Southern Nuclear

Cheryl A. Gayheart
Regulatory Affairs Director

3535 Colonnade Parkway
Birmingham, AL 35243
205 992 5316 tel
205 992 7795 fax

cagayhea@southernco.com

APR 2 3 2019

Docket Nos.: 50-321
50-366

NL-19-0250

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

Edwin I. Hatch Nuclear Plant - Units 1 and 2
Application to Revise Technical Specifications to Adopt TSTF-564, "Safety Limit MCPR"

Ladies and Gentlemen:

Pursuant to 10 CFR 50.90, Southern Nuclear Operating Company (SNC) is submitting a request for amendments to the Technical Specifications (TS) for Edwin I. Hatch Nuclear Plant (HNP), Units 1 and 2.

SNC requests adoption of TSTF-564, "Safety Limit MCPR," Revision 2, which is an approved change to the Improved Standard Technical Specifications (ISTS), into the HNP Unit 1 and Unit 2 TS. The proposed amendment for each unit 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 an SL.

Enclosure 1 provides a description and assessment of the proposed changes. Enclosure 2 provides the existing TS pages marked to show the proposed changes. Enclosure 3 provides revised (clean) TS pages. Enclosure 4 provides existing TS Bases pages marked to show the proposed changes for information only.

Approval of the proposed amendment is requested by October 30, 2019. Once approved, the amendment for Unit 1 will be implemented prior to reaching Mode 4 following Refueling Outage 1R29 (Spring 2020), and the amendment for Unit 2 will be implemented prior to reaching Mode 4 following Refueling Outage 2R26 (Spring 2021).

In accordance with 10 CFR 50.91, a copy of this application, with attachments, is being provided to the designated Georgia Official.

No regulatory commitments are made in this submittal.

If you should have any questions regarding this submittal, please contact Jamie Coleman at 205.992.6611.

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 23rd day of April 2019.

Respectfully submitted,

C. A. Gayheart
Director, Regulatory Affairs
Southern Nuclear Operating Company

CAG/TLE/scm

Enclosures:
1. Description and Assessment
2. Proposed Technical Specifications Changes (Mark-Up)
3. Revised Technical Specifications Pages
4. Proposed Technical Specification Bases Changes (Mark-Up) for Information Only

cc: Regional Administrator, Region II
NRR Project Manager – Hatch
Senior Resident Inspector – Hatch
Director, Environmental Protection Division – State of Georgia
RType: CHA02.004

Edwin I. Hatch Nuclear Plant - Units 1 and 2
Application to Revise Technical Specifications to Adopt TSTF-564, "Safety Limit"

Enclosure 1

Description and Assessment

1.0 DESCRIPTION

Southern Nuclear Operating Company (SNC) requests adoption of Technical Specification Task Force (TSTF) traveler TSTF-564, "Safety Limit MCPR," Revision 2, into the Edwin I. Hatch Nuclear Plant (HNP) Unit 1 and Unit 2 Technical Specifications (TS). TSTF-564 is an NRC-approved change to the Improved Standard Technical Specifications. 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 an SL.

2.0 ASSESSMENT

2.1 Applicability of Safety Evaluation

SNC 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 NRC staff's evaluation, as well as the information provided in TSTF-564. As described herein, SNC has concluded that the justifications presented in TSTF-564 and the safety evaluation prepared by the NRC staff are applicable to HNP Units 1 and 2 and justify this amendment for the incorporation of the changes to the HNP Units 1 and 2 TS.

The HNP Unit 1 reactor is currently fueled with GE14 and GNF2 fuel bundles. SNC currently plans to load GNF3 fuel bundles in Unit 1 during Refueling Outage 1R29 (Spring 2020). The HNP Unit 2 reactor is currently fueled with GNF2 and GNF3 fuel bundles, and previously contained GE14 fuel bundles. As permitted by Technical Specification 4.2.1, both reactors also contain a limited number of lead test assemblies in nonlimiting core locations. For both units, 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 5.6.5, "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

SNC is proposing the following variations from the TS changes described in TSTF-564 or the applicable parts of the NRC staff's safety evaluation.

