August 28, 2017

TSTF-17-12
PROJ0753

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

SUBJECT: Transmittal of TSTF-564, "Safety Limit MCPR"

Enclosed for NRC review is TSTF-564, "Safety Limit MCPR."

The following information is provided to assist the NRC staff in prioritizing their review of TSTF-564:

- Applicability: TSTF-564 is applicable to all General Electric Boiling Water Reactor (BWR) designs.
- Classification: TSTF-564 will revise the current Technical Specification Safety Limit calculation method such that it is no longer cycle-specific.
- Specialized Resource Availability: The TSTF requests approval of the traveler within one year. NRC approval of TSTF-564 will reduce the burden on licensees and the Nuclear Regulatory Commission associated with the preparation and review of license amendments with accelerated NRC review.

The Technical Specifications Task Force should be billed for the review of the traveler.

11921 Rockville Pike, Suite 100, Rockville, MD 20852
Phone: 301-984-4400, Fax: 301-984-7600
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Should you have any questions, please do not hesitate to contact us.

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Technical Specifications Task Force
Improved Standard Technical Specifications Change Traveler

Safety Limit MCPR

NUREGs Affected: ☐ 1430 ☐ 1431 ☐ 1432 ☑ 1433 ☑ 1434 ☐ 2194

Classification: 1) Technical Change

Recommended for CLIIP?: Yes

Correction or Improvement: Improvement

NRC Fee Status: Not Exempt

Changes Marked on ISTS Rev 4.0

See attached justification.

Revision History

OG Revision 0 **Revision Status: Active**

Revision Proposed by: BWROG

Revision Description:
Original Issue

Owners Group Review Information

Date Originated by OG: 30-May-17

Owners Group Comments
Reviewed by the Owners Groups in October 2016.

A presubmittal teleconfernce was held on March 20, 2017. The traveler was revised to address NRC comments:

Owners Group Resolution: Approved Date: 18-Jul-17

TSTF Review Information

TSTF Received Date: 02-Aug-17 Date Distributed for Review 02-Aug-17

TSTF Comments:

(No Comments)

TSTF Resolution: Approved Date: 28-Aug-17

NRC Review Information

NRC Received Date: 28-Aug-17

Affected Technical Specifications

Bkgnd 2.1.1 Bases	Reactor Core SLs
S/A 2.1.1 Bases	Reactor Core SLs
SL 2.1.1.2	Safety Limit MCPR
Bkgnd 3.2.2 Bases	Minimum Critical Power Ratio (MCPR)
S/A 3.2.2 Bases	Minimum Critical Power Ratio (MCPR)
LCO 3.2.2 Bases	Minimum Critical Power Ratio (MCPR)
Appl. 3.2.2 Bases	Minimum Critical Power Ratio (MCPR)
5.6.3	Core Operating Limits Report

1. SUMMARY DESCRIPTION

The Technical Specification (TS) Safety Limit (SL) value and method of calculation for the Minimum Critical Power Ratio (MCPR) limit, SL 2.1.1.2, is revised for Boiling Water Reactor (BWR) plants using Global Nuclear Fuel or Westinghouse fuel. The proposed change is not applicable to plants using Areva fuel. The revised 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 transition boiling to ensuring that there is a 95% probability at a 95% confidence level that no rods will be susceptible to transition boiling. The revised MCPR SL is consistent with the regulatory requirements while being cycle-independent, thereby minimizing the need for TS license amendment requests to revise this value for each operating cycle.

2. DETAILED DESCRIPTION

2.1. Current Design and Licensing Basis

MCPR is defined in Section 1.1 of the BWR TS as:

> The MCPR shall be the smallest critical power ratio (CPR) that exists in the core [for each class of fuel]. The CPR is that power in the assembly that is calculated by application of the appropriate correlation(s) to cause some point in the assembly to experience boiling transition, divided by the actual assembly operating power.

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 are 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). Although fuel damage does not necessarily occur if a fuel rod experiences transition boiling, the critical power at which boiling transition is calculated to occur has been adopted as a fuel design criterion. Because plant operating conditions and bundle power levels are monitored and determined relatively easily, monitoring the MCPR is a convenient metric for ensuring that fuel failures due to inadequate cooling do not occur.

