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TSB

APPROVED AMENDMENT TO THE
UNIT 1 TECHNICAL SPECIFICATIONS BASES MANUAL
REVISION 45

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SUSQUEHANNA STEAM ELECTRIC STATION
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BASES

BACKGROUND (continued)

Operation above the boundary of the nucleate boiling regime could result in excessive cladding temperature because of the onset of transition boiling and the resultant sharp reduction in heat transfer coefficient. Inside the steam film, high cladding temperatures are reached, and a cladding water (zirconium water) reaction may take place. This chemical reaction results in oxidation of the fuel cladding to a structurally weaker form. This weaker form may lose its integrity, resulting in an uncontrolled release of activity to the reactor coolant.

APPLICABLE SAFETY ANALYSES

The fuel cladding must not sustain damage as a result of normal operation and AOOs. The reactor core SLs are established to preclude violation of the fuel design criterion that an MCPR limit is to be established, such that at least 99.9% of the fuel rods in the core would not be expected to experience the onset of transition boiling.

The Reactor Protection System setpoints (LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation"), in combination with the other LCOs, are designed to prevent any anticipated combination of transient conditions for Reactor Coolant System water level, pressure, and THERMAL POWER level that would result in reaching the MCPR limit.

<u>2.1.1.1</u> <u>Fuel Cladding Integrity</u>

The use of the ANFB-10 (Reference 4) correlation is valid for critical power calculations at pressures > 571 psia and bundle mass fluxes > 0.115×10^6 lb/hr-ft^2. For operation at low pressures or low flows, the fuel cladding integrity SL is established by a limiting condition on core THERMAL POWER, with the following basis:

> Provided that the water level in the vessel downcomer is maintained above the top of the active fuel, natural circulation is sufficient to ensure a minimum bundle flow for all fuel assemblies that have a relatively high power and potentially can approach a critical heat flux condition.

(continued)

APPLICABLE SAFETY ANALYSES

2.1.1.1 Fuel Cladding Integrity (continued)

For the SPC ATRIUM-10 design, the minimum bundle flow is > 28 x 10^3 lb/hr. For the ATRIUM-10 fuel design, the coolant minimum bundle flow and maximum area are such that the mass flux is always > .25 x 10^6 lb/hr-ft^2. Full scale critical power test data taken from various SPC and GE fuel designs at pressures from 14.7 psia to 1400 psia indicate the fuel assembly critical power at 0.25 x 10^6 lb/hr-ft^2 is approximately 3.35 MWt. At 25% RTP, a bundle power of approximately 3.35 MWt corresponds to a bundle radial peaking factor of approximately 3.0, which is significantly higher than the expected peaking factor. Thus, a THERMAL POWER limit of 25% RTP for reactor pressures < 785 psig is conservative.

2.1.1.2 MCPR

The MCPR SL ensures sufficient conservatism in the operating MCPR limit that, in the event of an AOO from the limiting condition of operation, at least 99.9% of the fuel rods in the core would be expected to avoid boiling transition. The margin between calculated boiling transition (i.e., MCPR = 1.00) and the MCPR SL is based on a detailed statistical procedure that considers the uncertainties in monitoring the core operating state. One specific uncertainty included in the SL is the uncertainty in the critical power correlation. References 2 and 4 describe the methodology used in determining the MCPR SL.

The ANFB-10 critical power correlation is based on a significant body of practical test data. As long as the core pressure and flow are within the range of validity of the correlations (refer to Section B.2.1.1.1), the assumed reactor conditions used in defining the SL introduce conservatism into the limit because bounding high radial power factors and bounding flat local peaking distributions are used to estimate the number of rods in boiling transition. These conservatisms and the inherent accuracy of the ANFB-10 correlation provide a reasonable degree of assurance that during sustained operation at the MCPR SL there would be no transition boiling in the core.

(continued)

APPLICABLE SAFETY ANALYSES

2.1.1.2 MCPR (continued)

If boiling transition were to occur, there is reason to believe that the integrity of the fuel would not be compromised.

Significant test data accumulated by the NRC and private organizations indicate that the use of a boiling transition limitation to protect against cladding failure is a very conservative approach. Much of the data indicate that BWR fuel can survive for an extended period of time in an environment of boiling transition.

SPC Atrium -10 fuel is monitored using the ANFB-10 Critical Power Correlation. The effects of channel bow on MCPR are explicitly included in the calculation of the MCPR SL. Explicit treatment of channel bow in the MCPR SL addresses the concerns of NRC Bulletin No. 90-02 entitled "Loss of Thermal Margin Caused by Channel Box Bow."

Monitoring required for compliance with the MCPR SL is specified in LCO 3.2.2, Minimum Critical Power Ratio.

2.1.1.3 Reactor Vessel Water Level

During MODES 1 and 2 the reactor vessel water level is required to be above the top of the active fuel to provide core cooling capability. With fuel in the reactor vessel during periods when the reactor is shut down, consideration must be given to water level requirements due to the effect of decay heat. If the water level should drop below the top of the active irradiated fuel during this period, the ability to remove decay heat is reduced. This reduction in cooling capability could lead to elevated cladding temperatures and clad perforation in the event that the water level becomes < 2/3 of the core height. The reactor vessel water level SL has been established at the top of the active irradiated fuel to provide a point that can be

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.1.1 (continued)

the highest worth control rod is determined by analysis or testing.

Local critical tests require the withdrawal of control rods in a sequence that is not in conformance with BPWS. This testing would therefore require re-programming or bypassing of the rod worth minimizer to allow the withdrawal of control rods not in conformance with BPWS, and therefore additional requirements must be met (see LCO 3.10.7, "Control Rod Testing - Operating").

The Frequency of 4 hours after reaching criticality is allowed to provide a reasonable amount of time to perform the required calculations and have appropriate verification.

