the maximum suppression chamber water temperature and pressure are 187.3°F and 49.1 psig, respectively, well below the design temperature of 281°F and maximum allowable pressure of 62 psig.

The RHRSW System satisfies Criterion 3 of the NRC Policy Statement (Ref 5).

LCO

Four RHRSW subsystems are required to be OPERABLE to provide the required redundancy to ensure that the system functions to remove post accident heat loads, assuming the worst case single active failure occurs coincident with the loss of offsite power. Additionally, since the RHRSW pumps are shared between the three BFN units, the number of OPERABLE pumps required is also dependent on the number of units fueled.

An RHRSW subsystem is considered OPERABLE when:

a. At least one RHRSW pump (i.e., one required RHRSW pump) is OPERABLE; and

b. An OPERABLE flow path is capable of taking suction from the intake structure and transferring the water to the associated RHR heat exchanger at the assumed flow rate.

(continued)

BASES

LCO (continued)

In addition to the required number of OPERABLE subsystems, there must be an adequate number of pumps OPERABLE to provide cooling for the fueled non-accident units.

The total number of required RHRSW pumps must take into consideration the required number of pumps required for the specific unit along with the number of pumps required for other units that are fueled. Hence, when one unit contains fuel, four RHRSW pumps are required to be OPERABLE. When two units contain fuel, six RHRSW pumps are required to be OPERABLE. When three units contain fuel, eight RHRSW pumps are required to be OPERABLE. The minimum specified number of pumps gives consideration to all units capable of producing heat in aggregate and accounts for a single active failure.

The above pre-accident configuration ensures that during a design basis accident with a postulated single active failure, the resulting configuration for the accident unit has at least two RHRSW subsystems OPERABLE to supply 100 percent of the long term RHR cooling water. The resulting configuration for the non-accident units has at least two RHRSW subsystems per unit OPERABLE to supply 100 percent of the required cooling capacity to maintain safe shutdown conditions.

(continued)

BASES

LCO (continued)

The number of required OPERABLE RHRSW pumps is modified by a Note which specifies that the number of required RHRSW pumps may be reduced by one for each fueled unit that has been in MODE 4 or 5 for over 24 hours. This Note acknowledges the fact that decay heat removal requirements are substantially reduced for fueled units in MODE 4 or 5 for over 24 hours.

APPLICABILITY

In MODES 1, 2, and 3, the RHRSW System is required to be OPERABLE to support the OPERABILITY of the RHR System for primary containment cooling (LCO 3.6.2.3, "Residual Heat Removal (RHR) Suppression Pool Cooling," and LCO 3.6.2.4, "Residual Heat Removal (RHR) Suppression Pool Spray") and decay heat removal (LCO 3.4.7, "Residual Heat Removal (RHR) Shutdown Cooling System - Hot Shutdown"). The Applicability is therefore consistent with the requirements of these systems.

In MODES 4 and 5, the OPERABILITY requirements of the RHRSW System are determined by the systems it supports.

(continued)

BASES (continued)

ACTIONS

Since the RHRSW System is common to all three units, the following requirements must be followed when multiple units contain fuel:

a. With one or more required RHRSW pumps inoperable, all applicable ACTIONS must be entered for each unit.

b. With one or more RHRSW subsystems inoperable, all applicable ACTIONS for inoperable subsystems must be entered on the unit(s) that have the inoperable subsystem.

The Required Actions and associated Completion Times of Conditions A, B, C, and D are based on a reduction in redundancy of the RHRSW System, not a loss of RHRSW safety function. The Required Actions and associated Completion Times of Conditions E, F, and G consider that the RHRSW safety function is lost.

RHRSW safety function is maintained when at least two RHRSW subsystems, with two separate RHRSW pumps (i.e. one per subsystem), on a per fueled unit basis, are OPERABLE. Additionally, the total number of RHRSW pumps must be such that the RHRSW pumps credited for maintaining the RHRSW safety function for a specific unit are not credited for maintaining the RHRSW safety function for a different fueled unit.

When there are three fueled units, the RHRSW safety function is maintained when:

- Two RHRSW subsystems per fueled unit are OPERABLE;

(continued)

BASES

ACTIONS (continued)

- Six RHRSW pumps are OPERABLE (two RHRSW pumps per fueled unit);
- Each OPERABLE RHRSW subsystem has one required RHRSW pump OPERABLE; and
- The required RHRSW pump that is OPERABLE in an RHRSW subsystem is not credited for maintaining the RHRSW safety function for another fueled unit (i.e., it is not one of the two RHRSW pumps that is required to be OPERABLE for another fueled unit).

When there are two fueled units, the RHRSW safety function is maintained when:

- Two RHRSW subsystems per fueled unit are OPERABLE;
- Four RHRSW pumps are OPERABLE (two RHRSW pumps per fueled unit);
- Each OPERABLE RHRSW subsystem has one required RHRSW pump OPERABLE; and
- The required RHRSW pump that is OPERABLE in an RHRSW subsystem is not credited for maintaining the RHRSW safety function for another fueled unit (i.e., it is not one of the two RHRSW pumps that is required to be OPERABLE for another fueled unit).

(continued)

BASES

ACTIONS (continued)

When there is one fueled unit, the RHRSW safety function is maintained when:

- Two RHRSW subsystems are OPERABLE and
- Two RHRSW pumps are OPERABLE (i.e., one per OPERABLE RHRSW subsystem).

When any combination of pump(s) and other subsystem components, e.g., heat exchanger(s), are inoperable such that three or more components of the RHRSW System (on any or all fueled units) are inoperable, the capability to meet the safety function must be evaluated by all fueled units. When an RHRSW pump is credited by one fueled unit for maintaining the RHRSW safety function, then the other fueled units cannot also credit this same RHRSW pump with maintaining their RHRSW afety function since the capacity of a single RHRSW pump is not sufficient to support the required heat removal function of more than one RHR heat exchanger. Therefore, in this condition, the RHRSW pump credited with maintaining RHRSW safety function on a fueled unit must be considered inoperable for the other fueled units for purpose of determining if RHRSW safety function is maintained. The other fueled units must then include the additional inoperable RHRSW pump(s) with the total number of inoperable components when determining if RHRSW safety function is maintained. If RHRSW safety function is determined to be lost, then Condition E or F is required to be entered.

The examples, with respect to RHRSW pumps, used in the following descriptions of the ACTIONS assume that three units are fueled.

(continued)

BASES

ACTIONS (continued)

A.1 and A.2

With one required RHRSW pump inoperable, the inoperable RHRSW pump must be restored to OPERABLE status within 30 days. With the unit in this condition, the remaining OPERABLE RHRSW pumps are adequate to perform the RHRSW heat removal function. However, the overall reliability is reduced because a single failure could result in reduced primary containment cooling capability. The 30 day Completion Time is based on the availability of equipment in excess of normal redundancy requirements and the low probability of an event occurring requiring RHRSW during this period.

Alternatively, five RHRSW pumps may be verified to be OPERABLE with power being supplied from separate 4 kV shutdown boards.

Required Action A1 is modified by two Notes. Note 1 indicates that the Required Action is applicable only when two units are fueled. In the two unit fueled condition, a single failure (loss of a 4 kV shutdown board) could result in inadequate RHRSW pumps if two pumps are powered from the same power supply. If five RHRSW pumps are powered from separate 4 kV shutdown boards, then no postulated single active failure could occur to prevent the RHRSW system from performing its design function. Operation can continue indefinitely if Required Action A.1 is met.

Note 2 requires only four RHRSW pumps powered from separate 4 kV shutdown boards to be OPERABLE if the other fueled unit has been in Mode 4 or 5 greater than 24 hours. This acknowledges the fact that decay heat removal requirements are substantially reduced for fueled units in Mode 4 or 5 for greater than 24 hours.

These two Notes clarify the situations under Required Action A.1 would be the appropriate Required Action.

(continued)

BASES

ACTIONS (continued)

B.1

With one RHRSW subsystem inoperable (e.g., one RHR heat exchanger inoperable or an RHRSW header isolated) for reasons other than inoperable RHRSW pumps, which are covered by separate Conditions, the inoperable RHRSW subsystem must be restored to OPERABLE status within 30 days. With the unit in this condition, the remaining OPERABLE RHRSW subsystems are adequate to perform the RHRSW heat removal function. However, the overall reliability is reduced because a single failure could result in reduced primary containment cooling capability. The 30 day Completion Time is based on the availability of equipment in excess of normal redundancy requirements and the low probability of an event occurring requiring RHRSW during this period.

The Required Action is modified by a Note indicating that the applicable Conditions of LCO 3.4.7 be entered and Required Actions taken if the inoperable RHRSW subsystem results in inoperable RHR shutdown cooling. This is an exception to LCO 3.0.6 and ensures the proper actions are taken for these components.

(continued)

BASES

ACTIONS (continued)

C.1

With two required RHRSW pumps inoperable (i.e., one required RHRSW pump inoperable in each of two separate RHRSW subsystems or two RHRSW pumps inoperable in the same RHRSW subsystem), the remaining RHRSW pumps are adequate to perform the RHRSW heat removal function. However, the overall reliability is reduced because a single failure of the OPERABLE RHRSW pumps could result in a loss of RHRSW function. The seven day Completion Time is based on the redundant RHRSW capabilities afforded by the OPERABLE RHRSW pumps and the low probability of an event occurring during this period.

D.1

With two RHRSW subsystems inoperable (e.g., two RHR heat exchangers inoperable) for reasons other than inoperable RHRSW pumps, which are covered by separate Conditions, the remaining OPERABLE RHRSW subsystems are adequate to perform the RHRSW heat removal function. However, the overall reliability is reduced because a single failure could result in reduced primary containment cooling capability. The seven day Completion Time is based on the availability of equipment in excess of normal redundancy requirements and the low probability of an event occurring requiring RHRSW during this period.

The Required Action is modified by a Note indicating that the applicable Conditions of LCO 3.4.7 be entered and Required Actions taken if the inoperable RHRSW subsystem results in inoperable RHR shutdown cooling. This is an exception to LCO 3.0.6 and ensures the proper actions are taken for these components.

(continued)

BASES

ACTIONS (continued)

E.1

With three or more required RHRSW pumps inoperable, the RHRSW System is not capable of performing its intended function. The requisite number of pumps must be restored to OPERABLE status within eight hours. The eight hour Completion Time is based on the Completion Times provided for the RHR suppression pool cooling and spray functions.

F.1

With three or more required RHRSW subsystems inoperable (e.g., one RHR heat exchanger inoperable in each of three or four separate RHRSW subsystems) for reasons other than inoperable RHRSW pumps, which are covered by separate Conditions, the RHRSW System is not capable of performing its intended function. The requisite number of subsystems must be restored to OPERABLE status within eight hours. The eight hour Completion Time is based on the Completion Times provided for the RHR suppression pool cooling and spray functions.

