J. Kent Dittmer
Vice President, Engineering
P.O. Box 968, Mail Drop PE01
Richland, WA 99352-0968
Ph. 509-377-4348 F. 509-377-2354
jkdittmer@energy-northwest.com

January 27, 2020
GO2-20-004

10 CFR 50.90

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

Subject: **COLUMBIA GENERATING STATION, DOCKET NO. 50-397 APPLICATION TO REVISE TECHNICAL SPECIFICATIONS TO ADOPT TSTF-564, "SAFETY LIMIT MCPR"**

Dear Sir or Madam:

Pursuant to 10 CFR 50.90, Energy Northwest hereby requests a license amendment to revise the Columbia Generating Station (Columbia) Technical Specification (TS).

Energy Northwest requests adoption of TSTF-564 "Safety Limit MCPR," Revision 2, which is an approved change to the Improved Standard Technical Specifications (ISTS), into the Columbia TS. The proposed amendment revises the TS safety limit (SL) on minimum critical power ratio (MCPR) to reduce the need for cycle-specific changes to the value while still meeting the regulatory requirement for a SL.

The proposed change has been evaluated in accordance with 10 CFR 50.91(a)(1) using criteria in 10 CFR 50.92(c), and it has been determined that this change involves no significant hazards considerations. The bases for these determinations are included in Enclosure 1 of this submittal.

The proposed TS markup pages are included as Enclosure 2 to this submittal. Markups of the proposed TS Bases are included for information only as Enclosure 3 of this submittal. Clean pages of the proposed TS changes are included as Enclosure 4 of this submittal.

This letter and its enclosures contain no regulatory commitments.

Approval of the proposed amendment is requested within one year of the date of the submittal. Once approved, the amendment shall be implemented after startup from the next refueling outage (RFO).

In accordance with 10 CFR 50.91, Energy Northwest is notifying the State of Washington of this amendment request by transmitting a copy of this letter and enclosures to the designated State Official.

If there are any questions or if additional information is needed, please contact Ms. D.M. Wolfgramm, Licensing Supervisor, at 509-377-4792.

I declare under penalty of perjury that the foregoing is true and correct.

Executed this 27TH day of January, 2020.

Respectfully,

J. Kent Dittmer
Vice President, Engineering

Enclosures: As stated

cc: NRC RIV Regional Administrator
NRC NRR Project Manager
NRC Senior Resident Inspector/988C
CD Sonoda – BPA/1399 (email)
EFSECutc.wa.gov – EFSEC (email)
E Fordham – WDOH (email)
R Brice – WDOH (email)
L Albin – WDOH (email)

DESCRIPTION AND ASSESSMENT

1.0 DESCRIPTION

Energy Northwest requests adoption of Technical Specifications Task Force (TSTF) TSTF-564, "Safety Limit MCPR," Revision 2, which is an approved change to the Improved Standard Technical Specifications (ISTS), into the Columbia Generating Station (Columbia) Technical Specifications (TS). The proposed amendment revises the TS safety limit (SL) on minimum critical power ratio (MCPR) to reduce the need for cycle-specific changes to the value while still meeting the regulatory requirement for a SL.

2.0 ASSESSMENT

2.1 Applicability of Safety Evaluation

Energy Northwest has reviewed the safety evaluation for TSTF-564 provided to the Technical Specifications Task Force in a letter dated November 16, 2018. This review included a review of the Nuclear Regulatory Commission (NRC) staff's evaluation, as well as the information provided in TSTF-564. Energy Northwest has concluded that the justifications presented in TSTF-564 and the safety evaluation prepared by the NRC staff are applicable to Columbia and justify this amendment for the incorporation of the changes to the Columbia TS.

Columbia is currently fueled with Global Nuclear Fuel (GNF) GNF2 fuel bundles. The proposed Safety Limit in SL 2.1.1.2 is 1.07, consistent with Table 1 of TSTF-564.

The MCPR value calculated as the point at which 99.9% of the fuel rods would not be susceptible to boiling transition (i.e., reduced heat transfer) during normal operation and anticipated operational occurrences is referred to as $MCPR_{99.9\%}$. Technical Specification 5.6.3, "Core Operating Limits Report (COLR)," is revised to require the $MCPR_{99.9\%}$ value to be included in the cycle-specific COLR.

2.2 Variations

Columbia is a Boiling Water Reactor (BWR)/5 plant. Columbia's TS 2.1.1 and 5.6.3 are aligned with BWR/6 Standard Technical Specifications (STS) (NUREG-1434).