During MODE 5, adequate SDM is required to ensure that the reactor does not reach criticality during control rod withdrawals. An evaluation of each planned in-vessel fuel movement during fuel loading (including shuffling fuel within the core) is required to ensure adequate SDM is maintained during refueling. This evaluation ensures that the intermediate loading patterns are bounded by the safety analyses for the final core loading pattern. For example, bounding analyses that demonstrate adequate SDM for the most reactive configurations during the refueling may be performed to demonstrate acceptability of the entire fuel movement sequence. These bounding analyses include additional margins to the associated uncertainties. Spiral offload/reload sequences inherently satisfy the SR, provided the fuel assemblies are reloaded in the same configuration analyzed for the new cycle. Removing fuel from the core will always result in an increase in SDM.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 26.
2. FSAR, Section 15.
3. PL-NF-90-001-A, "Application of Reactor Analysis Methods for BWR Design and Analysis," Sections 2.2 and 2.8, July 1992, Supplement 1-A, August 1995, Supplement 2-A, July 1996 and Supplement 3-A, March 2001.
4. FSAR, Section 15.4.1.1.

BASES

REFERENCES (continued)

5. Final Policy Statement on Technical Specifications Improvements, July 22, 1993 (58 FR 39132).

6. FSAR, Section 4.3.

7. PL-NF-90-001-A, "Application of Reactor Analysis Methods for BWR Design and Analysis," Section 2.4, July 1992, Supplement 1-A, August 1995, Supplement 2-A, July 1996, and Supplement 3-A, March 2001.

BASES

REFERENCES (continued)

4. FSAR, Section 15.0.

5. PL-NF-90-001-A, Applicability of Reactor Analysis Methods for BWR Design and Analysis," Section 4.1.2, July 1992, and Supplement 1-A, August 1995, Supplement 2-A, July 1996, and Supplement 3-A, March 2001.

6. Final Policy Statement on Technical Specifications Improvements, July 22, 1993 (58 FR 39132).

7. Letter from R.F. Janecek (BWROG) to R.W. Starostecki (NRC), "BWR Owners Group Revised Reactivity Control System Technical Specifications," BWROG-8754, September 17, 1987.

BASES

ACTIONS (continued)

B.1 and B.2

If nine or more OPERABLE control rods are out of sequence, the control rod pattern significantly deviates from the prescribed sequence. Control rod withdrawal should be suspended immediately to prevent the potential for further deviation from the prescribed sequence. Control rod insertion to correct control rods withdrawn beyond their allowed position is allowed since, in general, insertion of control rods has less impact on control rod worth than withdrawals have. Required Action B.1 is modified by a Note which allows the RWM to be bypassed to allow the affected control rods to be returned to their correct position. LCO 3.3.2.1 requires verification of control rod movement by a qualified member of the technical staff.

When nine or more OPERABLE control rods are not in compliance with BPWS, the reactor mode switch must be placed in the shutdown position within 1 hour. With the mode switch in shutdown, the reactor is shut down, and as such, does not meet the applicability requirements of this LCO. The allowed Completion Time of 1 hour is reasonable to allow insertion of control rods to restore compliance, and is appropriate relative to the low probability of a CRDA occurring with the control rods out of sequence.

SURVEILLANCE REQUIREMENTS

SR 3.1.6.1

The control rod pattern is verified to be in compliance with the BPWS at a 24 hour Frequency to ensure the assumptions of the CRDA analyses are met. The 24 hour Frequency was developed considering that the primary check on compliance with the BPWS is performed by the RWM (LCO 3.3.2.1), which provides control rod blocks to enforce the required sequence and is required to be OPERABLE when operating at ≤ 10% RTP.

REFERENCES

1. "PL-NF-90-001-A, "Application of Reactor Analysis Methods for BWR Design and Analysis," Section 2.8, July 1992, Supplement 1-A, August 1995, Supplement 2-A, July 1996, and Supplement 3-A, March 2001.

(continued)

B. 3.2 POWER DISTRIBUTION LIMITS

B 3.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)

BASES

BACKGROUND

The APLHGR is a measure of the average LHGR of all the fuel rods in a fuel assembly at any axial location. Limits on the APLHGR are specified to ensure that limits specified in 10 CFR 50.46 are not exceeded during the postulated design basis loss of coolant accident (LOCA).

APPLICABLE SAFETY ANALYSES

SPC performed LOCA calculations for the SPC ATRIUM™-10 fuel design. The analytical methods and assumptions used in evaluating the fuel design limits from 10 CFR 50.46 are presented in References 3, 4, 5, and 6 for the SPC analysis. The analytical methods and assumptions used in evaluating Design Basis Accidents (DBAs) that determine the APLHGR Limits are presented in References 3 through 9.

LOCA analyses are performed to ensure that the APLHGR limits are adequate to meet the Peak Cladding Temperature (PCT), maximum cladding oxidation, and maximum hydrogen generation limits of 10 CFR 50.46. The analyses are performed using calculational models that are consistent with the requirements of 10 CFR 50, Appendix K. A complete discussion of the analysis codes are provided in References 3, 4, 5, and 6 for the SPC analysis. The PCT following a postulated LOCA is a function of the average heat generation rate of all the rods of a fuel assembly at any axial location and is not strongly influenced by the rod to rod power distribution within the assembly.

APLHGR limits are developed as a function of fuel type and exposure. The SPC LOCA analyses also consider several alternate operating modes in the development of the APLHGR limits (e.g., Extended Load Line Limit Analysis (ELLA), Suppression Pool Cooling Mode, and Single Loop Operation (SLO)). LOCA analyses were performed for the regions of the power/flow map bounded by the 100% rod line and the APRM rod block line (i.e., the ELLA region). The ELLA region is analyzed to determine whether an APLHGR multiplier as a function of core flow is required. The results of the analysis demonstrate the PCTs are within the 10 CFR 50.46 limit, and that APLHGR multipliers as a function of core flow are not required.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The SPC LOCA analyses consider the delay in Low Pressure Coolant Injection (LPCI) availability when the unit is operating in the Suppression Pool Cooling Mode. The delay in LPCI availability is due to the time required to realign valves from the Suppression Pool Cooling Mode to the LPCI mode. The results of the analyses demonstrate that the PCTs are within the 10 CFR 50.46 limit.

Finally, the SPC LOCA analyses were performed for Single-Loop Operation. The results of the SPC analysis for ATRIUM™-10 fuel shows that an APLHGR limit which is 0.8 times the two-loop APLHGR limit meets the 10 CFR 50.46 acceptance criteria, and that the PCT is less than the limiting two-loop PCT.