The Required Action is modified by a Note indicating that the applicable Conditions of LCO 3.4.7 be entered and Required Actions taken if the inoperable RHRSW subsystem results in inoperable RHR shutdown cooling. This is an exception to LCO 3.0.6 and ensures the proper actions are taken for these components.

(continued)

BASES

ACTIONS (continued)

G.1 and G.2

If the RHRSW subsystem(s) or the RHRSW pump(s) cannot be restored to OPERABLE status within the associated Completion Times, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 12 hours and in MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.1.1

Verifying the correct alignment for each manual and power operated valve in each RHRSW subsystem flow path provides assurance that the proper flow paths will exist for RHRSW operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since these valves are verified to be in the correct position prior to locking, sealing, or securing. A valve is also allowed to be in the nonaccident position, and yet considered in the correct position, provided it can be realigned to its accident position. This is acceptable because the RHRSW System is a manually initiated system.

This SR does not require any testing or valve manipulation; rather, it involves verification that those valves capable of being mispositioned are in the correct position. This SR does not apply to valves that cannot be inadvertently misaligned, such as check valves.

The 31 day Frequency is based on engineering judgment, is consistent with the procedural controls governing valve operation, and ensures correct valve positions.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

REFERENCES

1. FSAR, Section 10.9.
2. FSAR, Chapter 5.
3. FSAR, Chapter 14.
4. FSAR, Section 14.6.3.3.2.
5. NRC No. 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.

BASES

LCO (continued)

The EECW System is considered OPERABLE when it has an OPERABLE UHS, three OPERABLE pumps, and two OPERABLE flow paths capable of taking suction from the intake structure and transferring the water to the appropriate equipment.

The OPERABILITY of the UHS for EECW is based on having a maximum water temperature of 95°F.

The isolation of the EECW System to components or systems may render those components or systems inoperable, but does not affect the OPERABILITY of the EECW System.

APPLICABILITY

In MODES 1, 2, and 3, the EECW System and UHS are required to be OPERABLE to support OPERABILITY of the equipment serviced by the EECW System. Therefore, the EECW System and UHS are required to be OPERABLE in these MODES.

In MODES 4 and 5, the OPERABILITY requirements of the EECW System and UHS are determined by the systems they support.

ACTIONS

A.1

With one required EECW pump inoperable, the required EECW pump must be restored to OPERABLE status within 7 days. With the system in this condition, the remaining OPERABLE EECW pumps are adequate to perform the heat removal function. However, the overall reliability is reduced because a single failure in the EECW System could result in loss of EECW function.

(continued)

BASES

ACTIONS

A.1 (continued)

The 7 day Completion Time is based on the redundant EECW System capabilities afforded by the remaining OPERABLE pumps, the low probability of an accident occurring during this time period and is consistent with the allowed Completion Time for restoring an inoperable DG.

B.1 and B.2

If the required EECW pump cannot be restored to OPERABLE status within the associated Completion Time, or two or more EECW pumps are inoperable or the UHS is determined inoperable, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 12 hours and in MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.2.1

Verification of the UHS temperature ensures that the heat removal capability of the EECW System is within the assumptions of the DBA analysis. The 24 hour Frequency is based on operating experience related to trending of the parameter variations during the applicable MODES.

(continued)

B 3.7 PLANT SYSTEMS

B 3.7.5 Main Turbine Bypass System

BASES

BACKGROUND

The Main Turbine Bypass System is designed to control steam pressure when reactor steam generation exceeds turbine requirements during unit startup, sudden load reduction, and cooldown. It allows excess steam flow from the reactor to the condenser without going through the turbine. The bypass capacity of the system is 23% of the Nuclear Steam Supply System rated steam flow. Sudden load reductions within the capacity of the steam bypass can be accommodated without reactor scram. The Main Turbine Bypass System consists of nine valves connected to the main steam lines between the main steam isolation valves and the turbine stop valve bypass valve chest. Each of these nine valves is operated by hydraulic cylinders. The bypass valves are controlled by the pressure regulation function of the Pressure Regulator and Turbine Generator Control System, as discussed in the FSAR, Section 7.11.3.3 (Ref. 1). The bypass valves are normally closed, and the pressure regulator controls the turbine control valves that direct all steam flow to the turbine. If the speed governor or the load limiter restricts steam flow to the turbine, the pressure regulator controls the system pressure by opening the bypass valves. When the bypass valves open, the steam flows from the bypass chest, through connecting piping, to the pressure breakdown assemblies, where a series of orifices are used to further reduce the steam pressure before the steam enters the condenser.

(continued)

BASES (continued)

APPLICABILITY

The Main Turbine Bypass System is required to be OPERABLE at ≥ 23% RTP to ensure that the fuel cladding integrity Safety Limit is not violated during abnormal operational transients. As discussed in the Bases for LCO 3.2.1 and LCO 3.2.2, sufficient margin to these limits exists at < 23% RTP. Therefore, these requirements are only necessary when operating at or above this power level.

ACTIONS

A.1

If the Main Turbine Bypass System is inoperable (one or more bypass valves inoperable), or the APLHGR, MCPR, and LHGR limits for an inoperable Main Turbine Bypass System, as specified in the COLR, are not applied, the assumptions of the design basis transient analysis may not be met. Under such circumstances, prompt action should be taken to restore the Main Turbine Bypass System to OPERABLE status or adjust the APLHGR, MCPR, and LHGR limits accordingly. The 2 hour Completion Time is reasonable, based on the time to complete the Required Action and the low probability of an event occurring during this period requiring the Main Turbine Bypass System.

B.1

If the Main Turbine Bypass System cannot be restored to OPERABLE status or the APLHGR, MCPR, and LHGR limits for an inoperable Main Turbine Bypass System are not applied, THERMAL POWER must be reduced to < 23% RTP. As discussed in the Applicability section, operation at < 23% RTP results in sufficient margin to the required limits, and the Main

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

2.1.1.1 Fuel Cladding Integrity

The SPCB critical power correlation is used for AREVA fuel and is valid at pressures ≥ 700 psia and bundle mass fluxes ≥ 0.1 x 10^6 lb_m/hr-ft^2 (>12,000 lb_m/hr, i.e., >10% core flow, on a per bundle basis). For thermal margin monitoring at 23% power and higher, the hot channel flow rate will be >28,000 lb_m/hr (core flow not less than natural circulation, i.e., ~25%-30% core flow for 23% power); therefore, the fuel cladding integrity SL is conservative relative to the applicable range of the SPCB critical power correlation. For operation at low pressures or low flows, another basis is used, as follows:

> The static head across the fuel bundles due only to elevation effects from liquid only in the channel, core bypass region, and annulus at zero power, zero flow is approximately 4.5 psi. At all operating conditions, this pressure differential is maintained by the bypass region of the core and the annulus region of the vessel. The elevation head provided by the annulus produces natural circulation flow conditions which have balancing pressure head and loss terms inside the core shroud. This natural circulation principle maintains a core plenum to plenum pressure drop of about 4.5 to 5 psid along the natural circulation flow line of the P/F operating map. In the range of power levels of interest, approaching 23% of rated power below which thermal margin monitoring is not required, the pressure drop and density head terms tradeoff for power changes such that natural circulation flow is nearly independent of reactor power. This characteristic is represented by the nearly vertical portion of the natural circulation line on the P/F operating map. Analysis has shown that the hot channel flow rate is >28,000 lb_m/hr (>0.23 x 10^6 lb_m/hr-ft^2) in the region of operation with power ~23% and core pressure drop of about 4.5 to 5 psid. Full scale ATLAS test data taken at pressures from 14.7 psia to 800 psia indicate that the fuel assembly critical power at 28,000 lb_m/hr is approximately 3

(continued)

BASES

APPLICABLE SAFETY ANALYSES

2.1.1.1 Fuel Cladding Integrity (continued)

MW_t. With the design peaking factors, this corresponds to a core thermal power of more than 50%.

Thus operation up to 23% of rated power with normal natural circulation available is conservatively acceptable even if reactor pressure is equal to or below the lower pressure limit of the SPCB correlation. If reactor power is significantly less than 23% of rated (e.g., below 10% of rated), the core flow and the channel flow supported by the available driving head may be less than 28,000 lb_m/hr (along the lower portion of the natural circulation flow characteristic on the P/F map). However, the critical power that can be supported by the core and hot channel flow with normal natural circulation paths available remains well above the actual power conditions. The inherent characteristics of BWR natural circulation make power and core flow follow the natural circulation line as long as normal water level is maintained.

Thus, operation with core thermal power below 23% of rated without thermal margin surveillance is conservatively acceptable even for reactor operations at natural circulation. Adequate fuel thermal margins are also maintained without further surveillance for the low power conditions that would be present if core natural circulation is below the lower flow limit of the SPCB correlation.

(continued)

BASES

BACKGROUND (continued)

The worst case sodium pentaborate solution concentration required to shutdown the reactor with sufficient margin to account for 0.05 $\Delta k/k$ and Xenon poisoning effects is 9.2 weight percent. This corresponds to a 40°F saturation temperature. The worst case SLCS equipment area temperature is not predicted to fall below 50°F. This provides a 10°F thermal margin to unwanted precipitation of the sodium pentaborate. Tank heating components provide backup assurance that the sodium pentaborate solution temperature will never fall below 50°F but are not required for TS operability considerations.

APPLICABLE SAFETY ANALYSES

The SLC System is manually initiated from the main control room, as directed by the emergency operating instructions, if the operator believes the reactor cannot be shut down, or kept shut down, with the control rods. The SLC System is used in the event that enough control rods cannot be inserted to accomplish shutdown and cooldown in the normal manner. The SLC System injects borated water into the reactor core to add negative reactivity to compensate for all of the various reactivity effects that could occur during plant operations. To meet this objective, it is necessary to inject a quantity of boron, which produces a concentration of 720 ppm of natural boron, in the reactor coolant at 70°F. To allow for imperfect mixing, leakage and the volume in other piping connected to the reactor system, an amount of boron equal to 25% of the amount cited above is added (Ref. 2). This volume versus concentration limit and the temperature versus concentration limits in Figure 3.1.7-1 are calculated such that the required concentration is achieved accounting for dilution in the RPV with normal water level and including the water volume in the entire residual heat removal shutdown cooling piping and in the recirculation loop piping. This quantity of borated solution is the amount that is above the pump suction shutoff level in the boron solution storage tank. No credit is taken for the portion of the tank volume that cannot be injected.

(continued)

SURVEILLANCE REQUIREMENTS

SR 3.1.7.1 (continued)

pentaborate solution concentration requirements (≤ 9.2% by weight) and the required quantity of Boron-10 (≥ 203 lbs) establish the tank volume requirement. The 24 hour Frequency is based on operating experience that has shown there are relatively slow variations in the solution volume.