The TS contain two limits on MCPR: a safety limit (herein referred to as the SLMCPR) and a Limiting Condition for Operation (LCO) operating limit (herein referred to as the OLMCPR).

Title 10 of the Code of Federal Regulations (10 CFR), Part 50, paragraph 50.36(c)(1) defines "safety limits" as limits upon important process variables that are found to be necessary to reasonably protect the integrity of certain of the physical barriers that guard against the uncontrolled release of radioactivity. In the case of the MCPR safety limit, the physical barrier being protected is the fuel rod cladding. The current SLMCPR is calculated as the point at which 99.9% of the fuel rods are not susceptible to transition boiling (i.e., reduced heat transfer) during normal operation and anticipated operational occurrences, herein referred to as $MCPR_{99.9\%}$. The $MCPR_{99.9\%}$ limit is calculated for each BWR on a cycle-by-cycle basis using approved methodologies.

An LCO is defined in 10 CFR 50.36(c)(2) as the lowest functional capability or performance level of equipment required for safe operation of the facility. The OLMCPR LCO is required to be met to ensure that no fuel damage results during anticipated operational occurrences (AOOs). To ensure that the measured MCPR does not exceed the SLMCPR during any AOO that occurs with moderate frequency, transients are analyzed to determine the largest reduction in critical power ratio. The limiting transient yields the largest change in CPR (ΔCPR) during the event. The largest ΔCPR is combined with the $MCPR_{99.9\%}$ value to determine the OLMCPR LCO limit.

Together, SLMCPR and OLMCPR ensure that no fuel damage occurs due to transition boiling during normal operation or AOOs.

2.2. Current Technical Specifications Requirements

NUREG-1433 and NUREG-1434[1], Safety Limit 2.1.1.2 states:

2.1.1.2 With the reactor steam dome pressure ≥ 785 psig and core flow ≥ 10% rated core flow:

MCPR shall be ≥ [1.07] for two recirculation loop operation or ≥ [1.08] for single recirculation loop operation.

The values in brackets, [1.07] and [1.08], are plant-specific limits. The reactor steam dome pressure and core flow values are also plant-specific and differences do not affect the applicability of the proposed change.

NUREG-1433 and NUREG-1434, LCO 3.2.2, "Minimum Critical Power Ratio (MCPR)," states:

All MCPRs shall be greater than or equal to the MCPR operating limits specified in the [Core Operating Limits Report] COLR.

NUREG-1433 and NUREG-1434, LCO 3.2.2 is applicable when thermal power is ≥ 25% of rated thermal power. Plant-specific TS may have a different Applicability, which does not affect the justification for the proposed change.

NUREG-1433 and NUREG-1434, TS 5.6.3, "Core Operating Limits Report," states:

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:

[The individual specifications that address core operating limits must be referenced here.]

[1] NUREG-1433 is based on the BWR/4 plant design, but is also applicable of the BWR/2, BWR/3, and, for some requirements, to the BWR/5 plant designs. NUREG-1434 is based on the BWR/6 plant design, and is applicable, for some requirements, to the BWR/5 plant design.

2.3. Reason for the Proposed Change

The current SLMCPR (i.e., $MCPR_{99.9\%}$) is affected by the cycle-specific design, such as core power distribution, fuel type, and operating power-flow domain. These factors generally vary enough from cycle-to-cycle that changes to the SLMCPR TS values are common. The subsequent cycle core design is dependent on the core burnup of the previous cycle. As a result, the core design for the subsequent cycle is typically finalized late in the previous fuel cycle. Consequently, license amendment requests to modify the SLMCPR typically request an accelerated NRC review (i.e., less than the typical period of one year) to support the scheduled start of the subsequent fuel cycle. A review of the NRC Agency-wide Documents Access and Management System (ADAMS) identified a number of approved license amendments to revise the SLMCPR in 2015 and 2016.