The APLHGR satisfies Criterion 2 of the NRC Policy Statement (Ref. 10).

LCO

The APLHGR limits specified in the COLR are the result of the DBA analyses.

APPLICABILITY

The APLHGR limits are primarily derived from LOCA analyses that are 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. At THERMAL POWER levels < 25% RTP, the reactor is 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 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. 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 DBA occurring simultaneously with the APLHGR out of specification.

(continued)

BASES

ACTIONS (continued)

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 < 25% RTP within 4 hours. The allowed Completion Time is reasonable, based on operating experience, to reduce THERMAL POWER to < 25% 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 24 hours after THERMAL POWER is ≥ 25% RTP and then every 24 hours thereafter. Additionally, APLHGRs must be calculated prior to exceeding 50% RTP unless performed in the previous 24 hours. APLHGRs 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 24 hour allowance after THERMAL POWER ≥ 25% RTP is achieved is acceptable given the large inherent margin to operating limits at low power levels and because the APLHGRs must be calculated prior to exceeding 50% RTP.

REFERENCES

1. Not used.

2. Not used.

3. ANF-91-048(P)(A), "Advanced Nuclear Fuels Corporation Methodology for Boiling Water Reactors EXEM BWR Evaluation Model," January 1993.

4. ANF-CC-33(P)(A) Supplement 2, "HUXY: A Generalized Multirod Heatup Code with 10CFR50 Appendix K Heatup Option," January 1991.

5. XN-CC-33(P)(A) Revision 1, "HUXY: A Generalized Multirod Heatup Code with 10CFR50 Appendix K Heatup Option Users Manual," November 1975.

(continued)

BASES

References (continued)

6. XN-NF-80-19(P)(A), Volumes 2, 2A, 2B, and 2C "Exxon Nuclear Methodology for Boiling Water Reactors: EXEM BWR ECCS Evaluation Model," September 1982.
7. FSAR, Chapter 4.
8. FSAR, Chapter 6.
9. FSAR, Chapter 15.
10. Final Policy Statement on Technical Specifications Improvements, July 22, 1993 (58 FR 39132).

BASES

SURVEILLANCE REQUIREMENTS

SR 3.2.2.1 (continued)

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 24 hour allowance after THERMAL POWER ≥ 25% RTP is achieved is acceptable given the large inherent margin to operating limits at low power levels and because the MCPR must be calculated prior to exceeding 50% RTP.

SR 3.2.2.2

Because the transient analysis takes credit for conservatism in the scram time performance, it must be demonstrated that the specific scram time is consistent with those used in the transient analysis. SR 3.2.2.2 determines the scram time fraction which is a measure of the actual scram time compared with the assumed scram time. The COLR contains a table of scram time fractions based on the LCO 3.1.4 "Control Rod Scram Times" and the realistic scram times used in the transient analysis. The MCPR operating limit is then determined based on an interpolation between the applicable limits for scram times of LCO 3.1.4,"Control Rod Scram Times" and realistic scram time analyses using the scram time fraction. The scram time fraction and corresponding MCPR operating limit must be determined once within 72 hours after each set of scram time tests required by SR 3.1.4.1, SR 3.1.4.2, SR 3.1.4.3 and SR 3.1.4.4 because the effective scram times may change during the cycle. The 72 hour Completion Time is acceptable due to the relatively minor changes in the scram time fraction expected during the fuel cycle.

REFERENCES

1. NUREG-0562, June 1979.

2. PL-NF-90-001-A, "Application of Reactor Analysis Methods for BWR Design and Analysis," July 1992, Supplement 1-A, August 1995, Supplement 2-A, July 1996, and Supplement 3-A, March 2001.

(continued)

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.3 LINEAR HEAT GENERATION RATE (LHGR)

BASES

BACKGROUND

The LHGR is a measure of the heat generation rate of a fuel rod in a fuel assembly at any axial location. Limits on LHGR are specified to ensure that fuel design limits are not exceeded anywhere in the core during normal operation. Exceeding the LHGR limit could potentially result in fuel damage and subsequent release of radioactive materials. Fuel design limits are specified to ensure that fuel system damage, fuel rod failure, or inability to cool the fuel does not occur during the normal operations identified in Reference 1.

APPLICABLE SAFETY ANALYSES

The analytical methods and assumptions used in evaluating the fuel system design are presented in References 1, 2, 3, and 4. The fuel assembly is designed to ensure (in conjunction with the core nuclear and thermal hydraulic design, plant equipment, instrumentation, and protection system) that fuel damage will not result in the release of radioactive materials in excess of the guidelines of 10 CFR, Parts 20, 50, and 100. The mechanisms that could cause fuel damage during operational transients and that are considered in fuel evaluations are:

a. Rupture of the fuel rod cladding caused by strain from the relative expansion of the UO_2 pellet; and

b. Severe overheating of the fuel rod cladding caused by inadequate cooling.

A value of 1% plastic strain of the fuel cladding has been defined as the limit below which fuel damage caused by overstraining of the fuel cladding is not expected to occur (Ref. 3).

Fuel design evaluations have been performed and demonstrate that the 1% fuel cladding plastic strain design limit is not exceeded during continuous operation with LHGRs up to the operating limit specified in the COLR. A separate evaluation was performed to determine the limits of LHGR during anticipated operational occurrences. This limit,

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

Protection Against Power Transients (PAPT), defined in Reference 4 provides the acceptance criteria for LHGRs calculated in evaluation of the AOOs.

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

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.

APPLICABILITY

The LHGR limits are derived from fuel design analysis that is limiting at high power level conditions. At core thermal power levels < 25% 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 ≥ 25% RTP.

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

(continued)

BASES

REFERENCES (continued)

4. ANF-89-98(P)(A) Revision 1 and Revision 1 Supplement 1, "Generic Mechanical Design Criteria for BWR Fuel Designs," Advanced Nuclear Fuels Corporation, May 1995.