SR 3.1.7.2

SR 3.1.7.2 verifies the continuity of the explosive charges in the injection valves to ensure that proper operation will occur if required. An automatic continuity monitor may be used to continuously satisfy this requirement. Other administrative controls, such as those that limit the shelf life of the explosive charges, must be followed. The 31 day Frequency is based on operating experience and has demonstrated the reliability of the explosive charge continuity.

SR 3.1.7.3

SR 3.1.7.3 requires an examination of sodium pentaborate solution by using chemical analysis to ensure that the proper concentration of boron exists in the storage tank for post-LOCA suppression pool pH control. This parameter is used as input to determine the volume requirements for SR 3.1.7.1. The concentration is dependent upon the volume of water and quantity of boron in the storage tank.

SR 3.1.7.3 must be performed every 31 days or within 24 hours of when boron or water is added to the storage tank solution to determine that the boron solution concentration is within the specified limits. The 31 day Frequency of this Surveillance is appropriate because of the relatively slow variation of boron concentration between surveillances.

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SR 3.1.7.4 and SR 3.1.7.6

SR 3.1.7.4 requires an examination of the sodium pentaborate solution by using chemical analysis to ensure that the proper concentration of boron exists in the storage tank. The concentration is dependent upon the volume of water and quantity of boron in the storage tank. SR 3.1.7.6 requires verification that the SLC system conditions satisfy the following equation:

$$\frac{(\ C\ )(\ Q\ )(\ E\ )}{(8.7\ \mathrm{WT}\ \%)(50\ \mathrm{GPM})(94\ \mathrm{ATOM}\ \%)} >= 1.0$$

C = sodium pentaborate solution weight percent concentration
Q = SLC system pump flow rate in gpm
E = Boron-10 atom percent enrichment in the sodium pentaborate solution

To meet 10 CFR 50.62, the SLC System must have a minimum flow capacity and boron content equivalent in control capacity to 86 gpm of 13 weight percent natural sodium pentaborate solution. The atom percentage of natural B-10 is 19.8%. This equivalency requirement is met when the equation given above is satisfied. The equation can be satisfied by adjusting the solution concentration, pump flow rate or Boron-10 enrichment. If the results of the equation are < 1, the SLC System is no longer capable of shutting down the reactor with the margin described in Reference 2. As described in Reference 2, the BFN analysis assumes a flow capacity and boron content equivalent to 50 gpm of 8.7 weight percent and 94 atom percent B-10 enriched sodium pentaborate solution. This exceeds the requirement of 10 CFR 50.62, and the equation is adjusted to reflect the BFN requirements. The quantity of stored boron includes an additional margin (25%) beyond the amount needed to shut down the reactor to allow for possible imperfect mixing of the chemical solution in the reactor water, leakage, and the volume in other piping connected to the reactor system.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.1.7.4 and SR 3.1.7.6

The sodium pentaborate solution (SPB) concentration is allowed to be > 9.2 weight percent provided the concentration and temperature of the sodium pentaborate solution are verified

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.1.7.11

SR 3.1.7.11 verifies that each valve in the system is in its correct position, but does not apply to the squib (i.e., explosive) valves. Verifying the correct alignment for manual, power operated, and automatic valves in the SLC System Flowpath provides assurance that the proper flow paths will exist for system operation. A valve is also allowed to be in the nonaccident position provided it can be aligned to the accident position from the control room, or locally by a dedicated operator at the valve control. This is acceptable since the SLC System is a manually initiated system. This surveillance also does not apply to valves that are locked, sealed, or otherwise secured in position since they are verified to be in the correct position prior to locking, sealing or securing. This verification of valve alignment does not require any testing or valve manipulation; rather, it involves verification that those valves capable of being mispositioned are in the correct position. This SR does not apply to valves that cannot be inadvertently misaligned, such as check valves. The 31 day Frequency is based on engineering judgment and is consistent with the procedural controls governing valve operation that ensures correct valve positions.

REFERENCES

1. 10 CFR 50.62.
2. NEDC-33860P, “Safety Analysis Report for Browns Ferry Nuclear Plant Units 1, 2, and 3 Extended Power Uprate,” Section 2.8.
3. NRC No. 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.
4. FSAR, Section 14.6.

BASES (continued)

APPLICABILITY

APLHGR limits are primarily derived from fuel design evaluations and LOCA and transient analyses assumed to occur at high power levels. Design calculations and operating experience have shown that as power is reduced, the margin to the required APLHGR limits increases. This trend continues down to the power range of 5% to 15% RTP when entry into MODE 2 occurs. When in MODE 2, the intermediate range monitor scram function provides prompt scram initiation during any significant transient, thereby effectively removing any APLHGR limit compliance concern in MODE 2. Therefore, at THERMAL POWER levels ≤ 23% RTP, the reactor is operating with substantial margin to the APLHGR limits; thus, this LCO is not required.

(continued)

BASES (continued)

ACTIONS

A.1

If any APLHGR exceeds the required limits, an assumption regarding an initial condition of the DBA and transient analyses may not be met. Therefore, prompt action should be taken to restore the APLHGR(s) to within the required limits such that the plant operates within analyzed conditions and within design limits of the fuel rods. The 2 hour Completion Time is sufficient to restore the APLHGR(s) to within its limits and is acceptable based on the low probability of a transient or DBA occurring simultaneously with the APLHGR out of specification.

B.1

If the APLHGR cannot be restored to within its required limits within the associated Completion Time, the plant must be brought to a MODE or other specified condition in which the LCO does not apply. To achieve this status, THERMAL POWER must be reduced to < 23% RTP within 4 hours. The allowed Completion Time is reasonable, based on operating experience, to reduce THERMAL POWER to < 23% RTP in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.2.1.1

APLHGRs are required to be initially calculated within 12 hours after THERMAL POWER is ≥ 23% RTP and then every 24 hours thereafter. They are compared to the specified limits in the COLR to ensure that the reactor is operating within the assumptions of the safety analysis. The 24 hour Frequency is based on both engineering judgment and recognition of the slowness of changes in power distribution during normal operation. The 12 hour allowance after THERMAL POWER ≥ 23% RTP is achieved is acceptable given the large inherent margin to operating limits at low power levels.

(continued)

APPLICABLE SAFETY ANALYSES (continued)

evaluated are loss of flow, increase in pressure and power, positive reactivity insertion, and coolant temperature decrease. The limiting transient yields the largest change in CPR (ΔCPR). When the largest ΔCPR is added to the MCPR SL, the required operating limit MCPR is obtained.

The MCPR operating limits derived from the transient analysis are dependent on the operating core flow and power to ensure adherence to fuel design limits during the worst transient that occurs with moderate frequency. Flow dependent MCPR ($MCPR_f$) limits are determined by steady-state thermal hydraulic methods using the three-dimensional BWR simulator code (Reference 12) and the multichannel thermal hydraulic code (Reference 13). The operating limit is dependent on the maximum core flow limiter setting in the Recirculation Flow Control System.

Power-dependent MCPR limits ($MCPR_p$) are determined by the three-dimensional BWR simulator code (Ref. 12) and the one-dimensional transient codes (Refs. 14 and 15). Due to the sensitivity of the transient response to initial core flow levels at power levels below those at which the turbine stop valve closure and turbine control valve fast closure scrams are bypassed, high and low flow $MCPR_p$ operating limits are provided for operating between 23% RTP and the previously mentioned bypass power level.

The MCPR satisfies Criterion 2 of the NRC Policy Statement (Ref. 7).

(continued)

BASES (continued)

LCO

The MCPR operating limits specified in the COLR are the result of the Design Basis Accident (DBA) and transient analysis. Additional MCPR operating limits may be provided in the COLR to support analyzed equipment out-of-service operation. The operating limit MCPR is determined by the larger of the $MCPR_f$ and $MCPR_p$ limits.

APPLICABILITY

The MCPR operating limits are primarily derived from transient analyses that are assumed to occur at high power levels. Below 23% RTP, the reactor is operating at a minimum recirculation pump speed and the moderator void ratio is small. Surveillance of thermal limits below 23% RTP is unnecessary due to the large inherent margin that ensures that the MCPR SL is not exceeded even if a limiting transient occurs. Statistical analyses indicate that the nominal value of the initial MCPR expected at 23% RTP is > 3.5. Studies of the variation of limiting transient behavior have been performed over the range of power and flow conditions. These studies encompass the range of key actual plant parameter values important to typically limiting transients. The results of these studies demonstrate that a margin is expected between performance and the MCPR requirements, and that margins increase as power is reduced to 23% RTP. This trend is expected to continue to the 5% to 15% power range when entry into MODE 2 occurs. When in MODE 2, the intermediate range monitor provides rapid scram initiation for any significant power increase transient, which effectively eliminates any MCPR compliance concern. Therefore, at THERMAL POWER levels < 23% RTP, the reactor is operating with substantial margin to the MCPR limits and this LCO is not required.

(continued)

BASES (continued)

ACTIONS

A.1

If any MCPR is outside the required limits, an assumption regarding an initial condition of the design basis transient analyses may not be met. Therefore, prompt action should be taken to restore the MCPR(s) to within the required limits such that the plant remains operating within analyzed conditions. The 2 hour Completion Time is normally sufficient to restore the MCPR(s) to within its limits and is acceptable based on the low probability of a transient or DBA occurring simultaneously with the MCPR out of specification.

B.1

If the MCPR cannot be restored to within its required limits within the associated Completion Time, the plant must be brought to a MODE or other specified condition in which the LCO does not apply. To achieve this status, THERMAL POWER must be reduced to < 23% RTP within 4 hours. The allowed Completion Time is reasonable, based on operating experience, to reduce THERMAL POWER to < 23% RTP in an orderly manner and without challenging plant systems.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.2.2.1

The MCPR is required to be initially calculated within 12 hours after THERMAL POWER is ≥ 23% RTP and then every 24 hours thereafter. It is compared to the specified limits in the COLR to ensure that the reactor is operating within the assumptions of the safety analysis. The 24 hour Frequency is based on both engineering judgment and recognition of the slowness of changes in power distribution during normal operation. The 12 hour allowance after THERMAL POWER ≥ 23% RTP is achieved is acceptable given the large inherent margin to operating limits at low power levels.