The Babcock & Wilcox, Westinghouse, Combustion Engineering, and Advanced Passive 1000 (AP1000®) Standard Technical Specifications (NUREG-1430, NUREG-1431, NUREG-1432, and NUREG-2194) safety limits on fuel cladding are based on a Departure from Nucleate Boiling Ratio (DNBR) limit. The PWR DNBR limits are roughly analogous to the BWR SLMCPR, in that both protect fuel cladding integrity from inadequate cooling. The PWR DNBR safety limit corresponds to a 95% probability at a 95% confidence level that DNB will not occur, vice the BWR SLMCPR that is based on ensuring that 99.9% of the fuel rods will not be susceptible to boiling transition. Either approach is statistically valid, but this difference results in a PWR safety limit that is only dependent on the fuel type(s) in the reactor and the corresponding DNBR correlations. The PWR DNBR Safety Limits are not cycle dependent and are typically only revised when the type of fuel changes.

2.4. Description of the Proposed Change

The proposed change revises the standard TS in NUREG-1433 and NUREG-1434 for all BWR plants using Global Nuclear Fuel or Westinghouse fuel. The proposed change is not applicable to BWR plants using AREVA fuel due to differences in core reload design methodology.

The proposed change substantially reduces the need for cycle-specific changes to the SLMCPR and eliminates the need for accelerated NRC review of those changes.

The NUREG-1433 and NUREG-1434 Safety Limit 2.1.1.2 is revised to state:

> 2.1.1.2 With the reactor steam dome pressure ≥ 785 psig and core flow ≥ 10% rated core flow:
>
> MCPR shall be ≥ [1.07] *[*for two recirculation loop operation or ≥ [1.08] for single recirculation loop operation*]*.

The phrase "for two recirculation loop operation or ≥ [1.08] for single recirculation loop operation" are shown in brackets to retain compatibility for BWR plants that do not adopt the proposed change. The proposed SLMCPR methodology is not dependent on the number of recirculation loops in operation, so the distinction between a single loop and two loop operation is not needed.

Plants adopting the proposed change will revise their plant-specific SL to state:

2.1.1.2 With the reactor steam dome pressure ≥ 785 psig and core flow ≥ 10% rated core flow:

MCPR shall be ≥ [1.07].

The bracketed limit "[1.07]" will be replaced with a revised SLMCPR that ensures there is a 95% probability at a 95% confidence level that no rods will be susceptible to transition boiling, and is referred to as $SLMCPR_{95/95}$. The reactor steam dome pressure and core flow values are also plant-specific. Differences between the Standard Technical Specifications values and the plant-specific values do not affect the applicability of the proposed change.

For plants with Global Nuclear Fuel or Westinghouse fuel, the single $SLMCPR_{95/95}$ is based on the fuel type in the reactor core.

Table 1: Proposed $MCPR_{95/95}$ Values by Vendor and Fuel Bundle Type

Vendor	Fuel Type	Proposed $MCPR_{95/95}$
Global Nuclear Fuel	GE14	1.05
Global Nuclear Fuel	GNF2	1.07
Global Nuclear Fuel	GNF3	1.07
Westinghouse	Optima2	1.06
Westinghouse	Optima3	1.06

The derivation of these values is described in proprietary letters to the NRC from Global Nuclear Fuel and Westinghouse (References 1 and 2). When new fuel types are developed, the fuel vendor will describe to the NRC the derivation of the $MCPR_{95/95}$ value for that fuel type. This description may be referenced by a licensee requesting a change to $SLMCPR_{95/95}$.

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 $MCPR_{95/95}$ values for the fuel products that are fresh or once-burnt at the start of the cycle.

LCO 3.2.2 is not affected by the proposed change. However, licensees adopting the proposed change will include the $MCPR_{99.9\%}$ value (i.e., the value equivalent to the current, cycle-dependent SLMCPR) in the COLR values for LCO 3.2.2.