5. Final Policy Statement on Technical Specifications Improvements, July 22, 1993 (58 FR 39132).

BASES

APPLICABLE SAFETY ANALYSES (continued)

(MCPR)," and LCO 3.2.3, "LINEAR HEAT GENERATION RATE (LHGR)," limit the initial margins to these operating limits at rated conditions so that specified acceptable fuel design limits are met during transients initiated from rated conditions. At initial power levels less than rated levels, the margin degradation of either the LHGR or the MCPR during a transient can be greater than at the rated condition event. This greater margin degradation during the transient is primarily offset by the larger initial margin to limits at the lower than rated power levels. However, power distributions can be hypothesized that would result in reduced margins to the pre-transient operating limit. When combined with the increased severity of certain transients at other than rated conditions, the SLs could be approached. At substantially reduced power levels, highly peaked power distributions could be obtained that could reduce thermal margins to the minimum levels required for transient events. To prevent or mitigate such situations, the MCPR margin degradation at reduced power and flow is factored into the power and flow dependent MCPR limits (LCO 3.2.2). For LHGR (Ref. 4), either the APRM gain is adjusted upward by the ratio of the core limiting MFLPD to the FRTP, or the flow biased APRM scram level is reduced by the ratio of FRTP to the core limiting MFLPD. The adjustment in the APRM gain can be performed provided it is during power ascension up to 90% of RATED THERMAL POWER, that the adjusted APRM reading does not exceed 100% of RATED THERMAL POWER, the required gain adjustment increment does not exceed 10% of RATED THERMAL POWER, and a notice of the adjustment is posted on the reactor control panel. Either of these adjustments effectively counters the increased severity of some events at other than rated conditions by proportionally increasing the APRM gain or proportionally lowering the flow biased APRM scram setpoints, dependent on the increased peaking that may be encountered.

The APRM gain and setpoints satisfy Criteria 2 and 3 of the NRC Policy Statement (Ref. 5).

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.2.4.1 and SR 3.2.4.2 (continued)

is operating within the assumptions of the safety analysis. These SRs are only required to determine the MFLPD and, assuming MFLPD is greater than FRTP, the appropriate gain or setpoint, and is not intended to be a CHANNEL FUNCTIONAL TEST for the APRM gain or flow biased neutron flux scram circuitry. The 24 hour Frequency of SR 3.2.4.1 is chosen to coincide with the determination of other thermal limits, specifically those for the APLHGR (LCO 3.2.1). The 24 hour Frequency is based on both engineering judgment and recognition of the slowness of changes in power distribution during normal operation. The 24 hour allowance after THERMAL POWER ≥ 25% RTP is achieved is acceptable given the large inherent margin to operating limits at low power levels and because the MFLPD must be calculated prior to exceeding 50% RTP unless performed in the previous 24 hours. When MFLPD is greater than FRTP, SR 3.2.4.2 must be performed. The 12 hour Frequency of SR 3.2.4.2 requires a more frequent verification when MFLPD is greater than the fraction of rated thermal power (FRTP) because more rapid changes in power distribution are typically expected.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 10, GDC 13, GDC 20, and GDC 23.
2. FSAR, Section 4.
3. FSAR, Section 15.
4. ANF-89-98(P)(A) Revision 1 and Revision 1 Supplement 1, "Generic Mechanical Design Criteria for BWR Fuel Designs," Advanced Nuclear Fuels Corporation, May 1995.
5. Final Policy Statement on Technical Specifications Improvements, July 22, 1993 (58 FR 39132).

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.4.1.4 (continued)

The Frequency of 24 months has shown that channel bypass failures between successive tests are rare.

SR 3.3.4.1.5

This SR ensures that the individual channel response times are less than or equal to the maximum values assumed in the accident analysis. The EOC-RPT SYSTEM RESPONSE TIME acceptance criteria are included in Reference 5.

A Note to the Surveillance states that breaker interruption time may be assumed from the most recent performance of SR 3.3.4.1.6. This is allowed since the time to open the contacts after energization of the trip coil and the arc suppression time are short and do not appreciably change, due to the design of the breaker opening device and the fact that the breaker is not routinely cycled.

EOC-RPT SYSTEM RESPONSE TIME tests are conducted on an 24 month STAGGERED TEST BASIS. For this SR, STAGGERED TEST BASIS means that each 24 month test shall include at least the logic of one type of channel input, turbine control valve fast closure or turbine stop valve closure such that both types of channel inputs are tested at least one per 48 months. Response times cannot be determined at power because operation of final actuated devices is required. Therefore, the 24 month Frequency is consistent with the typical industry refueling cycle and is based upon plant operating experience, which shows that random failures of instrumentation components that cause serious response time degradation, but not channel failure, are infrequent occurrences.

SR 3.3.4.1.6

This SR ensures that the RPT breaker interruption time (arc suppression time plus time to open the contacts) is provided to the EOC-RPT SYSTEM RESPONSE TIME test. The 60 month Frequency of the testing is based on the difficulty of performing the test and the reliability of the circuit breakers.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

Plant specific LOCA analyses have been performed assuming only one operating recirculation loop. These analyses have demonstrated that, in the event of a LOCA caused by a pipe break in the operating recirculation loop, the Emergency Core Cooling System response will provide adequate core cooling, provided that the APLHGR limit for SPC ATRIUM™-10 fuel is modified.

The transient analyses of Chapter 15 of the FSAR have also been performed for single recirculation loop operation and demonstrate sufficient flow coastdown characteristics to maintain fuel thermal margins during the abnormal operational transients analyzed provided the MCPR requirements are modified. During single recirculation loop operation, modification to the Reactor Protection System (RPS) average power range monitor (APRM) instrument setpoints is also required to account for the different relationships between recirculation drive flow and reactor core flow. The APLHGR, LHGR, and MCPR limits for single loop operation are specified in the COLR. The APRM flow biased simulated THERMAL POWER setpoint is in LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation." In addition, a restriction on recirculation pump speed is incorporated to address reactor vessel internals vibration concerns and assumptions in the event analysis.

General Design Criterion 10 (GDC 10) requires that the reactor core be designed with appropriate margin to assure that fuel design limits will not be exceeded during any condition of normal operation including anticipated operational occurrences. GDC 12 requires assurance that power oscillations which can result in conditions exceeding specified acceptable fuel design limits are either not possible or can be reliably and readily detected and suppressed. The ACTIONS in this section ensure compliance with GDC 12, thereby providing protection from exceeding the fuel MCPR safety limit.