Columbia's TS contain a requirement that differs from the Standard Technical Specifications on which TSTF-564 was based, such as reactor steam dome pressure in SL 2.1.1.2, but this difference does not affect the applicability of the TSTF-564 justification.

3.0 REGULATORY ANALYSIS

3.1 No Significant Hazards Consideration Analysis

Energy Northwest requests adoption of Technical Specifications Task Force (TSTF) TSTF-564, "Safety Limit (SL) [Minimum Critical Power Ratio] (MCPR)," which is an approved change to the Improved Standard Technical Specifications (ISTS), into the Columbia Generating Station (Columbia) Technical Specifications (TS). The proposed change revises the TS safety limit on minimum critical power ratio (SLMCPR). The revised limit calculation method is based on using the Critical Power Ratio (CPR) data statistics and is revised from ensuring that 99.9% of the rods would not be susceptible to boiling transition to ensuring that there is a 95% probability at a 95% confidence level that no rods will be susceptible to transition boiling. A single SLMCPR value will be used instead of two values applicable when one or two recirculation loops are in operation. TS 5.6.3, "Core Operating Limits Report (COLR)," is revised to require the current SLMCPR value to be included in the COLR.

Energy Northwest has evaluated whether or not a significant hazards consideration is involved with the proposed amendment(s) by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

1. Does the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No

   The proposed amendment revises the TS SLMCPR and the list of core operating limits to be included in the Core Operating Limits Report (COLR). The SLMCPR is not an initiator of any accident previously evaluated. The revised safety limit values continue to ensure for all accidents previously evaluated that the fuel cladding will be protected from failure due to transition boiling. The proposed change does not affect plant operation or any procedural or administrative controls on plant operation that affect the functions of preventing or mitigating any accidents previously evaluated.

   Therefore, the proposed change does not involve a significant increase in the probability or consequence of an accident previously evaluated.

2. Does the proposed amendment create the possibility of a new or different kind of accident from any previously evaluated?

   Response: No

The proposed amendment revises the TS SLMCPR and the list of core operating limits to be included in the COLR. The proposed change will not affect the design function or operation of any structures, systems or components (SSCs). No new equipment will be installed. As a result, the proposed change will not create any credible new failure mechanisms, malfunctions, or accident initiators not considered in the design and licensing bases.

Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. Does the proposed amendment involve a significant reduction in a margin of safety?

Response: No

The proposed amendment revises the TS SLMCPR and the list of core operating limits to be included in the COLR. This will result in a change to a safety limit, but will not result in a significant reduction in the margin of safety provided by the safety limit. As discussed in the application, changing the SLMCPR methodology to one based on a 95% probability with 95% confidence that no fuel rods experience transition boiling during an anticipated transient instead of the current limit based on ensuring that 99.9% of the fuel rods are not susceptible to boiling transition does not have a significant effect on plant response to any analyzed accident. The SLMCPR and the TS Limiting Condition for Operation (LCO) on MCPR continue to provide the same level of assurance as the current limits and do not reduce a margin of safety.

Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, Energy Northwest concludes that the proposed change presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

3.2 Conclusion

In conclusion, based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission’s regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

4.0 ENVIRONMENTAL EVALUATION

The proposed change would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or would change an inspection or surveillance requirement. However, the proposed change does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluents that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed change meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed change.

Enclosure 2

Proposed Columbia Technical Specification Changes (Mark-Up)

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

2.1.1.1 With the reactor steam dome pressure < 686 psig or core flow < 10% rated core flow:

THERMAL POWER shall be ≤ 25% RTP.

2.1.1.2 With the reactor steam dome pressure ≥ 686 psig and core flow ≥ 10% rated core flow:

The MCPR shall be ≥ 1.~~10~~ 07~~for two recirculation loop operation or ≥ 1.13 for single recirculation loop operation~~.

2.1.1.3 Reactor vessel water level shall be greater than the top of active irradiated fuel.

2.1.2 Reactor Coolant System Pressure SL

Reactor steam dome pressure shall be ≤ 1325 psig.