SR 3.2.2.2

Because the transient analysis takes credit for conservatism in the scram speed performance, it must be demonstrated that the specific scram speed distribution is consistent with that used in the transient analysis. SR 3.2.2.2 determines the actual scram speed distribution and compares it with the assumed distribution. The MCPR operating limit is determined based either on the applicable limit associated with scram times of LCO 3.1.4, “Control Rod Scram Times,” or the nominal scram times. The scram speed-dependent MCPR limits are contained in the COLR. This determination must be performed within 72 hours after each set of control rod scram time tests required by SR 3.1.4.1 and SR 3.1.4.2 because the effective scram speed distribution may change during the cycle. The 72-hour Completion Time is acceptable due to the relatively minor changes in the actual control rod scram speed distribution expected during the fuel cycle.

(continued)

BASES (continued)

LCO	The LHGR is a basic assumption in the fuel design analysis. The fuel has been designed to operate at rated core power with sufficient design margin to the LHGR calculated to cause a 1% fuel cladding plastic strain. The operating limit to accomplish this objective is specified in the COLR. Additional LHGR operating limits adjustments may be provided in the COLR to support analyzed equipment out-of-service operation. Additional LHGR operating limits adjustments may be provided in the COLR to support analyzed equipment out-of-service operation.
APPLICABILITY	The LHGR limits are derived from fuel design analysis that is limiting at high power level conditions. At core thermal power levels < 23% RTP, the reactor is operating with a substantial margin to the LHGR limits and, therefore, the Specification is only required when the reactor is operating at ≥ 23% RTP.

(continued)

BASES (continued)

ACTIONS

A.1

If any LHGR exceeds its required limit, an assumption regarding an initial condition of the fuel design analysis is not met. Therefore, prompt action should be taken to restore the LHGR(s) to within its required limits such that the plant is operating within analyzed conditions. The 2 hour Completion Time is normally sufficient to restore the LHGR(s) to within its limits and is acceptable based on the low probability of a transient or Design Basis Accident occurring simultaneously with the LHGR out of specification.

B.1

If the LHGR cannot be restored to within its required limits within the associated Completion Time, the plant must be brought to a MODE or other specified condition in which the LCO does not apply. To achieve this status, THERMAL POWER is reduced to < 23% RTP within 4 hours. The allowed Completion Time is reasonable, based on operating experience, to reduce THERMAL POWER to < 23% RTP in an orderly manner and without challenging plant systems.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.2.3.1

The LHGR is required to be initially calculated within 12 hours after THERMAL POWER is ≥ 23% RTP and then every 24 hours thereafter. It is compared to the specified limits in the COLR to ensure that the reactor is operating within the assumptions of the safety analysis. The 24 hour Frequency is based on both engineering judgment and recognition of the slow changes in power distribution during normal operation. The 12 hour allowance after THERMAL POWER ≥ 23% RTP is achieved is acceptable given the large inherent margin to operating limits at lower power levels.

REFERENCES

1. FSAR, Chapter 14.
2. FSAR, Chapter 3.
3. NUREG-0800, Standard Review Plan 4.2, Section II.A.2(g), Revision 2, July 1981.
4. NRC No. 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)	The trip setpoints are then determined accounting for the remaining instrument errors (e.g., drift). The trip setpoints derived in this manner provide adequate protection because instrumentation uncertainties, process effects, calibration tolerances, instrument drift, and severe environmental effects (for channels that must function in harsh environments as defined by 10 CFR 50.49) are accounted for.

The OPERABILITY of scram pilot valves and associated solenoids, backup scram valves, and SDV valves, described in the Background section, are not addressed by this LCO.

The individual Functions are required to be OPERABLE in the MODES or other specified conditions in the Table, which may require an RPS trip to mitigate the consequences of a design basis accident or transient. To ensure a reliable scram function, a combination of Functions are required in each MODE to provide primary and diverse initiation signals.

The only MODES specified in Table 3.3.1.1-1 are MODES 1 (which encompasses ≥ 26% RTP) and 2, and MODE 5 with any control rod withdrawn from a core cell containing one or more fuel assemblies. No RPS Function is required in MODES 3 and 4 since all control rods are fully inserted and the Reactor Mode Switch Shutdown Position control rod withdrawal block (LCO 3.3.2.1) does not allow any control rod to be withdrawn. In MODE 5, control rods withdrawn from a core cell containing no fuel assemblies do not affect the reactivity of the core and, therefore, are not required to have the capability to scram. Provided all other control rods remain inserted, no RPS function is required. In this condition, the required SDM (LCO 3.1.1) and refuel position one-rod-out interlock (LCO 3.9.2) ensure that no event requiring RPS will occur.

The specific Applicable Safety Analyses, LCO, and Applicability discussions are listed below on a Function by Function basis.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

2.a. Average Power Range Monitor Neutron Flux - High, (Setdown)

For operation at low power (i.e., MODE 2), the Average Power Range Monitor Neutron Flux - High, (Setdown) Function is capable of generating a trip signal that prevents fuel damage resulting from abnormal operating transients in this power range. For most operation at low power levels, the Average Power Range Monitor Neutron Flux - High, (Setdown) Function will provide a secondary scram to the Intermediate Range Monitor Neutron Flux - High Function because of the relative setpoints. With the IRMs at Range 9 or 10, it is possible that the Average Power Range Monitor Neutron Flux - High, (Setdown) Function will provide the primary trip signal for a corewide increase in power.

No specific safety analyses take direct credit for the Average Power Range Monitor Neutron Flux - High, (Setdown) Function. However, this Function indirectly ensures that before the reactor mode switch is placed in the run position, reactor power does not exceed 23% RTP (SL 2.1.1.1) when operating at low reactor pressure and low core flow. Therefore, it indirectly prevents fuel damage during significant reactivity increases with THERMAL POWER < 23% RTP.

The Allowable Value is based on preventing significant increases in power when THERMAL POWER is < 23% RTP.

The Average Power Range Monitor Neutron Flux - High, (Setdown) Function must be OPERABLE during MODE 2 when control rods may be withdrawn since the potential for criticality exists. In MODE 1, the Average Power Range Monitor Neutron Flux - High Function provides protection against reactivity transients and the RWM and rod block monitor protect against control rod withdrawal error events.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

2.f. Oscillation Power Range Monitor (OPRM) Upscale

The OPRM Upscale Function provides compliance with GDC 10 and GDC 12, thereby providing protection from exceeding the fuel MCPR safety limit (SL) due to anticipated thermal hydraulic power oscillations.

References 13, 14, and 15 describe three algorithms for detecting thermal hydraulic instability related neutron flux oscillations: the period based detection algorithm, the amplitude based algorithm, and the growth rate algorithm. All three are implemented in the OPRM Upscale Function, but the safety analysis takes credit only for the period based detection algorithm. The remaining algorithms provide defense in depth and additional protection against unanticipated oscillations. OPRM Upscale Function OPERABILITY for Technical Specification purposes is based only on the period based detection algorithm.

The OPRM Upscale Function receives input signals from the local power range monitors (LPRMs) within the reactor core, which are combined into “cells” for evaluation of the OPRM algorithms.

The OPRM Upscale Function is required to be OPERABLE when the plant is in a region of power flow operation where anticipated events could lead to thermal hydraulic instability and related neutron flux oscillations. Within this region, the automatic trip is enabled when THERMAL POWER, as indicated by the APRM Simulated Thermal Power, is ≥ 23% RTP and reactor core flow, as indicated by recirculation drive flow is < 60% of rated flow, the operating region where actual thermal hydraulic oscillations may occur. Requiring the OPRM Upscale Function to be OPERABLE in MODE 1 provides consistency with operability requirements for other APRM functions and assures that the OPRM Upscale Function is OPERABLE whenever reactor power could increase into the region of concern without operator action.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

8. Turbine Stop Valve - Closure (continued)

Turbine Stop Valve - Closure signals are initiated from position switches located on each of the four TSVs. Two independent position switches are associated with each stop valve. One of the two switches provides input to RPS trip system A; the other, to RPS trip system B. Thus, each RPS trip system receives an input from four Turbine Stop Valve - Closure channels, each consisting of one position switch. The logic for the Turbine Stop Valve - Closure Function is such that three or more TSVs must be closed to produce a scram. This Function must be enabled at THERMAL POWER ≥ 26% RTP. This is normally accomplished automatically by pressure transmitters sensing turbine first stage pressure; therefore, opening the turbine bypass valves may affect this function.

The Turbine Stop Valve - Closure Allowable Value is selected to be high enough to detect imminent TSV closure, thereby reducing the severity of the subsequent pressure transient.

Eight channels of Turbine Stop Valve - Closure Function, with four channels in each trip system, are required to be OPERABLE to ensure that no single instrument failure will preclude a scram from this Function if any three TSVs should close. This Function is required, consistent with analysis assumptions, whenever THERMAL POWER is ≥ 26% RTP. This Function is not required when THERMAL POWER is < 26% RTP since the Reactor Vessel Steam Dome Pressure - High and the Average Power Range Monitor Fixed Neutron Flux - High Functions are adequate to maintain the necessary safety margins.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

9. Turbine Control Valve Fast Closure, Trip Oil Pressure - Low (PS-47-142, PS-47-144, PS-47-146, and PS-47-148)

Fast closure of the TCVs results in the loss of a heat sink that produces reactor pressure, neutron flux, and heat flux transients that must be limited. Therefore, a reactor scram is initiated on TCV fast closure in anticipation of the transients that would result from the closure of these valves. The Turbine Control Valve Fast Closure, Trip Oil Pressure - Low Function is the primary scram signal for the generator load rejection event analyzed in Reference 7. For this event, the reactor scram reduces the amount of energy required to be absorbed and, along with the actions of the EOC-RPT System, ensures that the MCPR SL is not exceeded.

Turbine Control Valve Fast Closure, Trip Oil Pressure - Low signals are initiated by the electrohydraulic control (EHC) fluid pressure at each control valve. One pressure switch is associated with each control valve, and the signal from each switch is assigned to a separate RPS logic channel. This Function must be enabled at THERMAL POWER ≥ 26% RTP. This is normally accomplished automatically by pressure transmitters sensing turbine first stage pressure; therefore, opening the turbine bypass valves may affect this function.

The Turbine Control Valve Fast Closure, Trip Oil Pressure - Low Allowable Value is selected high enough to detect imminent TCV fast closure.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

9. Turbine Control Valve Fast Closure, Trip Oil Pressure - Low (PS-47-142, PS-47-144, PS-47-146, and PS-47-148) (continued)

Four channels of Turbine Control Valve Fast Closure, Trip Oil Pressure - Low Function with two channels in each trip system arranged in a one-out-of-two logic are required to be OPERABLE to ensure that no single instrument failure will preclude a scram from this Function on a valid signal. This Function is required, consistent with the analysis assumptions, whenever THERMAL POWER is ≥ 26% RTP. This Function is not required when THERMAL POWER is < 26% RTP, since the Reactor Vessel Steam Dome Pressure - High and the Average Power Range Monitor Fixed Neutron Flux - High Functions are adequate to maintain the necessary safety margins.