BWR cores may exhibit thermal-hydraulic reactor instabilities in high power and low flow portions of the core power to flow operating domain. GDC 12 requires

(continued)

BASES

BACKGROUND (continued)

The ADS (Ref. 4) consists of 6 of the 16 S/RVs. It is designed to provide depressurization of the RCS during a small break LOCA if HPCI fails or is unable to maintain required water level in the RPV. ADS operation reduces the RPV pressure to within the operating pressure range of the low pressure ECCS subsystems (CS and LPCI), so that these subsystems can provide coolant inventory makeup. Each of the S/RVs used for automatic depressurization is equipped with two gas accumulators and associated inlet check valves. The accumulators provide the pneumatic power to actuate the valves.

APPLICABLE SAFETY ANALYSES

The ECCS performance is evaluated for the entire spectrum of break sizes for a postulated LOCA. The accidents for which ECCS operation is required are presented in References 5, 6, and 7. The required analyses and assumptions are defined in Reference 8. The results of these analyses are also described in Reference 9.

This LCO helps to ensure that the following acceptance criteria for the ECCS, established by 10 CFR 50.46 (Ref. 10), will be met following a LOCA, assuming the worst case single active component failure in the ECCS:

a. Maximum fuel element cladding temperature is ≤ 2200°F;

b. Maximum cladding oxidation is ≤ 0.17 times the total cladding thickness before oxidation;

c. Maximum hydrogen generation from a zirconium water reaction is ≤ 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react;

d. The core is maintained in a coolable geometry; and

e. Adequate long term cooling capability is maintained.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

SPC performed LOCA calculations for the SPC ATRIUM™-10 fuel design. The limiting single failures for the SPC analyses are discussed in Reference 11. For a large break LOCA, the SPC analyses identify the recirculation loop suction piping as the limiting break location. The SPC analysis identifies the failure of the LPCI injection valve into the intact recirculation loop as the most limiting single failure.

For a small break LOCA, the SPC analyses identify the recirculation loop discharge piping as the limiting break location, and a battery failure as the most severe single failure. One ADS valve failure is analyzed as a limiting single failure for events requiring ADS operation. The remaining OPERABLE ECCS subsystems provide the capability to adequately cool the core and prevent excessive fuel damage.

The ECCS satisfy Criterion 3 of the NRC Policy Statement (Ref. 15).

LCO

Each ECCS injection/spray subsystem and six ADS valves are required to be OPERABLE. The ECCS injection/spray subsystems are defined as the two CS subsystems, the two LPCI subsystems, and one HPCI System. The low pressure ECCS injection/spray subsystems are defined as the two CS subsystems and the two LPCI subsystems.

With less than the required number of ECCS subsystems OPERABLE, the potential exists that during a limiting design basis LOCA concurrent with the worst case single failure, the limits specified in Reference 10 could be exceeded. All ECCS subsystems must therefore be OPERABLE to satisfy the single failure criterion required by Reference 10.

LPCI subsystems may be considered OPERABLE during alignment and operation for decay heat removal when below the actual RHR cut in permissive pressure in MODE 3, if capable of

(continued)

B 3.7 PLANT SYSTEMS

B 3.7.6 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 full bypass capacity of the system is approximately 25% 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 five valves connected to the main steam lines between the main steam isolation valves and the turbine stop valve bypass valve chest. Each of these five valves is operated by hydraulic cylinders. The bypass valves are controlled by the pressure regulation function of the Turbine Electro Hydraulic Control System, as discussed in the FSAR, Section 7.7.1.5 (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.

APPLICABLE SAFETY ANALYSES

The Main Turbine Bypass System fast opening feature is assumed to function for all bypass valves assumed on the safety analysis during the turbine generator load rejection and feedwater controller failure transients, as discussed in the FSAR, Section 15.2.2 (Ref. 2). Opening the bypass valves during the pressurization event mitigates the increase in reactor vessel pressure, which affects the MCPR during the event. An inoperable Main Turbine Bypass System may result in an MCPR penalty.

The Main Turbine Bypass System satisfies Criterion 3 of the NRC Policy Statement. (Ref. 3)

(continued)

BASES (continued)

LCO

The Main Turbine Bypass System fast opening feature is required to be OPERABLE to limit peak pressure in the main steam lines and maintain reactor pressure within acceptable limits during events that cause rapid pressurization, so that the Safety Limit MCPR is not exceeded. With the Main Turbine Bypass System inoperable, modifications to the MCPR limits (LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)") may be applied to allow this LCO to be met. The MCPR limit for the inoperable Main Turbine Bypass System is specified in the COLR. An OPERABLE Main Turbine Bypass System requires the bypass valves to open in response to increasing main steam line pressure. Licensing analysis credits an OPERABLE Main Turbine Bypass System as having the bypass valve fast opening feature in response to turbine control valve or turbine stop valve closure. The cycle specific safety analyses assume a certain number of OPERABLE main turbine bypass valves as an input (i.e., one through five). Therefore, the Main Turbine Bypass System is considered OPERABLE when the number of OPERABLE bypass valves is greater than or equal to the number assumed in the safety analyses. The number of bypass valves assumed in the safety analyses is specified in the COLR. This response is within the assumptions of the applicable analysis (Ref. 2).

APPLICABILITY

The Main Turbine Bypass System is required to be OPERABLE at ≥ 25% RTP to ensure that the fuel cladding integrity Safety Limit is not violated during all applicable transients. As discussed in the Bases for LCO 3.2.2, sufficient margin to these limits exists at < 25% 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 and the MCPR 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 MCPR limits accordingly. The 2 hour Completion Time is reasonable, based on the time to complete the Required

(continued)

BASES

ACTIONS (continued)

B.1

Action and the low probability of an event occurring during this period requiring the Main Turbine Bypass System.