2.2 SL Violations

With any SL violation, the following actions shall be completed within 2 hours:

2.2.1 Restore compliance with all SLs; and

2.2.2 Insert all insertable control rods.

5.6 Reporting Requirements

5.6.3 CORE OPERATING LIMITS REPORT (COLR)

a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:

1. The APLHGR for Specification 3.2.1;

2. The MCPR and $MCPR_{99.9\%}$ for Specification 3.2.2;

3. The LHGR for Specification 3.2.3;

4. Deleted;

5. The Oscillation Power Range Monitor (OPRM) Instrumentation for Specification 3.3.1.1; and

6. The Rod Block Monitor Instrumentation for Specification 3.3.2.1.

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

1. XN-NF-81-58(P)(A), "RODEX2 Fuel Rod Thermal-Mechanical Response Evaluation Model," Exxon Nuclear Company

2. XN-NF-85-67(P)(A), "Generic Mechanical Design for Exxon Nuclear Jet Pump BWR Reload Fuel," Exxon Nuclear Company

3. EMF-85-74(P) Supplement 1(P)(A) and Supplement 2(P)(A), "RODEX2A (BWR) Fuel Rod Thermal-Mechanical Evaluation Model," Siemens Power Corporation

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

5. XN-NF-80-19(P)(A) Volume 1, "Exxon Nuclear Methodology for Boiling Water Reactors - Neutronic Methods for Design and Analysis," Exxon Nuclear Company

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

Enclosure 3

Proposed Technical Specification Bases Markup Pages
For information Only

B 2.0 SAFETY LIMITS (SLs)

B 2.1.1 Reactor Core SLs

BASES

BACKGROUND

GDC 10 (Ref. 1) requires, and SLs ensure, that specified acceptable fuel design limits are not exceeded during steady state operation, normal operational transients, and anticipated operational occurrences (AOOs).

The fuel cladding integrity SL is set such that no fuel damage is calculated to occur if the limit is not violated. Because fuel damage is not directly observable, a stepback approach is used to establish an SL, such that the Minimum Critical Power Ratio (MCPR) is not less than the limit specified in Specification 2.1.1.2. MCPR greater than the specified limit represents a conservative margin relative to the conditions required to maintain fuel cladding integrity.

The fuel cladding is one of the physical barriers that separate the radioactive materials from the environs. The integrity of this cladding barrier is related to its relative freedom from perforations or cracking. Although some corrosion or use related cracking may occur during the life of the cladding, fission product migration from this source is incrementally cumulative and continuously measurable. Fuel cladding perforations, however, can result from thermal stresses, which occur from reactor operation significantly above design conditions.

While fission product migration from cladding perforation is just as measurable as that from use related cracking, the thermally caused cladding perforations signal a threshold beyond which still greater thermal stresses may cause gross, rather than incremental, cladding deterioration. Therefore, the fuel cladding SL is defined with a margin to the conditions that would produce onset of transition boiling (i.e., MCPR = 1.00). These conditions represent a significant departure from the condition intended by design for planned operation. This is accomplished by having a Safety Limit Minimum Critical Power Ratio (SLMCPR) design basis, referred to as $SLMCPR_{95/95}$, which corresponds to the 95% probability at a 95% confidence level (the 95/95 MCPR criterion) that transition boiling will not occur.

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

BASES

BACKGROUND (continued)

form. This weaker form may lose its integrity, resulting in an uncontrolled release of activity to the reactor coolant.

The reactor vessel water level SL ensures that adequate core cooling capability is maintained during all MODES of reactor operation. Establishment of Emergency Core Cooling System initiation setpoints higher than this safety limit provides margin such that the safety limit will not be reached or exceeded.

APPLICABLE SAFETY ANALYSES

The fuel cladding must not sustain damage as a result of normal operation and AOOs. The Tech Spec SL is set generically on a fuel product MCPR correlation basis as the MCPR which corresponds to a 95% probability at a 95% confidence level that transition boiling will not occur, referred to as $SLMCPR_{95/95}$. ~~The reactor core SLs are established to preclude violation of the fuel design criterion that a 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 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.

2.1.1.1 Fuel Cladding Integrity

GE critical power correlations are applicable for all critical power calculations at pressures > 686 psig and core flows > 10% of rated flow. For operation at low pressures or low flows, another basis is used, as follows:

Since the pressure drop in the bypass region is essentially all elevation head, the core pressure drop at low power and flows will always be > 4.5 psi. Analyses (Ref. 2) show that with a bundle flow of 28 E3 lb/hr, bundle pressure drop is nearly independent of bundle power and has a value of 3.5 psi. Thus, the bundle flow with a 4.5 psi driving head will be > 28 E3 lb/hr. Full scale ATLAS test data taken at pressures from 14.7 psia to 800 psia indicate that the fuel assembly critical power at this flow is approximately 3.35 MWt. With the design peaking factors, this corresponds to a THERMAL POWER > 50% RTP. Thus, a THERMAL POWER limit of 25% RTP for reactor pressure < 686 psig is conservative.