For this instrument function, the nominal trip setpoint including the as-left tolerances is defined as the LSSS. The acceptable as-found band is based on a statistical combination of possible measurable uncertainties (i.e., setting tolerance, drift, temperature effects, and measurement and test equipment). During instrument calibrations, if the as-found setpoint is found to be conservative with respect to the Allowable Value, but outside its acceptable as-found band (tolerance range), as defined by its associated Surveillance Requirement procedure, then there shall be an initial determination to ensure confidence that the channel can perform as required before returning the channel to service in accordance with the Surveillance. The technician performing the Surveillance will evaluate the instrument's ability to maintain a stable setpoint within the as-left tolerance. The technician's evaluation will be reviewed by on shift personnel during the approval of the Surveillance data prior to returning the channel back to service at the completion of the Surveillance. This shall constitute the initial determination of operability. If a channel is found to exceed the channel's Allowable Value or cannot be reset within the

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.1.1.2

To ensure that the APRMs are accurately indicating the true core average power, the APRMs are calibrated to the reactor power calculated from a heat balance. The Frequency of once per 7 days is based on minor changes in LPRM sensitivity, which could affect the APRM reading, between performances of SR 3.3.1.1.7.

A restriction to satisfying this SR when < 23% RTP is provided that requires the SR to be met only at ≥ 23% RTP because it is difficult to accurately maintain APRM indication of core THERMAL POWER consistent with a heat balance when < 23% RTP. At low power levels, a high degree of accuracy is unnecessary because of the large, inherent margin to thermal limits (MCPR and APLHGR). At ≥ 23% RTP, the Surveillance is required to have been satisfactorily performed within the last 7 days, in accordance with SR 3.0.2. A Note is provided which allows an increase in THERMAL POWER above 23% if the 7 day Frequency is not met per SR 3.0.2. In this event, the SR must be performed within 12 hours after reaching or exceeding 23% RTP. Twelve hours is based on operating experience and in consideration of providing a reasonable time in which to complete the SR.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.1.1.15

This SR ensures that scrams initiated from the Turbine Stop Valve - Closure and Turbine Control Valve Fast Closure, Trip Oil Pressure - Low Functions will not be inadvertently bypassed when THERMAL POWER is ≥ 26% RTP. This involves calibration of the bypass channels (PIS-1-81A, PIS-1-81B, PIS-1-91A, and PIS-1-91B). Adequate margins for the instrument setpoint methodologies are incorporated into the actual setpoint.

If any bypass channel's setpoint is nonconservative (i.e., the Functions are bypassed at ≥ 26% RTP, either due to open main turbine bypass valve(s) or other reasons), then the affected Turbine Stop Valve - Closure and Turbine Control Valve Fast Closure, Trip Oil Pressure - Low Functions are considered inoperable. Alternatively, the bypass channel can be placed in the conservative condition (nonbypass). If placed in the nonbypass condition (Turbine Stop Valve - Closure and Turbine Control Valve Fast Closure, Trip Oil Pressure - Low Functions are enabled), this SR is met and the channel is considered OPERABLE.

The Frequency of 24 months is based upon the assumption of a 24 month calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.1.1.17

This SR ensures that scrams initiated from OPRM Upscale Function (Function 2.f) will not be inadvertently bypassed when THERMAL POWER, as indicated by the APRM Simulated Thermal Power, is ≥ 23% RTP and core flow, as indicated by recirculation drive flow, is < 60% rated core flow. This normally involves confirming the bypass setpoints. Adequate margins for the instrument setpoint methodologies are incorporated into the actual setpoint. The actual surveillance ensures that the OPRM Upscale Function is enabled (not bypassed) for the correct values of APRM Simulated Thermal Power and recirculation drive flow. Other surveillances ensure that the APRM Simulated Thermal Power and recirculation flow properly correlate with THERMAL POWER and core flow, respectively.

If any bypass setpoint is nonconservative (i.e., the OPRM Upscale Function is bypassed when APRM Simulated Thermal Power ≥ 23% RTP and recirculation drive flow < 60% rated), then the affected channel is considered inoperable for the OPRM Upscale Function. Alternatively, the bypass setpont may be adjusted to place the channel in a conservative condition (unbypass). If placed in the unbypassed condition, this SR is met and the channel is considered OPERABLE.

The frequency of 24 months is based on engineering judgment and reliability of the components.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

The feedwater and main turbine high water level trip instrumentation is assumed to be capable of providing a turbine trip in the design basis transient analysis for a feedwater controller failure, maximum demand event (Ref. 1). The reactor vessel high water level trip indirectly initiates a reactor scram from the main turbine trip (above 26% RTP) and trips the feedwater pumps, thereby terminating the event. The reactor scram mitigates the reduction in MCPR.

Feedwater and main turbine high water level trip instrumentation satisfies Criterion 3 of the NRC Policy Statement (Ref. 3).

LCO

The LCO requires two channels of the Reactor Vessel Water Level - High instrumentation per trip system to be OPERABLE to ensure that no single instrument failure will prevent the feedwater pump turbines and main turbine trip on a valid Reactor Vessel Water Level - High signal. Both channels in either trip system are needed to provide trip signals in order for the feedwater and main turbine trips to occur. Each channel must have its setpoint set within the specified Allowable Value of SR 3.3.2.2.3. The Allowable Value is set to ensure that the thermal limits are not exceeded during the event. The actual setpoint is calibrated to be consistent with the applicable setpoint methodology assumptions. Nominal trip setpoints are specified in the setpoint calculations. The nominal setpoints are selected to ensure that the setpoints do not exceed the Allowable Value between successive CHANNEL CALIBRATIONS. Operation with a trip setpoint less conservative than the nominal trip setpoint, but within its Allowable Value, is acceptable.

(continued)

BASES

LCO (continued)

Trip setpoints are those predetermined values of output at which an action should take place. The setpoints are compared to the actual process parameter (e.g., reactor vessel water level), and when the measured output value of the process parameter exceeds the setpoint, the associated device (e.g., trip unit) changes state. The analytic limits are derived from the limiting values of the process parameters obtained from the safety analysis. The Allowable Values are derived from the analytic limits, corrected for calibration, process, and some of the instrument errors. A channel is inoperable if its actual trip setpoint is not within its required Allowable Value. The trip setpoints are then determined accounting for the remaining instrument errors (e.g., drift). The trip setpoints derived in this manner provide adequate protection because instrumentation uncertainties, process effects, calibration tolerances, instrument drift, and severe environmental effects (for channels that must function in harsh environments as defined by 10 CFR 50.49) are accounted for.

APPLICABILITY

The feedwater and main turbine high water level trip instrumentation is required to be OPERABLE at ≥ 23% RTP to ensure that the fuel cladding integrity Safety Limit and the cladding 1% plastic strain limit are not violated during the feedwater controller failure, maximum demand event. As discussed in the Bases for LCO 3.2.1, "Average Planar Linear Heat Generation Rate (APLHGR)," LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)," and LCO 3.2.3, “LINEAR HEAT GENERATION RATE (LHGR),” sufficient margin to these limits exists below 23% RTP; therefore, these requirements are only necessary when operating at or above this power level.

(continued)

BASES

ACTIONS

B.1 (continued)

The 2 hour Completion Time is sufficient for the operator to take corrective action, and takes into account the likelihood of an event requiring actuation of feedwater and main turbine high water level trip instrumentation occurring during this period. It is also consistent with the 2 hour Completion Time provided in LCO 3.2.2 for Required Action A.1, since this instrumentation's purpose is to preclude a MCPR violation.

C.1

With the required channels not restored to OPERABLE status or placed in trip, THERMAL POWER must be reduced to < 23% RTP within 4 hours. As discussed in the Applicability section of the Bases, operation below 23% RTP results in sufficient margin to the required limits, and the feedwater and main turbine high water level trip instrumentation is not required to protect fuel integrity during the feedwater controller failure, maximum demand event. The allowed Completion Time of 4 hours is based on operating experience to reduce THERMAL POWER to < 23% RTP from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

The Surveillances are modified by a Note to indicate that when a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided the associated Function maintains feedwater and main turbine high water level trip capability. Upon completion of the Surveillance, or expiration of the 6 hour allowance, the channel must be returned to OPERABLE status

(continued)

BASES

BACKGROUND (continued)

Each EOC-RPT trip system is a two-out-of-two logic for each Function; thus, either two TSV - Closure or two TCV Fast Closure, Trip Oil Pressure - Low signals are required for a trip system to actuate. If either trip system actuates, both recirculation pumps will trip. There are two EOC-RPT breakers in series per recirculation pump. One trip system trips one of the two EOC-RPT breakers for each recirculation pump, and the second trip system trips the other EOC-RPT breaker for each recirculation pump.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

The TSV - Closure and the TCV Fast Closure, Trip Oil Pressure - Low Functions are designed to trip the recirculation pumps in the event of a turbine trip or generator load rejection to mitigate the increase in neutron flux, heat flux, and reactor pressure, and to increase the margin to the MCPR SL and LHGR limits. The analytical methods and assumptions used in evaluating the turbine trip and generator load rejection are summarized in References 2, 3, and 4.

To mitigate pressurization transient effects, the EOC-RPT must trip the recirculation pumps after initiation of closure movement of either the TSVs or the TCVs. The combined effects of this trip and a scram reduce fuel bundle power more rapidly than a scram alone, resulting in an increased margin to the MCPR SL and LHGR limits. Alternatively, MCPR limits for an inoperable EOC-RPT, as specified in the COLR, are sufficient to prevent violation of the MCPR Safety Limit and fuel mechanical limits. The EOC-RPT function is automatically disabled when turbine first stage pressure is < 26% RTP.

EOC-RPT instrumentation satisfies Criterion 3 of the NRC Policy Statement (Ref. 6).

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY	Turbine Stop Valve - Closure (continued)

Turbine Stop Valve - Closure (continued)

Closure of the TSVs is determined by measuring the position of each valve. There are two separate position signals associated with each stop valve, the signal from each switch being assigned to a separate trip channel. The logic for the TSV - Closure Function is such that two or more TSVs must be closed to produce an EOC-RPT. This Function must be enabled at THERMAL POWER ≥ 26% RTP. This is normally accomplished automatically by pressure transmitters sensing turbine first stage pressure; therefore, opening the turbine bypass valves may affect this function. To consider this function OPERABLE, bypass of the function must not occur when bypass valves are open. Four channels of TSV - Closure, with two channels in each trip system, are available and required to be OPERABLE to ensure that no single instrument failure will preclude an EOC-RPT from this Function on a valid signal. The TSV - Closure Allowable Value is selected to detect imminent TSV closure.