TS 5.6.3, "Core Operating Limits Report,", paragraph a, is revised to require the $MCPR_{99.9\%}$ value to be in the cycle-specific 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:

   [The individual specifications that address core operating limits must be referenced here. *The $MCPR_{99.9\%}$ value used to calculate the LCO 3.2.2, "MCPR," limit shall be specified in the COLR.*]

The proposed change is supported by changes to the TS Bases. The SL 2.1.1.2 Bases and TS 3.2.2 Bases are revised to reflect the proposed limits for Global Nuclear Fuel and Westinghouse fuel. In sections of the Bases applicable to all fuel types, the existing text and the proposed text are both presented in brackets, signifying that the licensee should choose the applicable description. A reviewer's note is added to the Bases to explain these options. The regulation at 10 CFR 50.50.36, states, "A summary statement of the bases or reasons for such specifications, other than those covering administrative controls, shall also be included in the application, but shall not become part of the technical specifications." A licensee may make changes to the TS Bases without prior NRC staff review and approval in accordance with the Technical Specifications Bases Control Program. The proposed TS Bases changes are consistent with the proposed TS changes and provide the purpose for each requirement in the specification consistent with the Commission's Final Policy Statement on Technical Specifications Improvements for Nuclear Power Reactors, dated July 2, 1993 (58 FR 39132). Therefore, the Bases changes are provided for information and approval of the Bases is not requested.

A model application is included in the proposed change as Enclosure 1. The model may be used by licensees desiring to adopt the traveler following NRC approval.

3. TECHNICAL EVALUATION

The proposed change revises the TS limit for the SLMCPR for the applicable plants and places the existing SLMCPR value (i.e., $MCPR_{99.9\%}$) in the COLR. 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 transition boiling to ensuring that there is a 95% probability at a 95% confidence level that no rods will be susceptible to transition boiling. The revised SLMCPR, referred to as $SLMCPR_{95/95}$, is based only on the CPR correlation uncertainty determined for the Global Nuclear Fuel or Westinghouse fuel type. Plant and cycle-specific uncertainties are not included in the $SLMCPR_{95/95}$. These uncertainties are currently and will continue to be included in the OLMCPR LCO. Reactor coolant flow is one of the uncertainties removed from the SLMCPR calculation and retained in the OLMCPR. Therefore, the SLMCPR for dual recirculation loop operation and single recirculation loop operation are replaced with a single $SLMCPR_{95/95}$.

The LCO 3.2.2 limits (i.e., the OLMCPR values) are not changed and will be based on the existing SLMCPR, referred to as $MCPR_{99.9\%}$. The OLMCPR will continue to be determined based on the transient ΔCPR components and the cycle-specific $MCPR_{99.9\%}$ value that will be included in the COLR. Therefore, the margin to boiling transition remains unchanged.

3.1. Statistical Treatment of $MCPR_{95/95}$

For each Global Nuclear Fuel and Westinghouse BWR fuel product (designated i), the $MCPR_{95/95}(i)$ is calculated using that product's experimentally determined critical power statistics as follows:

$$MCPR_{95/95}(i) = \mu_i + \kappa_i * \sigma_i \qquad \textbf{(Eq. 1)}$$

Where,

μ_i is the mean Experimental Critical Power Ratio (ECPR),

σ_i is the standard deviation of the ECPRs, and

κ_i is a statistical parameter chosen to provide 95% probability at 95% confidence (95/95) for the one-sided upper tolerance limit that depends on the number of samples (N_i) in the critical power database.

i is a fuel product line, such as GE14, GNF2, GNF3, OPTIMA2, and OPTIMA3.

The statistical parameter, κ_i, is calculated using formulas attributed to Mary Gibbons Natrella (1963) as recommended by the National Institute of Standards and Technology (NIST) in their Engineering Statistics Handbook (Reference 3). For a 95/95 probability/confidence level, the κ_i values are shown in the table below as a function of database size (N_i).

Table 2: Statistical Parameter, κ_i, at (95/95) for the One-Sided Upper Tolerance Limit

Database Size, N_i	κ_i
500	1.7625
750	1.7401
1000	1.7270
1250	1.7181
1500	1.7115
2000	1.7024

Assuming a typical critical power database of 1000 data points with no bias (i.e., $\mu_i = 1.0$), the following table illustrates representative $MCPR_{95/95}(i)$ values as a function of the database standard deviation.

Table 3: Representative $MCPR_{95/95}$ Values for N_i=1000

Standard Deviation, σ_i (%)	$MCPR_{95/95}$
2.0	1.03
2.5	1.04
3.0	1.05
3.5	1.06
4.0	1.07
4.5	1.08
5.0	1.09

For cores loaded with a single fuel product, the $SLMCPR_{95/95}$ is the $MCPR_{95/95}(i)$ value for that particular product line.