The HNP TS utilize different numbering and titles than the Improved Standard Technical Specifications (ISTS) on which TSTF-564 was based. Specifically, Specification 5.6.3, "Core Operating Limits Report" is Specification 5.6.5, "Core Operating Limits Report (COLR)" in the SNC TS. This difference is administrative and does not affect the applicability of TSTF-564 to the HNP TS.

The HNP TS contain requirements that differ from the Standard Technical Specifications on which TSTF-564 was based, such as reactor steam dome pressure and Applicability (thermal power as % rated thermal power) in TS 3.2.2, but these differences do not affect the applicability of the TSTF-564 justification.

The traveler and safety evaluation discuss the applicable regulatory requirements and guidance, including the 10 CFR 50, Appendix A, General Design Criteria (GDC). HNP Unit 2 was licensed to the 10 CFR 50, Appendix A, GDC. However, HNP Unit 1 was not licensed to the 10 CFR 50, Appendix A, GDC. HNP Unit 1 was licensed to the applicable Atomic Energy Commission preliminary general design criteria identified in Federal Register 32 FR 10213, published July 11, 1967 (ADAMS Accession No. ML043310029). The applicable AEC proposed criteria were compared to the 10 CFR 50, Appendix A, General Design Criteria, as documented in the Hatch Updated Final Safety Analysis Report (UFSAR), Appendix F, "Conformance to the Atomic Energy Commission (AEC) Criteria," as discussed below.

TSTF-564 references 10 CFR 50, Appendix A, GDC 10, "Reactor Design," which states:

> The reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences.

A design evaluation of GDC Criterion 10 is included in Section F.3 of the HNP UFSAR, "Evaluation with Respect to 1971 General Design Criteria."

Following implementation of the proposed change, HNP Unit 1 will remain in compliance with applicable AEC design criteria as described in the HNP UFSAR. Therefore, this difference does not alter the conclusion that the proposed change is applicable to HNP Unit 1.

3.0 REGULATORY ANALYSIS

3.1 No Significant Hazards Consideration Analysis

Southern Nuclear Operating Company (SNC) requests adoption of TSTF-564, "Safety Limit MCPR," which is an approved change to the Improved Standard Technical Specifications (ISTS), into the Edwin I. Hatch Nuclear Plant (HNP), Units 1 and 2, 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 5.6.5, "Core Operating Limits Report (COLR)," is revised to require the current SLMCPR value to be included in the COLR.

SNC 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 consequences of an accident previously evaluated.

(2) Does the proposed amendment create the possibility of a new or different kind of accident from any accident 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, SNC 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.

Edwin I. Hatch Nuclear Plant - Units 1 and 2
<u>Application to Revise Technical Specifications to Adopt TSTF-564, "Safety Limit"</u>

Enclosure 2

Proposed Technical Specifications 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 < 685 psig or core flow < 10% rated core flow:

THERMAL POWER shall be ≤ 24% RTP.

2.1.1.2 With the reactor steam dome pressure ≥ 685 psig and core flow ≥ 10% rated core flow:

MCPR shall be ≥ ~~1.09~~1.07 ~~for two recirculation loop operation or ≥ 1.12 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 (RCS) 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.

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

2.1.1.1 With the reactor steam dome pressure < 685 psig or core flow < 10% rated core flow:

THERMAL POWER shall be ≤ 24% RTP.

2.1.1.2 With the reactor steam dome pressure ≥ 685 psig and core flow ≥ 10% rated core flow:

MCPR shall be ≥ ~~1.10~~1.07 ~~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 (RCS) 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.

No Changes. Included for Reference.

3.2 POWER DISTRIBUTION LIMITS

3.2.2 MINIMUM CRITICAL POWER RATIO (MCPR)

LCO 3.2.2 All MCPRs shall be greater than or equal to the MCPR operating limits specified in the COLR.

APPLICABILITY: THERMAL POWER ≥ 24% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Any MCPR not within limits.	A.1 Restore MCPR(s) to within limits.	2 hours
B. Required Action and associated Completion Time not met.	B.1 Reduce THERMAL POWER to < 24% RTP.	4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.2.1	Verify all MCPRs are greater than or equal to the limits specified in the COLR.	Once within 12 hours after ≥ 24% RTP <u>AND</u> In accordance with the Surveillance Frequency Control Program

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.2.2.2	Determine the MCPR limits.	Once within 72 hours after each completion of SR 3.1.4.1 <u>AND</u> Once within 72 hours after each completion of SR 3.1.4.2

3.2 POWER DISTRIBUTION LIMITS

3.2.2 MINIMUM CRITICAL POWER RATIO (MCPR)

LCO 3.2.2 All MCPRs shall be greater than or equal to the MCPR operating limits specified in the COLR.