This protection is required, consistent with the safety analysis assumptions, whenever THERMAL POWER is ≥ 26% RTP. Below 26% RTP, the Reactor Vessel Steam Dome Pressure - High and the Average Power Range Monitor (APRM) Fixed Neutron Flux - High Functions of the Reactor Protection System (RPS) are adequate to maintain the necessary margin to the MCPR SL and LHGR limits.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)	Turbine Control Valve Fast Closure, Trip Oil Pressure - Low (PS-47-142, PS-47-144, PS-47-146, and PS-47-148)

Fast closure of the TCVs during a generator load rejection results in the loss of a heat sink that produces reactor pressure, neutron flux, and heat flux transients that must be limited. Therefore, an RPT is initiated on TCV Fast Closure, Trip Oil Pressure - Low in anticipation of the transients that would result from the closure of these valves. The EOC-RPT decreases reactor power and aids the reactor scram in ensuring that the MCPR SL and LHGR limits are not exceeded during the worst case transient.

Fast closure of the TCVs is determined by measuring the electrohydraulic control fluid pressure at each control valve. There is one pressure switch associated with each control valve, and the signal from each switch is assigned to a separate trip channel. The logic for the TCV Fast Closure, Trip Oil Pressure - Low Function is such that two or more TCVs must be closed (pressure switch trips) to produce an EOC-RPT. This Function must be enabled at THERMAL POWER ≥ 26% RTP. This is normally accomplished automatically by pressure transmitters sensing turbine first stage pressure; therefore, opening the turbine bypass valves may affect this function. To consider this function OPERABLE, bypass of the function must not occur when bypass valves are open. Four channels of TCV Fast Closure, Trip Oil Pressure - Low, with two channels in each trip system, are available and required to be OPERABLE to ensure that no single instrument failure will preclude an EOC-RPT from this Function on a valid signal. The TCV Fast Closure, Trip Oil Pressure - Low Allowable Value is selected high enough to detect imminent TCV fast closure.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY	Turbine Control Valve Fast Closure, Trip Oil Pressure - Low (PS-47-142, PS-47-144, PS-47-146, and PS-47-148) (continued) This protection is required consistent with the safety analysis whenever THERMAL POWER is ≥ 26% RTP. Below 26% RTP, the Reactor Vessel Steam Dome Pressure - High and the APRM Fixed Neutron Flux - High Functions of the RPS are adequate to maintain the necessary safety margins.
ACTIONS	A Note has been provided to modify the ACTIONS related to EOC-RPT instrumentation channels. Section 1.3, Completion Times, specifies that once a Condition has been entered, subsequent divisions, subsystems, components, or variables expressed in the Condition, discovered to be inoperable or not within limits, will not result in separate entry into the Condition. Section 1.3 also specifies that Required Actions of the Condition continue to apply for each additional failure, with Completion Times based on initial entry into the Condition. However, the Required Actions for inoperable EOC-RPT instrumentation channels provide appropriate compensatory measures for separate inoperable channels. As such, a Note has been provided that allows separate Condition entry for each inoperable EOC-RPT instrumentation channel.

(continued)

BASES

ACTIONS (continued)

B.1 and B.2

Required Actions B.1 and B.2 are intended to ensure that appropriate actions are taken if multiple, inoperable, untripped channels within the same Function result in the Function not maintaining EOC-RPT trip capability. A Function is considered to be maintaining EOC-RPT trip capability when sufficient channels are OPERABLE or in trip, such that the EOC-RPT System will generate a trip signal from the given Function on a valid signal and both recirculation pumps can be tripped. Alternately, Required Action B.2 requires the MCPR and LHGR limits for inoperable EOC-RPT, as specified in the COLR, to be applied. This also restores the margin to MCPR and LHGR limits assumed in the safety analysis.

The 2 hour Completion Time is sufficient time for the operator to take corrective action, and takes into account the likelihood of an event requiring actuation of the EOC-RPT instrumentation during this period. It is also consistent with the 2 hour Completion Time provided in LCO 3.2.2 for Required Action A.1, since this instrumentation's purpose is to preclude a MCPR or LHGR violation.

C.1

With any Required Action and associated Completion Time not met, THERMAL POWER must be reduced to < 26% RTP within 4 hours. The allowed Completion Time of 4 hours is reasonable, based on operating experience, to reduce THERMAL POWER to < 26% RTP from full power conditions in an orderly manner and without challenging plant systems.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.4.1.2

This SR ensures that an EOC-RPT initiated from the TSV - Closure and TCV Fast Closure, Trip Oil Pressure - Low Functions will not be inadvertently bypassed when THERMAL POWER is ≥ 26% RTP. This involves calibration of the bypass channels. Adequate margins for the instrument setpoint methodologies are incorporated into the actual setpoint. If any bypass channel's setpoint is nonconservative (i.e., the Functions are bypassed at ≥ 26% RTP, either due to open main turbine bypass valves or other reasons), the affected TSV - Closure and TCV Fast Closure, Trip Oil Pressure - Low Functions are considered inoperable. Alternatively, the bypass channel can be placed in the conservative condition (nonbypass). If placed in the nonbypass condition, this SR is met with the channel considered OPERABLE.

The Frequency of 24 months is based upon the assumption of a 24 month calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis.

SR 3.3.4.1.3

CHANNEL CALIBRATION is a complete check of the instrument loop and the sensor. This test verifies the channel responds to the measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drifts between successive calibrations consistent with the plant specific setpoint methodology. The Frequency is based upon the assumption of an 24 month calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis.

(continued)

BASES

SURVEILLANCE REQUIREMENTS	SR 3.4.2.1 (continued) Note 2 allows this SR not to be performed until 24 hours after THERMAL POWER exceeds 23% of RTP. During low flow conditions, jet pump noise approaches the threshold response of the associated flow instrumentation and precludes the collection of repeatable and meaningful data. The 24 hours is an acceptable time to establish conditions appropriate to perform this SR.
REFERENCES	1. FSAR, Section 14.6.3. 2. GE Service Information Letter No. 330, "Jet Pump Beam Cracks," June 9, 1980. 3. NUREG/CR-3052, "Closeout of IE Bulletin 80-07: BWR Jet Pump Assembly Failure," November 1984. 4. NRC No. 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.

BASES

APPLICABLE SAFETY ANALYSES (continued)	The maximum allowable leakage rate for the primary containment (L_a) is 2.0% by weight of the containment air per 24 hours at the design basis LOCA maximum peak containment pressure (P_a) of 49.1 psig (Ref. 1). Primary containment satisfies Criterion 3 of the NRC Policy Statement (Ref. 6).
LCO	Primary containment OPERABILITY is maintained by limiting leakage to ≤ 1.0 L_a, except prior to the first startup after performing a required Primary Containment Leakage Rate Testing Program leakage test. At this time, applicable leakage limits must be met. Compliance with this LCO will ensure a primary containment configuration, including equipment hatches, that is structurally sound and that will limit leakage to those leakage rates assumed in the safety analyses. Individual leakage rates specified for the primary containment air lock are addressed in LCO 3.6.1.2.
APPLICABILITY	In MODES 1, 2, and 3, a DBA could cause a release of radioactive material to primary containment. In MODES 4 and 5, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, primary containment is not required to be OPERABLE in MODES 4 and 5 to prevent leakage of radioactive material from primary containment.

(continued)

BASES

SURVEILLANCE REQUIREMENTS	SR 3.6.1.1.2 (continued) Satisfactory performance of this SR can be achieved by establishing a known differential pressure between the drywell and the suppression chamber and verifying that the pressure in either the suppression chamber or the drywell does not change by more than 0.25 inch of water per minute over a 10 minute period. The leakage test is performed every 24 months. The 24 month Frequency was developed considering it is prudent that this Surveillance be performed during a unit outage and also in view of the fact that component failures that might have affected this test are identified by other primary containment SRs.

REFERENCES

1. NEDC-33860P, “Safety Analysis Report for Browns Ferry Nuclear Plant Units 1, 2, and 3 Extended Power Uprate,” Section 2.6.
2. FSAR, Section 14.6.
3. 10 CFR 50, Appendix J, Option B.
4. NEI 94-01, Revision O, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J."
5. ANSI/ANS-56.8-1994, "American National Standard for Containment System Leakage Testing Requirement."
6. NRC No. 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.

BASES

BACKGROUND (continued)

the air lock to remain open for extended periods when frequent primary containment entry is necessary. Under some conditions allowed by this LCO, the primary containment may be accessed through the air lock, when the interlock mechanism has failed, by manually performing the interlock function.

The primary containment air lock forms part of the primary containment pressure boundary. As such, air lock integrity and leak tightness are essential for maintaining primary containment leakage rate to within limits in the event of a DBA. Not maintaining air lock integrity or leak tightness may result in a leakage rate in excess of that assumed in the unit safety analysis.

APPLICABLE SAFETY ANALYSES

The DBA that postulates the maximum release of radioactive material within primary containment is a LOCA. In the analysis of this accident, it is assumed that primary containment is OPERABLE, such that release of fission products to the environment is controlled by the rate of primary containment leakage. The primary containment is designed with a maximum allowable leakage rate (L_a) of 2.0% by weight of the containment air per 24 hours at the calculated maximum peak containment pressure (P_a) of 49.1 psig (Ref. 3). This allowable leakage rate forms the basis for the acceptance criteria imposed on the SRs associated with the air lock.

Primary containment air lock OPERABILITY is also required to minimize the amount of fission product gases that may escape primary containment through the air lock and contaminate and pressurize the secondary containment.

The primary containment air lock satisfies Criterion 3 of the NRC Policy Statement (Ref. 4).

(continued)

B 3.6 CONTAINMENT SYSTEMS

B 3.6.3.1 Containment Atmosphere Dilution (CAD) System

BASES

BACKGROUND

The CAD System functions to maintain combustible gas concentrations within the primary containment at or below the flammability limits following a postulated loss of coolant accident (LOCA) by diluting hydrogen and oxygen with nitrogen. To ensure that a combustible gas mixture does not occur, oxygen concentration is kept < 5.0 volume percent (v/o), or hydrogen concentration is kept < 4.0 v/o.

The CAD System is manually initiated and consists of two independent, 100% capacity subsystems, each of which is capable of supplying nitrogen through separate piping systems to the drywell and suppression chamber of each unit. Each subsystem includes a liquid nitrogen supply tank, ambient vaporizer, electric heater, and a manifold with branches to each primary containment (for Units 1, 2, and 3). The nitrogen storage tanks each contain ≥ 2615 gal, which is adequate for 7 days of CAD subsystem operation (Ref. 4).