If the Main Turbine Bypass System cannot be restored to OPERABLE status or the MCPR limits for an inoperable Main Turbine Bypass System are not applied, THERMAL POWER must be reduced to < 25% RTP. As discussed in the Applicability section, operation at < 25% RTP results in sufficient margin to the required limits, and the Main Turbine Bypass System is not required to protect fuel integrity during the applicable transients. The 4-hour Completion Time is 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 3.7.6.1

Cycling each required main turbine bypass valve through one complete cycle of full travel (including the fast opening feature) demonstrates that the valves are mechanically OPERABLE and will function when required. The 31-day Frequency is based on engineering judgment, is consistent with the procedural controls governing valve operation, and ensures correct valve positions. Operating experience has shown that these components usually pass the SR when performed at the 31 day Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.

SR 3.7.6.2

The Main Turbine Bypass System is required to actuate automatically to perform its design function. This SR demonstrates that, with the required system initiation signals (simulate automatic actuation), the valves will actuate to their required position. The 24-month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and because of the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown the 24-month Frequency, which is based on the refueling cycle, is acceptable from a reliability standpoint.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.10.7.2

When the RWM provides conformance to the special test sequence, the test sequence must be verified to be correctly loaded into the RWM prior to control rod movement. This Surveillance demonstrates compliance with SR 3.3.2.1.8, thereby demonstrating that the RWM is OPERABLE. A Note has been added to indicate that this Surveillance does not need to be performed if SR 3.10.7.1 is satisfied.

REFERENCE

1. FSAR 15.4.9

2. PL-NF-90-001-A "Application of Reactor Analysis Methods for BWR Design and Analysis," July 1992 and Supplement 1-A, August 1995, Supplement 2-A, July 1996, and Supplement 3-A, March 2001.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.10.8.4

Periodic verification of the administrative controls established by this LCO will ensure that the reactor is operated within the bounds of the safety analysis. The 12 hour Frequency is intended to provide appropriate assurance that each operating shift is aware of and verifies compliance with these Special Operations LCO requirements.

SR 3.10.8.5

Coupling verification is performed to ensure the control rod is connected to the control rod drive mechanism and will perform its intended function when necessary. The verification is required to be performed any time a control rod is withdrawn to the "full out" notch position, or prior to declaring the control rod OPERABLE after work on the control rod or CRD System that could affect coupling. This Frequency is acceptable, considering the low probability that a control rod will become uncoupled when it is not being moved as well as operating experience related to uncoupling events.

SR 3.10.8.6

CRD charging water header pressure verification is performed to ensure the motive force is available to scram the control rods in the event of a scram signal. A minimum accumulator pressure is specified, below which the capability of the accumulator to perform its intended function becomes degraded and the accumulator is considered inoperable. The minimum accumulator pressure of 940 psig is well below the expected pressure of 1100 psig. The 7 day Frequency has been shown to be acceptable through operating experience and takes into account indications available in the control room.

REFERENCE

1. PL-NF-90-001-A, "Application of Reactor Analysis Methods for BWR Design and Analysis," July 1992 and Supplement 1-A, August 1995, Supplement 2-A, July 1996, and Supplement 3-A, March 2001.

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APPROVED AMENDMENT TO THE
UNIT 2 TECHNICAL SPECIFICATIONS BASES MANUAL
REVISION 43

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(continued)

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BASES

APPLICABLE SAFETY ANALYSES

2.1.1.3 Reactor Vessel Water Level (continued)

The reactor vessel water level SL has been established at the top of the active irradiated fuel to provide a point that can be monitored and to also provide adequate margin for effective action.

SAFETY LIMITS

The reactor core SLs are established to protect the integrity of the fuel clad barrier to the release of radioactive materials to the environs. SL 2.1.1.1 and SL 2.1.1.2 ensure that the core operates within the fuel design criteria. SL 2.1.1.3 ensures that the reactor vessel water level is greater than the top of the active irradiated fuel in order to prevent elevated clad temperatures and resultant clad perforations.

APPLICABILITY

SLs 2.1.1.1, 2.1.1.2, and 2.1.1.3 are applicable in all MODES.

SAFETY LIMIT VIOLATIONS

Exceeding an SL may cause fuel damage and create a potential for radioactive releases in excess of 10 CFR 100, "Reactor Site Criteria," limits (Ref. 3). Therefore, it is required to insert all insertable control rods and restore compliance with the SLs within 2 hours. The 2 hour Completion Time ensures that the operators take prompt remedial action and also ensures that the probability of an accident occurring during this period is minimal.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 10.

2. ANFB 524 (P)(A), Revision 2, "Critical Power Methodology for Boiling Water Reactors," Supplement 1 Revision 2 and Supplement 2, November 1990.

3. 10 CFR 100.

4. EMF-1997(P)(A), Revision 0, "ANFB-10 Critical Power Correlation," July 1998 and EMF-1997(P)(A) Supplement 1 Revision 0," ANFB-10 Critical Power Correlation: High Local Peaking Results," July 1998.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.1.1 (continued)

the highest worth control rod is determined by analysis or testing.

Local critical tests require the withdrawal of control rods in a sequence that is not in conformance with BPWS. This testing would therefore require re-programming or bypassing of the rod worth minimizer to allow the withdrawal of control rods not in conformance with BPWS, and therefore additional requirements must be met (see LCO 3.10.7, "Control Rod Testing—Operating").

The Frequency of 4 hours after reaching criticality is allowed to provide a reasonable amount of time to perform the required calculations and have appropriate verification.

During MODE 5, adequate SDM is required to ensure that the reactor does not reach criticality during control rod withdrawals. An evaluation of each planned in-vessel fuel movement during fuel loading (including shuffling fuel within the core) is required to ensure adequate SDM is maintained during refueling. This evaluation ensures that the intermediate loading patterns are bounded by the safety analyses for the final core loading pattern. For example, bounding analyses that demonstrate adequate SDM for the most reactive configurations during the refueling may be performed to demonstrate acceptability of the entire fuel movement sequence. These bounding analyses include additional margins to the associated uncertainties. Spiral offload/reload sequences inherently satisfy the SR, provided the fuel assemblies are reloaded in the same configuration analyzed for the new cycle. Removing fuel from the core will always result in an increase in SDM.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 26.
2. FSAR, Section 15.
3. PL-NF-90-001-A, "Application of Reactor Analysis Methods for BWR Design and Analysis," Sections 2.2 and 2.8, July 1992, Supplement 1-A, August 1995, Supplement 2-A, July 1996, and Supplement 3-A, March 2001. .
4. FSAR, Section 15.4.1.1.