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APPLICABLE SAFETY ANALYSES (continued)

2.1.1.2 MCPR

The fuel cladding integrity SL is set such that no significant fuel damage is calculated to occur if the limit is not violated. Since the parameters that result in fuel damage are not directly observable during reactor operation, the thermal and hydraulic conditions that result in the onset of transition boiling have been used to mark the beginning of the region in which fuel damage could occur. Although it is recognized that the onset of transition boiling would not result in damage to BWR fuel rods, the critical power at which boiling transition is calculated to occur has been adopted as a convenient limit. The Technical Specifications SL value is dependent on the fuel product line and the corresponding MCPR correlation, which is cycle independent. The value is based on the Critical Power Ratio (CPR) data statistics and a 95% probability with 95% confidence that rods are not susceptible to boiling transition, referred to as $SLMCPR_{95/95}$. ~~However, the uncertainties in monitoring the core operating state and in the procedures used to calculate the critical power result in an uncertainty in the value of the critical power. Therefore, the fuel cladding integrity SL is defined as the critical power ratio in the limiting fuel assembly for which more than 99.9% of the fuel rods in the core are expected to avoid boiling transition, considering the power distribution within the core and all uncertainties.~~

~~The MCPR SL is determined using a statistical model that combines all the uncertainties in operating parameters and the procedures used to calculate critical power. The probability of the occurrence of boiling transition is determined using the approved GE critical power correlations. Details of the fuel cladding integrity SL calculation are given in Reference 2. Reference 2 also includes a tabulation of the uncertainties used in the determination of the MCPR SL and of the nominal values of the parameters used in the MCPR SL statistical analysis.~~
The SL is based on GNF2 fuel. For cores with a single fuel product line, the $SLMCPR_{95/95}$ is the $MCPR_{95/95}$ for the fuel type. For cores loaded with a mix of applicable fuel types, the $SLMCPR_{95/95}$ is based on the largest (i.e., most limiting) of the MCPR values for the fuel product lines that are fresh or once-burnt at the start of the cycle.

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 irradiated 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

BASES

SAFETY LIMITS

The reactor core SLs are established to protect the integrity of the fuel clad barrier to prevent 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 50.67 limits (Ref. 3). Therefore, it is required to insert all insertable control rods and restore compliance with the SL within 2 hours. The 2 hour Completion Time ensures that the operators take prompt remedial action and the probability of an accident occurring during this period is minimal.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 10.
2. NEDE-24011-P-A, “General Electric Standard Application for Reactor Fuel,” ~~Revision 16~~(most recent approved version referenced in COLR).
3. 10 CFR 50.67, “Accident Source Term.”
4. ~~NEDC-33419P, “GEXL97 Correlation Applicable to ATRIUM-10 Fuel,” Revision 0, June 2008.~~

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.2 MINIMUM CRITICAL POWER RATIO (MCPR)

BASES

BACKGROUND

MCPR is a ratio of the fuel assembly power that would result in the onset of boiling transition to the actual fuel assembly power. ~~The MCPR Safety Limit (SL) is such that 99.9% of the fuel rods avoid boiling transition if the limit is not violated (refer to the Bases for SL 2.1.1.2).~~ The operating limit MCPR is established to ensure that no fuel damage results during anticipated operational occurrences (AOOs) and that 99.9% of the fuel rods are not susceptible to boiling transition if the limit is not violated. Although fuel damage does not necessarily occur if a fuel rod actually experiences boiling transition (Ref. 5), the critical power at which boiling transition is calculated to occur has been adopted as a fuel design criterion.

The onset of transition boiling is a phenomenon that is readily detected during the testing of various fuel bundle designs. Based on these experimental data, correlations have been developed to predict critical bundle power (i.e., the bundle power level at the onset of transition boiling) for a given set of plant parameters (e.g., reactor vessel pressure, flow, and subcooling). Because plant operating conditions and bundle power levels are monitored and determined relatively easily, monitoring the MCPR is a convenient way of ensuring that fuel failures due to inadequate cooling do not occur.

APPLICABLE SAFETY ANALYSES

The analytical methods and assumptions used in evaluating the AOOs to establish the operating limit MCPR are presented in the FSAR, Chapters 4, 6, and 15, and References 6, 7, 8 and 9. To ensure that the MCPR Safety Limit (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~~ combined with the $MCPR_{99.9\%}$, the required operating limit MCPR is obtained.