For cores with a mix of fuel products, the corresponding $SLMCPR_{95/95}$ is based on the largest (i.e., most limiting) of the $MCPR_{95/95}(i)$ values for the product lines that are fresh or once-burnt at the start of the cycle. The $MCPR_{95/95}(i)$ values for product lines that are twice-burnt or more at the start of the cycle may be ignored, as these higher exposure bundles operate with considerable MCPR margin relative to the more limiting fresh and once-burnt bundles.

The $SLMCPR_{95/95}$ will be reported in the TS to two digits past the decimal using standard rounding practices. The $SLMCPR_{95/95}$ also serves as the minimum value for the cycle-specific $MCPR_{99.9\%}$.

The revised method for calculation of the SL for Global Nuclear Fuel and Westinghouse fuel will continue to meet the regulatory definition of a safety limit and to reasonably protect the integrity of the fuel rod cladding against the uncontrolled release of radioactivity. The proposed change is also consistent with equivalent safety limits for other plant designs.

3.2. <u>Cycle-Specific OLMCPR</u>

The current $MCPR_{99.9\%}$ statistical limit calculation will continue to be performed using the approved methodology (e.g., References 4 through 7 or the plant-specific equivalents) and will be reported in the COLR. The OLMCPR limit in LCO 3.2.2 will continue to be determined based on the transient ΔCPR component and the cycle-specific $MCPR_{99.9\%}$ value. No changes to the method of determining the OLMCPR (i.e., the LCO 3.2.2 limit) are proposed, and the LCO limits and the $MCPR_{99.9\%}$ value will be reported in the COLR.

4. REGULATORY EVALUATION

Title 10 of the Code of Federal Regulations (10 CFR), Part 50, paragraph 50.36(c)(1)(i)(A) states:

> Safety limits for nuclear reactors are limits upon important process variables that are found to be necessary to reasonably protect the integrity of certain of the physical barriers that guard against the uncontrolled release of radioactivity. If any safety limit is exceeded, the reactor must be shut down. The licensee shall notify the Commission, review the matter, and record the results of the review, including the cause of the condition and the basis for corrective action taken to preclude recurrence. Operation must not be resumed until authorized by the Commission.

The purpose of the MCPR safety limit (SLMCPR) is to protect the physical barrier of the fuel cladding against the uncontrolled release of radioactivity. The SLMCPR is set such that no significant fuel damage is calculated to occur if the limit is met. Although it is recognized that the onset of transition boiling would not necessarily 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. Therefore, the proposed change to the SLMCPR will continue to protect the fuel cladding physical barrier from uncontrolled release of radioactivity.

10 CFR 50, Appendix A, General Design Criterion 10 states that specified acceptable fuel design limits will not be exceeded during steady state operation, normal operational transients, and Anticipated Operational Occurrences (AOOs). Most plants have a plant-specific design criterion similar to GDC 10. This design criterion will continue to be met. The OLMCPR, which is not affected by the proposed change, is established to ensure that no fuel damage results during normal operation, normal operational transients, and AOOs.

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 approval of the proposed change will not be inimical to the common defense and security or to the health and safety of the public.

5. REFERENCES

1. Letter from Brian R. Moore, Global Nuclear Fuel, to U.S. NRC, "Information Supporting TSTF-564 Safety Limit Minimum Critical Power Ratio," June 16, 2017, ADAMS Accession No. ML17167A108.

2. Letter from James A. Gresham, Westinghouse Electric Company, to U.S. NRC, "Submittal of 'Calculation for Technical Specification SLM CPR Values Applying to Westinghouse Fuel in Support of TSTF-564'," May 16, 2017, ADAMS Accession No. ML17142A319.

3. NIST/SEMATECH e-Handbook of Statistical Methods, http://www.itl.nist.gov/div898/handbook/, April 2012.

4. GE Nuclear Energy, "General Electric BWR Thermal Analysis Basis (GETAB): Data, Correlation and Design Application," NEDO-10958-A, January 1977, ADAMS Accession No. ML102290144.

5. GE Nuclear Energy, "Power Distribution Uncertainties for Safety Limit MCPR Evaluations," NEDC-32694P-A, August 1999, ADAMS Accession No. ML003740166.