APPLICABILITY: THERMAL POWER ≥ 24% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Any MCPR not within limits.	A.1 Restore MCPR(s) to within limits.	2 hours
B. Required Action and associated Completion Time not met.	B.1 Reduce THERMAL POWER to < 24% RTP.	4 hours

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.2.2.1	Verify all MCPRs are greater than or equal to the limits specified in the COLR.	Once within 12 hours after ≥ 24% RTP <u>AND</u> In accordance with the Surveillance Frequency Control Program

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.2.2.2	Determine the MCPR limits.	Once within 72 hours after each completion of SR 3.1.4.1 AND Once within 72 hours after each completion of SR 3.1.4.2

5.6 Reporting Requirements

5.6.2 Annual Radiological Environmental Operating Report (continued)

format of the table in the Radiological Assessment Branch Technical Position, Revision 1, November 1979. In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted in a supplementary report as soon as possible.

5.6.3 Radioactive Effluent Release Report

--NOTE--
A single submittal may be made for a multiple unit station. The submittal should combine sections common to all units at the station; however, for units with separate radwaste systems, the submittal shall specify the releases of radioactive material from each unit.

The Radioactive Effluent Release Report covering the operation of the unit shall be submitted in accordance with 10 CFR 50.36a. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit. The material provided shall be consistent with the objectives outlined in the ODCM and the Process Control Program and in conformance with 10 CFR 50.36a and 10 CFR 50, Appendix I, Section IV.B.1.

5.6.4 Deleted.

5.6.5 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 Average Planar Linear Heat Generation Rate for Specification 3.2.1.

2) The Minimum Critical Power Ratio (MCPR) for Specification 3.2.2 and the $MCPR_{99.9\%}$ value used to calculate the Specification 3.2.2 MCPR.

3) The Linear Heat Generation Rate for Specification 3.2.3.

(continued)

5.6 Reporting Requirements

5.6.2 Annual Radiological Environmental Operating Report (continued)

table in the Radiological Assessment Branch Technical Position, Revision 1, November 1979. In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted in a supplementary report as soon as possible.

5.6.3 Radioactive Effluent Release Report

--NOTE--

A single submittal may be made for a multiple unit station. The submittal should combine sections common to all units at the station; however, for units with separate radwaste systems, the submittal shall specify the releases of radioactive material from each unit.

The Radioactive Effluent Release Report covering the operation of the unit shall be submitted in accordance with 10 CFR 50.36a. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit. The material provided shall be consistent with the objectives outlined in the ODCM and the Process Control Program and in conformance with 10 CFR 50.36a and 10 CFR 50, Appendix I, Section IV.B.1.

5.6.4 Deleted.

5.6.5 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 Average Planar Linear Heat Generation Rate for Specification 3.2.1.

   2) The Minimum Critical Power Ratio (MCPR) for Specification 3.2.2 and the $MCPR_{99.9\%}$ value used to calculate the Specification 3.2.2 MCPR.

   3) The Linear Heat Generation Rate for Specification 3.2.3.

(continued)

Edwin I. Hatch Nuclear Plant - Units 1 and 2
Application to Revise Technical Specifications to Adopt TSTF-564, "Safety Limit"

Enclosure 3

Revised Technical Specifications Pages

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

2.1.1.1 With the reactor steam dome pressure < 685 psig or core flow < 10% rated core flow:

THERMAL POWER shall be ≤ 24% RTP.

2.1.1.2 With the reactor steam dome pressure ≥ 685 psig and core flow ≥ 10% rated core flow:

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 (RCS) 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.

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

2.1.1.1 With the reactor steam dome pressure < 685 psig or core flow < 10% rated core flow:

THERMAL POWER shall be ≤ 24% RTP.

2.1.1.2 With the reactor steam dome pressure ≥ 685 psig and core flow ≥ 10% rated core flow:

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 (RCS) 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

5.6.2 Annual Radiological Environmental Operating Report (continued)

format of the table in the Radiological Assessment Branch Technical Position, Revision 1, November 1979. In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted in a supplementary report as soon as possible.