The CAD System operates in conjunction with emergency operating procedures that are used to reduce primary containment pressure periodically during CAD System operation. This combination results in a feed and bleed approach to maintaining hydrogen and oxygen concentrations below combustible levels.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.6.3.1.1

Verifying that there is ≥ 2615 gal of liquid nitrogen supply in each nitrogen storage tank will ensure at least 7 days of post-LOCA CAD operation. This minimum volume of liquid nitrogen represents the analytical limit assumed in the analysis of the primary containment atmosphere following a postulated LOCA and does not include allowance for potential nitrogen boiloff and tank level instrumentation inaccuracies. This minimum volume of liquid nitrogen allows sufficient time after an accident to replenish the nitrogen supply for long term inerting. This is verified every 31 days to ensure that the system is capable of performing its intended function when required. The 31 day Frequency is based on operating experience, which has shown 31 days to be an acceptable period to verify the liquid nitrogen supply and on the availability of other hydrogen mitigating systems.

SR 3.6.3.1.2

Verifying the correct alignment for manual, power operated, and automatic valves in each of the CAD subsystem flow paths provides assurance that the proper flow paths exist for system operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since these valves were verified to be in the correct position prior to locking, sealing, or securing.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.3.1.2 (continued)

A valve is also allowed to be in the nonaccident position provided it can be aligned to the accident position within the time assumed in the accident analysis. This is acceptable because the CAD System is manually initiated. This SR does not apply to valves that cannot be inadvertently misaligned, such as check valves. This SR does not require any testing or valve manipulation; rather, it involves verification that those valves capable of being mispositioned are in the correct position.

The 31 day Frequency is appropriate because the valves are operated under procedural control, improper valve position would only affect a single subsystem, the probability of an event requiring initiation of the system is low, and the system is a manually initiated system.

REFERENCES

1. AEC Safety Guide 7, Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident, March 10, 1971.

2. FSAR, Section 5.2.6.

3. NRC No. 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.

4. ANP-3403P, “Fuel Uprate Safety Analysis Report for Browns Ferry Nuclear Plant Units 1, 2, and 3,” Section 2.6.4, August 2015.

B 3.7 PLANT SYSTEMS

B 3.7.1 Residual Heat Removal Service Water (RHRSW) System

BASES

BACKGROUND

The RHRSW System is designed to provide cooling water for the Residual Heat Removal (RHR) System heat exchangers, required for a safe reactor shutdown following a Design Basis Accident (DBA) or transient. The RHRSW System is operated whenever the RHR heat exchangers are required to operate in the shutdown cooling mode or in the suppression pool cooling or spray mode of the RHR System.

The RHRSW System is common to the three BFN units and consists of four independent and redundant subsystems, each of which feeds one RHR heat exchanger in each unit. Each subsystem is made up of a header, two 4500 gpm pumps, a suction source, valves, piping, and associated instrumentation. Two subsystems, with one pump operating in each subsystem, are capable of providing 100% of the required cooling capacity to maintain safe shutdown conditions for one unit. The RHRSW System is designed with sufficient redundancy so that no single active component failure can prevent it from achieving its design function. The RHRSW System is described in the FSAR, Section 10.9 (Ref. 1).

Cooling water is pumped by the RHRSW pumps from the Wheeler Reservoir through the tube side of the RHR heat exchangers, and discharged back to the Wheeler Reservoir.

(continued)

BASES

BACKGROUND (continued)

The system is initiated manually from each of the three units control rooms. If operating during a loss of coolant accident (LOCA), the system is automatically tripped on degraded bus voltage to allow the diesel generators to automatically power only that equipment necessary to reflood the core. The system can be manually started any time the degraded bus voltage signal clears, and is assumed to be manually started within 10 minutes after the LOCA.

APPLICABLE SAFETY ANALYSES

The RHRSW System removes heat from the suppression pool to limit the suppression pool temperature and primary containment pressure following a LOCA. This ensures that the primary containment can perform its function of limiting the release of radioactive materials to the environment following a LOCA. The ability of the RHRSW System to support long term cooling of the reactor or primary containment is discussed in the FSAR, Chapters 5 and 14 (Refs. 2 and 3, respectively). These analyses explicitly assume that the RHRSW System will provide adequate cooling support to the equipment required for safe shutdown. These analyses include the evaluation of the long term primary containment response after a design basis LOCA.

The safety analyses for long term cooling were performed for various combinations of RHR System failures and considers the number of units fueled. With one unit fueled, the worst case single failure that would affect the performance of the RHRSW System is any failure that would disable two subsystems of the RHRSW System.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

With two and three units fueled, a worst case single failure could also include the loss of two RHRSW pumps caused by losing a 4 kV shutdown board since there are certain alignment configurations that allow two RHRSW pumps to be powered from the same 4 kV shutdown board. As discussed in the FSAR, Section 14.6.3.3.2 (Ref. 4) for these analyses, manual initiation of the OPERABLE RHRSW subsystems and the associated RHR System is assumed to occur 10 minutes after a DBA. The analyses assume that there are two RHRSW subsystems operating in each unit, with one RHRSW pump in each subsystem capable of producing 4000 gpm of flow. In this case, the maximum suppression chamber water temperature and pressure are 187.3°F and 49.1 psig, respectively, well below the design temperature of 281°F and maximum allowable pressure of 62 psig.

The RHRSW System satisfies Criterion 3 of the NRC Policy Statement (Ref 5).

LCO

Four RHRSW subsystems are required to be OPERABLE to provide the required redundancy to ensure that the system functions to remove post accident heat loads, assuming the worst case single active failure occurs coincident with the loss of offsite power. Additionally, since the RHRSW pumps are shared between the three BFN units, the number of OPERABLE pumps required is also dependent on the number of units fueled.

An RHRSW subsystem is considered OPERABLE when:

a. At least one RHRSW pump (i.e., one required RHRSW pump) is OPERABLE, and

b. An OPERABLE flow path is capable of taking suction from the intake structure and transferring the water to the associated RHR heat exchanger at the assumed flow rate.

(continued)

BASES

LCO (continued)

In addition to the required number of OPERABLE subsystems, there must be an adequate number of pumps OPERABLE to provide cooling for the fueled non-accident units.

The total number of required RHRSW pumps must take into consideration the required number of pumps required for the specific unit along with the number of pumps required for other units that are fueled. Hence, when one unit contains fuel, four RHRSW pumps are required to be OPERABLE. When two units contain fuel, six RHRSW pumps are required to be OPERABLE. When three units contain fuel, eight RHRSW pumps are required to be OPERABLE. The minimum specified number of pumps gives consideration to all units capable of producing heat in aggregate and accounts for a single active failure.

The above pre-accident configuration ensures that during a design bases accident with a postulated single active failure, the resulting configuration for the accident unit has at least two RHRSW subsystems OPERABLE to supply 100 percent of the long term RHR cooling water. The resulting configuration for the non-accident units has at least two RHRSW subsystems per unit OPERABLE to supply 100 percent of the required cooling capacity to maintain safe shutdown conditions.

(continued)

BASES

LCO (continued)

The number of required OPERABLE RHRSW pumps is modified by a Note which specifies that the number of required RHRSW pumps may be reduced by one for each fueled unit that has been in MODE 4 or 5 for over 24 hours. This Note acknowledges the fact that decay heat removal requirements are substantially reduced for fueled units in MODE 4 or 5 for over 24 hours.

APPLICABILITY

In MODES 1, 2, and 3, the RHRSW System is required to be OPERABLE to support the OPERABILITY of the RHR System for primary containment cooling (LCO 3.6.2.3, "Residual Heat Removal (RHR) Suppression Pool Cooling," and LCO 3.6.2.4, "Residual Heat Removal (RHR) Suppression Pool Spray") and decay heat removal (LCO 3.4.7, "Residual Heat Removal (RHR) Shutdown Cooling System - Hot Shutdown"). The Applicability is therefore consistent with the requirements of these systems.

In MODES 4 and 5, the OPERABILITY requirements of the RHRSW System are determined by the systems it supports.

(continued)

BASES (continued)

ACTIONS

Since the RHRSW System is common to all three units, the following requirements must be followed when multiple units contain fuel:

a. With one or more required RHRSW pumps inoperable, all applicable ACTIONS must be entered for each unit.

b. With one or more RHRSW subsystem inoperable, all applicable ACTIONS for inoperable subsystems must be entered on the unit(s) that have the inoperable subsystem.

The Required Actions and associated Completion Times of Conditions A, B, C, and D are based on a reduction in redundancy of the RHRSW System, not a loss of RHRSW safety function. The Required Actions and associated Completion Times of Conditions E, F, and G consider that the RHRSW safety function is lost.

RHRSW safety function is maintained when at least two RHRSW subsystems, with two separate RHRSW pumps (i.e., one per subsystem), on a per fueled unit basis, are OPERABLE. Additionally, the total number of RHRSW pumps must be such that the RHRSW pumps credited for maintaining the RHRSW safety function for a specific unit are not credited for maintaining the RHRSW safety function for a different fueled unit.

When there are three fueled units, the RHRSW safety function is maintained when:

- Two RHRSW subsystems per fueled unit are OPERABLE;

(continued)

BASES (continued)

ACTIONS (continued)

- Six RHRSW pumps are OPERABLE (two RHRSW pumps per fueled unit);
- Each OPERABLE RHRSW subsystem has one required RHRSW pump OPERABLE; and
- The required RHRSW pump that is OPERABLE in an RHRSW subsystem is not credited for maintaining the RHRSW safety function for another fueled unit (i.e., it is not one of the two RHRSW pumps that is required to be OPERABLE for another fueled unit).

When there are two fueled units, the RHRSW safety function is maintained when:

- Two RHRSW subsystems per fueled unit are OPERABLE;
- Four RHRSW pumps are OPERABLE (two RHRSW pumps per fueled unit);
- Each OPERABLE RHRSW subsystem has one required RHRSW pump OPERABLE; and
- The required RHRSW pump that is OPERABLE in an RHRSW subsystem is not credited for maintaining the RHRSW safety function for another fueled unit (i.e., it is not one of the two RHRSW pumps that is required to be OPERABLE for another fueled unit).

When there is one fueled unit, the RHRSW safety function is maintained when:

- Two RHRSW subsystems are OPERABLE and
- Two RHRSW pumps are OPERABLE (i.e., one per OPERABLE RHRSW subsystem).

(continued)

BASES (continued)

ACTIONS (continued)

When any combination of pump(s) and other subsystem components, e.g., heat exchanger(s), are inoperable such that three or more components of the RHRSW System (on any or all fueled units) are inoperable, the capability to meet the safety function must be evaluated by all fueled units. When an RHRSW pump is credited by one fueled unit for maintaining the RHRSW safety function, then the other two fueled units cannot also credit this same RHRSW pump with maintaining their RHRSW safety function since the capacity of a single RHRSW pump is not sufficient to support the required heat removal function of more than one RHR heat exchanger. Therefore, in this condition, the RHRSW pump credited with maintaining RHRSW safety function on a fueled unit must be considered inoperable for the other fueled units for purpose of determining if RHRSW safety function is maintained. The other fueled units must then include the additional inoperable RHRSW pump(s) with the total number of inoperable components when determining if RHRSW safety function is maintained. If RHRSW safety function is determined to be lost, then Condition E or F is required to be entered.