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BASES

REFERENCES (continued)

5. Final Policy Statement on Technical Specifications Improvements, July 22, 1993 (58 FR 39132).

6. FSAR, Section 4.3.

7. PL-NF-90-001-A, "Application of Reactor Analysis Methods for BWR Design and Analysis," Section 2.4, July 1992, Supplement 1-A, August 1995, Supplement 2-A, July 1996, and Supplement 3-A, March 2001.

BASES

REFERENCES (continued)

4. FSAR, Section 15.0

5. PL-NF-90-001-A, Applicability of Reactor Analysis Methods for BWR Design and Analysis," Section 4.1.2, July 1992, and Supplement 1-A, August 1995, Supplement 2-A, July 1996, Supplement 3-A, March 2001.

6. Final Policy Statement on Technical Specifications Improvements, July 22, 1993 (58 FR 39132).

7. Letter from R.F. Janecek (BWROG) to R.W. Starostecki (NRC), "BWR Owners Group Revised Reactivity Control System Technical Specifications," BWROG-8754, September 17, 1987.

BASES

ACTIONS (continued)

B.1 and B.2

If nine or more OPERABLE control rods are out of sequence, the control rod pattern significantly deviates from the prescribed sequence. Control rod withdrawal should be suspended immediately to prevent the potential for further deviation from the prescribed sequence. Control rod insertion to correct control rods withdrawn beyond their allowed position is allowed since, in general, insertion of control rods has less impact on control rod worth than withdrawals have. Required Action B.1 is modified by a Note which allows the RWM to be bypassed to allow the affected control rods to be returned to their correct position. LCO 3.3.2.1 requires verification of control rod movement by a qualified member of the technical staff.

When nine or more OPERABLE control rods are not in compliance with BPWS, the reactor mode switch must be placed in the shutdown position within 1 hour. With the mode switch in shutdown, the reactor is shut down, and as such, does not meet the applicability requirements of this LCO. The allowed Completion Time of 1 hour is reasonable to allow insertion of control rods to restore compliance, and is appropriate relative to the low probability of a CRDA occurring with the control rods out of sequence.

SURVEILLANCE REQUIREMENTS

SR 3.1.6.1

The control rod pattern is verified to be in compliance with the BPWS at a 24 hour Frequency to ensure the assumptions of the CRDA analyses are met. The 24 hour Frequency was developed considering that the primary check on compliance with the BPWS is performed by the RWM (LCO 3.3.2.1), which provides control rod blocks to enforce the required sequence and is required to be OPERABLE when operating at ≤ 10% RTP.

REFERENCES

1. PL-NF-90-001-A, "Application of Reactor Analysis Methods for BWR Design and Analysis," Section 2.8, July 1992, Supplement 1-A, August 1995, Supplement 2-A, July 1996, and Supplement 3-A, March 2001.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.2.2.2 (continued)

scram time fraction. The scram time fraction and corresponding MCPR operating limit must be determined once within 72 hours after each set of scram time tests required by SR 3.1.4.1, SR 3.1.4.2, SR 3.1.4.3 and SR 3.1.4.4 because the effective scram times may change during the cycle. The 72 hour Completion Time is acceptable due to the relatively minor changes in the scram time fraction expected during the fuel cycle.

REFERENCES

1. NUREG-0562, June 1979.

2. PL-NF-90-001-A, "Application of Reactor Analysis Methods for BWR Design and Analysis," July 1992, Supplement 1-A, August 1995, Supplement 2-A, July 1996, and Supplement 3-A, March 2001.

3. PL-NF-87-001-A, "Qualification of Steady State core Physics Methods for BWR Design and Analysis," April 28, 1988.

4. PL-NF-89-005-A, "Qualification of Transient Analysis Methods for BWR Design and Analysis," July 1992, including Supplements 1 and 2.

5. XN-NF-80-19 (P)(A), Volume 4, Revision 1, "Exxon Nuclear Methodology for Boiling Water Reactors: Application of the ENC Methodology to BWR Reloads," Exxon Nuclear Company, June 1986.

6. NE-092-001, Revision 1, "Susquehanna Steam Electric Station Units 1 & 2: Licensing Topical Report for Power Uprate with Increased Core Flow," December 1992, and NRC Approval Letter: Letter from T. E. Murley (NRC) to R. G. Byram (PP&L), "Licensing Topical Report for Power Uprate With Increased Core Flow, Revision 0, Susquehanna Steam Electric Station, Units 1 and 2 (PLA-3788) (TAC Nos. M83426 and M83427)," November 30, 1993.

7. EMF-1997, Revision 0 (October 1997) and Supplement 1, Revision 0 (January 1998), "ANFB-10 Critical Power Correlation," and associated NRC SER dated 7/17/98.

8. XN-NF-79-71(P)(A) Revision 2, Supplements 1, 2, and 3, "Exxon Nuclear Plant Transient Methodology for Boiling Water Reactors," March 1986.

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BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

3. e. Manual Initiation (continued)

The Manual Initiation Function is not assumed in any accident or transient analyses in the FSAR. However, the Function is retained for overall redundancy and diversity of the HPCI function as required by the NRC in the plant licensing basis.

There is no Allowable Value for this Function since the channel is mechanically actuated based solely on the position of the push button. One channel of the Manual Initiation Function is required to be OPERABLE only when the HPCI System is required to be OPERABLE. Refer to LCO 3.5.1 for HPCI Applicability Bases.

Automatic Depressurization System

4.a, 5.a. Reactor Vessel Water Level—Low Low Low, Level 1

Low RPV water level indicates that the capability to cool the fuel may be threatened. Should RPV water level decrease too far, fuel damage could result. Therefore, ADS receives one of the signals necessary for initiation from this Function. The Reactor Vessel Water Level—Low Low Low, Level 1 is one of the Functions assumed to be OPERABLE and capable of initiating the ADS during the accident analyzed in Reference 1. The core cooling function of the ECCS, along with the scram action of the RPS, ensures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46.

