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 the fuel design limits identified in Reference 1 are not exceeded and that the limits specified in 10 CFR 50.46 are not exceeded during the postulated design basis loss of coolant accident (LOCA). As a result, core geometry will be maintained following a design basis LOCA.

APPLICABLE SAFETY ANALYSES

The analytical methods and assumptions used in evaluating the fuel design limits are presented in Reference 1. The analytical methods and assumptions used in evaluating Design Basis Accidents (DBAs) and normal operations that determine APLHGR limits are presented in USAR, Chapters 4, 6, 15 and Appendix A, and in References 1, 2, and 3.

LOCA analyses are performed to ensure that the specified APLHGR limits are adequate to meet the PCT and maximum oxidation limits of 10 CFR 50.46. The analysis is performed using calculational models that are consistent with the requirements of 10 CFR 50, Appendix K. A complete discussion of the analysis code is provided in Reference 1. 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 an assembly. The APLHGR limits specified are equivalent to the LHGR of the highest powered fuel rod assumed in the LOCA analysis divided by its local peaking factor. A conservative multiplier is applied to the LHGR assumed in the LOCA analysis to account for the uncertainty associated with the measurement of the APLHGR.

For single recirculation loop operation, a conservative multiplier is applied to the exposure dependent APLHGR limits for two loop operation (Ref. 2). This limit is due to the conservative analysis assumption of an earlier departure from nucleate boiling with one recirculation loop available, resulting in a more severe cladding heatup during a LOCA.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The APLHGR satisfies Criterion 2 of Reference 4.

LCO

The APLHGR limits specified in the COLR are the result of fuel design and DBA analyses. For two recirculation loops operating, the limit is dependent on bundle exposure. 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 APLHGR limit by a conservative multiplier determined by a specific single recirculation loop analysis (Ref. 2).

APPLICABILITY

The APLHGR limits are primarily derived from fuel design evaluations and LOCA analyses that are assumed to occur at high power levels. Studies 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 (IRM) 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 operates 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 analyses may not be met. Therefore, prompt action is taken to restore the APLHGR(s) to within the required limits such that the plant will be operating within analyzed conditions and within the 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 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 < 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 periodically 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 12 hour allowance after THERMAL POWER ≥ 23% RTP is achieved is acceptable given the large inherent margin to operating limits at low power levels. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.
REFERENCES	1. NEDE-24011-P-A, "GE Standard Application for Reactor Fuel," (revision specified in the COLR). 2. USAR, Chapter 15B. 3. USAR, Chapter 15G. 4. 10 CFR 50.36(c)(2)(ii).

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 set such that 99.9% of the fuel rods are expected to 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). Although fuel damage does not necessarily occur if a fuel rod actually experiences boiling transition (Ref. 1), 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 USAR, Chapters 4, 6, 15 and Appendix A, and References 2, 3 and 6. 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.

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BASES

APPLICABLE SAFETY ANALYSES (continued)

The MCPR operating limits derived from the transient analysis are dependent on the operating core flow and power state ($MCPR_f$ and $MCPR_p$, respectively) to ensure adherence to fuel design limits during the worst transient that occurs with moderate frequency as identified in USAR, Chapter 15B. The determination of MCPR limits is discussed in Reference 6.

The MCPR operating limit is the greater of either the flow dependent MCPR limit ($MCPR_f$) or the power dependent MCPR limit ($MCPR_p$). The power dependent multiplier increases at lower powers due to the feedwater controller failure transient because, for lower powers, the mismatch between runout and initial feedwater flow increases. This results in an increase in reactor subcooling and more severe changes in thermal limits during the event at offrated power. The flow dependent limit increases at lower flows due to recirculation flow increase events because, for lower flows, the difference between initial flow and maximum possible core flow increases. This results in an increase in reactor power and more severe changes in thermal limits during the event at offrated flow.

The MCPR satisfies Criterion 2 of Reference 4.

LCO

The MCPR operating limits specified in the COLR are the result of the Design Basis Accident (DBA) and transient analysis. The MCPR operating limit 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 slow 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 documented in Reference 5 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

(continued)

APPLICABILITY (continued)

occurs. When in MODE 2, the intermediate range monitor (IRM) 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 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 the 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 periodically 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 12 hour allowance after THERMAL POWER reaches ≥ 23% RTP is acceptable given the large inherent margin to operating limits at low power levels. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

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BASES

SURVEILLANCE REQUIREMENTS (continued)

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 value of τ, which is a measure of the actual scram speed distribution compared with the assumed distribution. The MCPR operating limit is then determined based on an interpolation between the applicable limits for Option A (scram times of LCO 3.1.4, "Control Rod Scram Times") and Option B (realistic scram times) analyses. The parameter τ 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, and SR 3.1.4.4 because the effective scram speed distribution may change during the cycle or after maintenance that could affect scram times. The 72 hour Completion Time is acceptable due to the relatively minor changes in τ expected during the fuel cycle.

REFERENCES

1. NUREG-0562, June 1979.

2. NEDE-24011-P-A, "GE Standard Application for Reactor Fuel," (revision specified in the COLR).

3. Supplemental Reload Licensing Report for Nine Mile Point Nuclear Station Unit 2, (revision specified in the COLR).

4. 10 CFR 50.36(c)(2)(ii).

5. "BWR/6 Generic Rod Withdrawal Error Analysis," General Electric Standard Safety Analysis Report, GESSAR – II, Appendix 15B.

6. NEDC-33286P, "Nine Mile Point Nuclear Station Unit 2 – APRM/RBM/Technical Specifications/Maximum Extended Load Line Limit Analysis (ARTS/MELLLA)," March 2007.

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 the LHGR are specified to ensure that fuel design limits are not exceeded anywhere in the core during normal operation, including anticipated operational occurrences (AOOs). 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 anticipated operating conditions 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 and 5. 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 and 50. 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. The analysis also includes allowances for short term transient operation above the operating limit to account for AOOs, plus an allowance for densification power spiking.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The LHGR limit is the applicable rated-power, rated-flow LHGR limit multiplied by the smaller of either the flow dependent multiplier or the power dependent multiplier as specified in the COLR. The power dependent multiplier increases at lower powers due to the feedwater controller failure transient because, for lower powers, the mismatch between runout and initial feedwater flow increases. This results in an increase in reactor subcooling and more severe changes in thermal limits during the event at offrated power. The flow dependent multiplier increases at lower flows due to recirculation flow increase events because, for lower flows, the difference between initial flow and maximum possible core flow increases. This results in an increase in reactor power and more severe changes in thermal limits during the event at offrated flow.

The LHGR satisfies Criterion 2 of Reference 4.

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 < 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.

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.

(continued)

BASES

ACTIONS (continued)	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 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 3.2.3.1 The LHGRs are required to be initially calculated within 12 hours after THERMAL POWER is ≥ 23% RTP and periodically thereafter. They are compared with the specified limits in the COLR to ensure that the reactor is operating within the assumptions of the safety analysis. 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. The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.
REFERENCES	1. NEDE-24011-P-A, "GE Standard Application for Reactor Fuel," (revision specified in the COLR). 2. Supplemental Reload Licensing Report for Nine Mile Point Nuclear Station Unit 2, (revision specified in the COLR). 3. NUREG-0800, Section II A.2(g), Revision 2, July 1981. 4. 10 CFR 50.36(c)(2)(ii). 5. NEDC-33286P, "Nine Mile Point Nuclear Station Unit 2 – APRM/RBM/Technical Specifications/Maximum Extended Load Line Limit Analysis (ARTS/MELLLA)," March 2007.