5.6.3 Radioactive Effluent Release Report

---NOTE---

A single submittal may be made for a multiple unit station. The submittal should combine sections common to all units at the station; however, for units with separate radwaste systems, the submittal shall specify the releases of radioactive material from each unit.

The Radioactive Effluent Release Report covering the operation of the unit shall be submitted in accordance with 10 CFR 50.36a. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit. The material provided shall be consistent with the objectives outlined in the ODCM and the Process Control Program and in conformance with 10 CFR 50.36a and 10 CFR 50, Appendix I, Section IV.B.1.

5.6.4 Deleted.

5.6.5 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 Average Planar Linear Heat Generation Rate for Specification 3.2.1.

2) The Minimum Critical Power Ratio (MCPR) for Specification 3.2.2 and the $MCPR_{99.9\%}$ value used to calculate the Specification 3.2.2 MCPR.

3) The Linear Heat Generation Rate for Specification 3.2.3.

(continued)

5.6 Reporting Requirements

5.6.2 Annual Radiological Environmental Operating Report (continued)

table in the Radiological Assessment Branch Technical Position, Revision 1, November 1979. In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted in a supplementary report as soon as possible.

5.6.3 Radioactive Effluent Release Report

---NOTE---

A single submittal may be made for a multiple unit station. The submittal should combine sections common to all units at the station; however, for units with separate radwaste systems, the submittal shall specify the releases of radioactive material from each unit.

The Radioactive Effluent Release Report covering the operation of the unit shall be submitted in accordance with 10 CFR 50.36a. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit. The material provided shall be consistent with the objectives outlined in the ODCM and the Process Control Program and in conformance with 10 CFR 50.36a and 10 CFR 50, Appendix I, Section IV.B.1.

5.6.4 Deleted.

5.6.5 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 Average Planar Linear Heat Generation Rate for Specification 3.2.1.

2) The Minimum Critical Power Ratio (MCPR) for Specification 3.2.2 and the $MCPR_{99.9\%}$ value used to calculate the Specification 3.2.2 MCPR.

3) The Linear Heat Generation Rate for Specification 3.2.3.

(continued)

Edwin I. Hatch Nuclear Plant - Units 1 and 2
Application to Revise Technical Specifications to Adopt TSTF-564, "Safety Limit"

Enclosure 4

Proposed Technical Specification Bases Changes (Mark-Up) 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 MCPR is not less than the limit specified in Specification 2.1.1.2 for General Electric (GE) Company fuel. 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 a 95% probability at a 95% confidence level (the 95/95 MCPR criterion) that transition boiling will not occur.~~The MCPR fuel cladding integrity SL ensures that during normal operation and during AOOs, at least 99.9% of the fuel rods in the core do not experience transition boiling.~~

(continued)

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.

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 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 SL.

2.1.1.1 Fuel Cladding Integrity

GE critical power correlations are applicable for all critical power calculations at pressures ≥ 685 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 x 10^3 lb/hr, bundle pressure drop is nearly independent of bundle

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

power and has a value of 3.5 psi. Thus, the bundle flow with a 4.5 psi driving head will be > 28 x 10^3 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 24% RTP for reactor pressure < 685 psig is conservative. Additional information on low flow conditions is provided in Reference 3.

2.1.1.2 MCPR

The fuel cladding integrity SL is set such that no 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 Specification 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 MCPR$_{95/95}$.

The SL is based on GE14, GNF2, and GNF3 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.

~~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 General Electric 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.~~

(continued)

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 MCPR is not less than the limit specified in Specification 2.1.1.2 for General Electric (GE) Company fuel. 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 a 95% probability at a 95% confidence level (the 95/95 MCPR criterion) that transition boiling will not occur.~~The MCPR fuel cladding integrity SL ensures that during normal operation and during AOOs, at least 99.9% of the fuel rods in the core do not experience transition boiling.~~

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

(continued)

BASES

BACKGROUND (continued)

to a structurally weaker 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 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 Safety Limit.

2.1.1.1 Fuel Cladding Integrity

GE critical power correlations are applicable for all critical power calculations at pressures ≥ 685 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 28×10^3 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 > 28×10^3 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 24% RTP for reactor pressure < 685 psig is conservative. Additional information on low flow conditions is provided in Reference 3.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

2.1.1.2 MCPR

The fuel cladding integrity SL is set such that no 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 Specification 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 $MCPR_{95/95}$.