Reactor Vessel Water Level—Low Low Low, Level 1 signals are initiated from four level instruments that sense the difference between the pressure due to a constant column of water (reference leg) and the pressure due to the actual water level (variable leg) in the vessel. Four channels of Reactor Vessel Water Level—Low Low Low, Level 1 Function are required to be OPERABLE only when ADS is required to be OPERABLE to ensure that no single instrument failure can preclude ADS initiation. Two channels input to ADS trip system A, while the other two channels input to ADS trip system B. Refer to LCO 3.5.1 for ADS Applicability Bases.

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B 3.7 PLANT SYSTEMS

B 3.7.6 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 full bypass capacity of the system is approximately 25% 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 five valves connected to the main steam lines between the main steam isolation valves and the turbine stop valve bypass valve chest. Each of these five valves is operated by hydraulic cylinders. The bypass valves are controlled by the pressure regulation function of the Turbine Electro Hydraulic Control System, as discussed in the FSAR, Section 7.7.1.5 (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.

APPLICABLE SAFETY ANALYSES

The Main Turbine Bypass System fast opening feature is assumed to function for all bypass valves assumed in the safety analysis during the turbine generator load rejection and feedwater controller failure transients, as discussed in the FSAR, Section 15.2.2 (Ref. 2). Opening the bypass valves during the pressurization event mitigates the increase in reactor vessel pressure, which affects the MCPR during the event. An inoperable Main Turbine Bypass System may result in an MCPR penalty.

The Main Turbine Bypass System satisfies Criterion 3 of the NRC Policy Statement. (Ref. 3)

(continued)

BASES (continued)

LCO

The Main Turbine Bypass System fast opening feature is required to be OPERABLE to limit peak pressure in the main steam lines and maintain reactor pressure within acceptable limits during events that cause rapid pressurization, so that the Safety Limit MCPR is not exceeded. With the Main Turbine Bypass System inoperable, modifications to the MCPR limits (LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)") may be applied to allow this LCO to be met. The MCPR limit for the inoperable Main Turbine Bypass System is specified in the COLR. An OPERABLE Main Turbine Bypass System requires the bypass valves to open in response to increasing main steam line pressure. Licensing analysis credits an OPERABLE Main Turbine Bypass System as having the bypass valve fast opening feature in response to turbine control valve or turbine stop valve closure. The cycle specific safety analyses assume a certain number of OPERABLE main turbine bypass valves as an input (i.e., one through five). Therefore the Main Turbine Bypass System is considered OPERABLE when the number of OPERABLE bypass valves is greater than or equal to the number assumed in the safety analyses. The number of bypass valves assumed in the safety analyses is specified in the COLR. This response is within the assumptions of the applicable analysis (Ref. 2).

APPLICABILITY

The Main Turbine Bypass System is required to be OPERABLE at ≥ 25% RTP to ensure that the fuel cladding integrity Safety Limit is not violated during all applicable transients. As discussed in the Bases for LCO 3.2.2, sufficient margin to these limits exists at < 25% 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 and the MCPR 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 MCPR 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.

(continued)

BASES

ACTIONS (continued)

B.1

If the Main Turbine Bypass System cannot be restored to OPERABLE status or the MCPR limits for an inoperable Main Turbine Bypass System are not applied, THERMAL POWER must be reduced to < 25% RTP. As discussed in the Applicability section, operation at < 25% RTP results in sufficient margin to the required limits, and the Main Turbine Bypass System is not required to protect fuel integrity during the during the applicable transients. The 4 hour Completion Time is 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 3.7.6.1

Cycling each required main turbine bypass valve through one complete cycle of full travel (including the fast opening feature) demonstrates that the valves are mechanically OPERABLE and will function when required. The 31 day Frequency is based on engineering judgment, is consistent with the procedural controls governing valve operation, and ensures correct valve positions. Operating experience has shown that these components usually pass the SR when performed at the 31 day Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.

SR 3.7.6.2

The Main Turbine Bypass System is required to actuate automatically to perform its design function. This SR demonstrates that, with the required system initiation signals (simulate automatic actuation), the valves will actuate to their required position. The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and because of the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown the 24 month Frequency, which is based on the refueling cycle, is acceptable from a reliability standpoint.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.10.7.2

When the RWM provides conformance to the special test sequence, the test sequence must be verified to be correctly loaded into the RWM prior to control rod movement. This Surveillance demonstrates compliance with SR 3.3.2.1.8, thereby demonstrating that the RWM is OPERABLE. A Note has been added to indicate that this Surveillance does not need to be performed if SR 3.10.7.1 is satisfied.

REFERENCE

1. FSAR 15.4.9
2. PL-NF-90-001-A "Application of Reactor Analysis Methods for BWR Design and Analysis," July 1992, Supplement 1-A, August 1995, Supplement 2-A, July 1996, and Supplement 3-A, March 2001.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.10.8.4

Periodic verification of the administrative controls established by this LCO will ensure that the reactor is operated within the bounds of the safety analysis. The 12 hour Frequency is intended to provide appropriate assurance that each operating shift is aware of and verfies compliance with these Special Operations LCO requirements.

SR 3.10.8.5

Coupling verification is performed to ensure the control rod is connected to the control rod drive mechanism and will perform its intended function when necessary. The verification is required to be performed any time a control rod is withdrawn to the "full out" notch position, or prior to declaring the control rod OPERABLE after work on the control rod or CRD System that could affect coupling. This Frequency is acceptable, considering the low probability that a control rod will become uncoupled when it is not being moved as well as operating experience related to uncoupling events.

SR 3.10.8.6

CRD charging water header pressure verification is performed to ensure the motive force is available to scram the control rods in the event of a scram signal. A minimum accumulator pressure is specified, below which the capability of the accumulator to perform its intended function becomes degraded and the accumulator is considered inoperable. The minimum accumulator pressure of 940 psig is well below the expected pressure of 1100 psig. The 7 day Frequency has been shown to be acceptable through operating experience and takes into account indications available in the control room.

REFERENCE

1. PL-NF-90-001-A, "Application of Reactor Analysis Methods for BWR Design and Analysis," July 1992, Supplement 1-A, August 1995, Supplement 2-A, July 1996, and Supplement 3-A, March 2001.