$MCPR_{99.9\%}$ is determined to ensure more than 99.9% of the fuel rods in the core are not susceptible to boiling transition using a statistical model that combines all the uncertainties in operating parameters and the procedures used to calculate critical power. The probability of the occurrence of boiling transition is determined using the approved Critical Power correlations. Details of the $MCPR_{99.9\%}$ calculation are given in Reference 2. Reference 2 also includes a tabulation of the uncertainties

and the nominal values of the parameters used in the $MCPR_{99.9\%}$ statistical analysis.

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APPLICABLE SAFETY ANALYSES (continued)

The MCPR operating limits are derived from the $MCPR_{99.9\%}$ value and the transient analysis, and are dependent on the operating core flow and power rate ($MCPR_f$ and $MCPR_p$, respectively) to ensure adherence to fuel design limits during the worst transient that occurs with moderate frequency as identified in FSAR, Chapter 15.

Flow dependent MCPR limits are determined by steady-state thermal hydraulic methods using the three-dimensional BWR simulator code (Ref. 7) and a multi-channel thermal-hydraulic code (Refs. 8 and 9). $MCPR_f$ curves are provided based on the maximum credible flow runout transient for ASD operation (i.e., runout of both loops).

Power dependent MCPR limits ($MCPR_p$) are determined by approved transient analysis models ~~the three-dimensional BWR simulator code (Ref. 7) and a multi-channel thermal-hydraulic code~~ (Refs. 8 and 9). 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 scram trips are bypassed, high and low flow $MCPR_p$ operating limits are provided for operating between 25% RTP and the previously mentioned bypass power level.

The MCPR satisfies Criterion 2 of Reference 4.

LCO

The MCPR operating limits specified in the COLR ($MCPR_{99.9\%}$ value, $MCPR_f$ values and $MCPR_p$ values) are the result of the Design Basis Accident (DBA) and transient analysis. MCPR operating limits that include the effects of analyzed equipment out-of-service are also included in the COLR. The MCPR operating limits are determined by the larger of the $MCPR_f$ and $MCPR_p$ limits, which are based on the $MCPR_{99.9\%}$ limit specified in the COLR.

APPLICABILITY

The MCPR operating limits are primarily derived from transient analyses that are assumed to occur at high power levels. Below 25% RTP, the reactor is operating at a slow recirculation pump speed and the moderator void ratio is small. Surveillance of thermal limits below 25% 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 at 25% RTP is expected to be very large. 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

Enclosure 4

Proposed Columbia Technical Specification Changes (Re-Typed)

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

2.1.1.1 With the reactor steam dome pressure < 686 psig or core flow < 10% rated core flow:

THERMAL POWER shall be ≤ 25% RTP.

2.1.1.2 With the reactor steam dome pressure ≥ 686 psig and core flow ≥ 10% rated core flow:

The MCPR shall be ≥ 1.07.

2.1.1.3 Reactor vessel water level shall be greater than the top of active irradiated fuel.

2.1.2 Reactor Coolant System Pressure SL

Reactor steam dome pressure shall be ≤ 1325 psig.

2.2 SL Violations

With any SL violation, the following actions shall be completed within 2 hours:

2.2.1 Restore compliance with all SLs; and

2.2.2 Insert all insertable control rods.

5.6 Reporting Requirements

5.6.3 CORE OPERATING LIMITS REPORT (COLR)

a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:

1. The APLHGR for Specification 3.2.1;

2. The MCPR and $MCPR_{99.9\%}$ for Specification 3.2.2;

3. The LHGR for Specification 3.2.3;

4. Deleted;

5. The Oscillation Power Range Monitor (OPRM) Instrumentation for Specification 3.3.1.1; and

6. The Rod Block Monitor Instrumentation for Specification 3.3.2.1.

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

1. XN-NF-81-58(P)(A), "RODEX2 Fuel Rod Thermal-Mechanical Response Evaluation Model," Exxon Nuclear Company

2. XN-NF-85-67(P)(A), "Generic Mechanical Design for Exxon Nuclear Jet Pump BWR Reload Fuel," Exxon Nuclear Company

3. EMF-85-74(P) Supplement 1(P)(A) and Supplement 2(P)(A), "RODEX2A (BWR) Fuel Rod Thermal-Mechanical Evaluation Model," Siemens Power Corporation

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

5. XN-NF-80-19(P)(A) Volume 1, "Exxon Nuclear Methodology for Boiling Water Reactors - Neutronic Methods for Design and Analysis," Exxon Nuclear Company

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