The SL is based on GE14, GNF2, and GNF3 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.

~~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 General Electric 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.~~

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

(continued)

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), 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 experienced 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 References 2, 3, 4, 5, 6, 7, and 8. 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 combined with~~added to~~ the $MCPR_{99.9\%}$ ~~SL~~, 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.

BASES

APPLICABLE SAFETY ANALYSES (continued)

The MCPR operating limits are derived from the $MCPR_{99.9\%}$ value and transient analysis, and 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 (Refs. 6, 7, and 8). Flow dependent MCPR limits are determined by steady state thermal hydraulic methods with key physics response inputs benchmarked using the three dimensional BWR simulator code (Ref. 9) to analyze slow flow runout transients. The operating limit is dependent on the maximum core flow limiter setting in the Recirculation Flow Control System.

Power dependent MCPR limits ($MCPR_p$) are determined by approved transient analysis models~~mainly by the one dimensional transient code~~ (Ref. 10). Due to the sensitivity of the transient response to initial core flow levels at power levels below those at which the turbine stop valve closure and turbine control valve fast closure scrams are bypassed, high and low flow $MCPR_p$ operating limits are provided for operating between 24% RTP and the previously mentioned bypass power level.

The MCPR satisfies Criterion 2 of the NRC Policy Statement (Ref. 11).

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. The operating limit MCPR is 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 24% RTP, the reactor is operating at a minimum recirculation pump speed and the moderator void ratio is small. Surveillance of thermal limits below 24% RTP is unnecessary due to the large inherent margin that ensures that the MCPR SL is not exceeded even if a limiting transient occurs. Statistical analyses indicate that the nominal value of the initial MCPR expected at 24% 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 24% RTP. This trend is expected to continue to the 5% to 15% power range when entry into MODE 2 occurs. When in MODE 2, the intermediate range monitor provides rapid scram initiation for any significant power increase transient, which effectively eliminates any MCPR compliance concern. Therefore, at THERMAL POWER levels < 24% RTP, the reactor is operating with substantial margin to the MCPR limits and this LCO is not required

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), 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 experienced 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 the AOOs to establish the operating limit MCPR are presented in References 2, 3, 4, 5, 6, 7, and 8. 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 combined with~~added to~~ the $MCPR_{99.9\%}$ ~~SL~~, 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.

(continued)

APPLICABLE SAFETY ANALYSES (continued)

The MCPR operating limits are derived from the $MCPR_{99.9\%}$ value and transient analysis, and 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, (Refs. 6, 7, and 8). Flow dependent MCPR limits are determined by steady state thermal hydraulic methods with key physics response inputs benchmarked using the three dimensional BWR simulator code (Ref. 9) to analyze slow flow runout transients. The operating limit is dependent on the maximum core flow limiter setting in the Recirculation Flow Control System.

Power dependent MCPR limits ($MCPR_p$) are determined by approved transient analysis models~~mainly by the one dimensional transient code~~ (Ref. 10). Due to the sensitivity of the transient response to initial core flow levels at power levels below those at which the turbine stop valve closure and turbine control valve fast closure scrams are bypassed, high and low flow $MCPR_p$ operating limits are provided for operating between 24% RTP and the previously mentioned bypass power level.

The MCPR satisfies Criterion 2 of the NRC Policy Statement (Ref. 11).

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. The operating limit MCPR is 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 24% RTP, the reactor is operating at a minimum recirculation pump speed and the moderator void ratio is small. Surveillance of thermal limits below 24% RTP is unnecessary due to the large inherent margin that ensures that the MCPR SL is not exceeded even if a limiting transient occurs. Statistical analyses indicate that the nominal value of the initial MCPR expected at 24% 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 24% RTP. This trend is expected to continue to the 5% to 15% power range when entry into MODE 2 occurs. When in MODE 2, the intermediate range monitor provides rapid scram initiation for any significant power increase transient, which effectively eliminates any MCPR compliance concern. Therefore, at THERMAL POWER levels < 24% RTP, the reactor is operating with substantial margin to the MCPR limits and this LCO is not required.

(continued)