The examples, with respect to RHRSW pumps, used in the following descriptions of the ACTIONS assume that all three units are fueled.

A.1 and A.2

With one required RHRSW pump inoperable, the inoperable RHRSW pump must be restored to OPERABLE status within 30 days. With the unit in this condition, the remaining OPERABLE RHRSW pumps are adequate to perform.

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

the RHRSW heat removal function. However, the overall reliability is reduced because a single failure could result in reduced primary containment cooling capability. The 30 day Completion Time is based on the availability of equipment in excess of normal redundancy requirements and the low probability of an event occurring requiring RHRSW during this period.

Alternatively, five RHRSW pumps may be verified to be OPERABLE with power being supplied from separate 4 kV shutdown boards.

Required Action A1 is modified by two Notes. Note 1 indicates that the Required Action is applicable only when two units are fueled. In the two unit fueled condition, a single failure (loss of a 4 kV shutdown board) could result in inadequate RHRSW pumps if two pumps are powered from the same power supply. If five RHRSW pumps are powered from separate 4 kV shutdown boards, then no postulated single active failure could occur to prevent the RHRSW system from performing its design function. Operation can continue indefinitely if Required Action A.1 is met.

Note 2 requires only four RHRSW pumps powered from separate 4 kV shutdown boards to be OPERABLE if the other fueled unit has been in MODE 4 or 5 greater than 24 hours. This acknowledges the fact that decay heat removal requirements are substantially reduced for fueled units in MODE 4 or 5 for greater than 24 hours.

These two Notes clarify the situations under which Required Action A.1 would be the appropriate Required Action.

(continued)

BASES

ACTIONS (continued)

B.1

With one RHRSW subsystem inoperable (e.g., one RHR heat exchanger inoperable or an RHRSW header isolated) for reasons other than inoperable RHRSW pumps, which are covered by separate Conditions, the inoperable RHRSW subsystem must be restored to OPERABLE status within 30 days. With the unit in this condition, the remaining OPERABLE RHRSW subsystems are adequate to perform the RHRSW heat removal function. However, the overall reliability is reduced because a single failure could result in reduced primary containment cooling capability. The 30 day Completion Time is based on the availability of equipment in excess of normal redundancy requirements and the low probability of an event occurring requiring RHRSW during this period.

The Required Action is modified by a Note indicating that the applicable Conditions of LCO 3.4.7 be entered and Required Actions taken if the inoperable RHRSW subsystem results in inoperable RHR shutdown cooling. This is an exception to LCO 3.0.6 and ensures the proper actions are taken for these components.

(continued)

BASES

ACTIONS (continued)

C.1

With two required RHRSW pumps inoperable (i.e., one required RHRSW pump inoperable in each of the two separate RHRSW subsystems or two RHRSW pumps inoperable in the same RHRSW subsystem), the remaining RHRSW pumps are adequate to perform the RHRSW heat removal function. However, the overall reliability is reduced because a single failure of the OPERABLE RHRSW pumps could result in a loss of RHRSW function. The seven day Completion Time is based on the redundant RHRSW capabilities afforded by the OPERABLE RHRSW pumps and the low probability of an event occurring during this period.

D.1

With two RHRSW subsystems inoperable (e.g., two RHR heat exchangers inoperable) for reasons other than inoperable RHRSW pumps, which are covered by separate Conditions, the remaining OPERABLE RHRSW subsystems are adequate to perform the RHRSW heat removal function. However, the overall reliability is reduced because a single failure could result in reduced primary containment cooling capability. The seven day Completion Time is based on the availability of equipment in excess of normal redundancy requirements and the low probability of an event occurring requiring RHRSW during this period.

The Required Action is modified by a Note indicating that the applicable Conditions of LCO 3.4.7 be entered and Required Actions taken if the inoperable RHRSW subsystem results in inoperable RHR shutdown cooling. This is an exception to LCO 3.0.6 and ensures the proper actions are taken for these components.

(continued)

BASES

ACTIONS (continued)

E.1

With three or more required RHRSW pumps inoperable, the RHRSW System is not capable of performing its intended function. The requisite number of pumps must be restored to OPERABLE status within eight hours. The eight hour Completion Time is based on the Completion Times provided for the RHR suppression pool cooling and spray functions.

F.1

With three or more required RHRSW subsystems inoperable (e.g., one RHR heat exchanger inoperable in each of three of four separate RHRSW subsystems) for reasons other than inoperable RHRSW pumps, which are covered by separate Conditions, the RHRSW System is not capable of performing its intended function. The requisite number of subsystems must be restored to OPERABLE status within eight hours. The eight hour Completion Time is based on the Completion Times provided for the RHR suppression pool cooling and spray functions.

The Required Action is modified by a Note indicating that the applicable Conditions of LCO 3.4.7 be entered and Required Actions taken if the inoperable RHRSW subsystem results in inoperable RHR shutdown cooling. This is an exception to LCO 3.0.6 and ensures the proper actions are taken for these components.

(continued)

BASES

ACTIONS (continued)

G.1 and G.2

If the RHRSW subsystem(s) or the RHRSW pump(s) cannot be restored to OPERABLE status within the associated Completion Times, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 12 hours and in MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.1.1

Verifying the correct alignment for each manual and power operated valve in each RHRSW subsystem flow path provides assurance that the proper flow paths will exist for RHRSW operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since these valves are verified to be in the correct position prior to locking, sealing, or securing. A valve is also allowed to be in the nonaccident position, and yet considered in the correct position, provided it can be realigned to its accident position. This is acceptable because the RHRSW System is a manually initiated system.

This SR does not require any testing or valve manipulation; rather, it involves verification that those valves capable of being mispositioned are in the correct position. This SR does not apply to valves that cannot be inadvertently misaligned, such as check valves.

The 31 day Frequency is based on engineering judgment, is consistent with the procedural controls governing valve operation, and ensures correct valve positions.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

REFERENCES

1. FSAR, Section 10.9.
2. FSAR, Chapter 5.
3. FSAR, Chapter 14.
4. FSAR, Section 14.6.3.3.2.
5. NRC No. 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.

BASES

LCO (continued)

The EECW System is considered OPERABLE when it has an OPERABLE UHS, three OPERABLE pumps, and two OPERABLE flow paths capable of taking suction from the intake structure and transferring the water to the appropriate equipment.

The OPERABILITY of the UHS for EECW is based on having a maximum water temperature of 95°F.

The isolation of the EECW System to components or systems may render those components or systems inoperable, but does not affect the OPERABILITY of the EECW System.

APPLICABILITY

In MODES 1, 2, and 3, the EECW System and UHS are required to be OPERABLE to support OPERABILITY of the equipment serviced by the EECW System. Therefore, the EECW System and UHS are required to be OPERABLE in these MODES.

In MODES 4 and 5, the OPERABILITY requirements of the EECW System and UHS are determined by the systems they support.

ACTIONS

A.1

With one required EECW pump inoperable, the required EECW pump must be restored to OPERABLE status within 7 days. With the system in this condition, the remaining OPERABLE EECW pumps are adequate to perform the heat removal function. However, the overall reliability is reduced because a single failure in the EECW System could result in loss of EECW function.

(continued)

BASES

ACTIONS

A.1 (continued)

The 7 day Completion Time is based on the redundant EECW System capabilities afforded by the remaining OPERABLE pumps, the low probability of an accident occurring during this time period and is consistent with the allowed Completion Time for restoring an inoperable DG.

B.1 and B.2

If the required EECW pump cannot be restored to OPERABLE status within the associated Completion Time, or two or more EECW pumps are inoperable or the UHS is determined inoperable, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 12 hours and in MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.2.1

Verification of the UHS temperature ensures that the heat removal capability of the EECW System is within the assumptions of the DBA analysis. The 24 hour Frequency is based on operating experience related to trending of the parameter variations during the applicable MODES.

(continued)

B 3.7 PLANT SYSTEMS

B 3.7.5 Main Turbine Bypass System

BASES

BACKGROUND

The Main Turbine Bypass System is designed to control steam pressure when reactor steam generation exceeds turbine requirements during unit startup, sudden load reduction, and cooldown. It allows excess steam flow from the reactor to the condenser without going through the turbine. The bypass capacity of the system is 23% of the Nuclear Steam Supply System rated steam flow. Sudden load reductions within the capacity of the steam bypass can be accommodated without reactor scram. The Main Turbine Bypass System consists of nine valves connected to the main steam lines between the main steam isolation valves and the turbine stop valve bypass valve chest. Each of these nine valves is operated by hydraulic cylinders. The bypass valves are controlled by the pressure regulation function of the Pressure Regulator and Turbine Generator Control System, as discussed in the FSAR, Section 7.11.3.3 (Ref. 1). The bypass valves are normally closed, and the pressure regulator controls the turbine control valves that direct all steam flow to the turbine. If the speed governor or the load limiter restricts steam flow to the turbine, the pressure regulator controls the system pressure by opening the bypass valves. When the bypass valves open, the steam flows from the bypass chest, through connecting piping, to the pressure breakdown assemblies, where a series of orifices are used to further reduce the steam pressure before the steam enters the condenser.

(continued)

BASES (continued)

APPLICABILITY

The Main Turbine Bypass System is required to be OPERABLE at ≥ 23% RTP to ensure that the fuel cladding integrity Safety Limit is not violated during abnormal operational transients. As discussed in the Bases for LCO 3.2.1 and LCO 3.2.2, sufficient margin to these limits exists at < 23% RTP. Therefore, these requirements are only necessary when operating at or above this power level.

ACTIONS

A.1

If the Main Turbine Bypass System is inoperable (one or more bypass valves inoperable), or the APLHGR, MCPR, and LHGR limits for an inoperable Main Turbine Bypass System, as specified in the COLR, are not applied, the assumptions of the design basis transient analysis may not be met. Under such circumstances, prompt action should be taken to restore the Main Turbine Bypass System to OPERABLE status or adjust the APLHGR, MCPR, and LHGR limits accordingly. The 2 hour Completion Time is reasonable, based on the time to complete the Required Action and the low probability of an event occurring during this period requiring the Main Turbine Bypass System.

B.1

If the Main Turbine Bypass System cannot be restored to OPERABLE status or the APLHGR, MCPR, and LHGR limits for an inoperable Main Turbine Bypass System are not applied, THERMAL POWER must be reduced to < 23% RTP. As discussed in the Applicability section, operation at < 23% RTP results in sufficient margin to the required limits, and the Main

(continued)