6. GE Nuclear Energy, "Methodology and Uncertainties for Safety Limit MCPR Evaluations," NEDC-32601-P-A, August 1999, ADAMS Accession No. ML003740166.

7. ABB Combustion Engineering Nuclear Operations, "Reference Safety Report for Boiling Water Reactor Reload Fuel," CENPD-300-P-A, July 1996. ADAMS Accession No. ML110260388.

Enclosure 1

Model Application

[DATE] 10 CFR 50.90

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

DOCKET NO.PLANT NAME
50-[xxx]
SUBJECT: Application to Revise Technical Specifications to Adopt TSTF-564, "Safety Limit MCPR"

Pursuant to 10 CFR 50.90, [LICENSEE] is submitting a request for an amendment to the Technical Specifications (TS) for [PLANT NAME, UNIT NOS.].

[LICENSEE] requests adoption of TSTF-564, "Safety Limit MCPR," which is an approved change to the Improved Standard Technical Specifications (ISTS), into the [PLANT NAME, UNIT NOS] Technical Specifications (TS). The proposed amendment revises the Technical Specification (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.

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

No regulatory commitments are made in this submittal.

Approval of the proposed amendment is requested by [date]. Once approved, the amendment shall be implemented within [ ] days.

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

[In accordance with 10 CFR 50.30(b), a license amendment request must be executed in a signed original under oath or affirmation. This can be accomplished by attaching a notarized affidavit confirming the signature authority of the signatory, or by including the following statement in the cover letter: "I declare under penalty of perjury that the foregoing is true and correct. Executed on (date)." The alternative statement is pursuant to 28 USC 1746. It does not require notarization.]

If you should have any questions regarding this submittal, please contact [NAME, TELEPHONE NUMBER].

Sincerely,

[Name, Title]

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

{Attachments 2, 3, and 4 are not included in the model application and are to be provided by the licensee.}

cc: NRC Project Manager
NRC Regional Office
NRC Resident Inspector
State Contact

ATTACHMENT 1 - DESCRIPTION AND ASSESSMENT

1.0 DESCRIPTION

[LICENSEE] requests adoption of TSTF-564, "Safety Limit MCPR," which is an approved change to the Improved Standard Technical Specifications (ISTS), into the [PLANT NAME, UNIT NOS] Technical Specifications (TS). The proposed amendment revises the Technical Specification (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

[LICENSEE] has reviewed the safety evaluation for TSTF-564 provided to the Technical Specifications Task Force in a letter dated [DATE]. This review included a review of the NRC staff's evaluation, as well as the information provided in TSTF-564. [As described herein,] [LICENSEE] has concluded that the justifications presented in TSTF-564 and the safety evaluation prepared by the NRC staff are applicable to [PLANT, UNIT NOS.] and justify this amendment for the incorporation of the changes to the [PLANT] TS.

The [PLANT], Unit [1], reactor [is currently][will be] fueled with [TYPE] fuel bundles [describe multiple types of fuel bundles and which type limits the SLMCPR consistent with discussion in the traveler]. Consistent with Table 1 of TSTF-564, the proposed Safety Limit in [SL 2.1.1.2] is [1.07].

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

[[LICENSEE] is not proposing any variations from the TS changes described in the TSTF-564 or the applicable parts of the NRC staff's safety evaluation dated [DATE].] [[LICENSEE] is proposing the following variations from the TS changes described in the TSTF-564 or the applicable parts of the NRC staff's safety evaluation: describe the variations]

[The [PLANT] TS utilize different [numbering][and][titles] than the Standard Technical Specifications on which TSTF-564 was based. Specifically, [describe differences between the plant-specific TS numbering and/or titles and the TSTF-564 numbering and titles.] These differences are administrative and do not affect the applicability of TSTF-564 to the [PLANT] TS.]

[The [PLANT] TS contain requirements that differ from the Standard Technical Specifications on which TSTF-564 was based, such as reactor steam dome pressure or core flow in SL 2.1.1.2, or Applicability in TS 3.2.2, but these differences do not affect the applicability of the TSTF-564 justification. [For differences other than reactor steam dome pressure, core flow, or applicability, describe the differences and why TSTF-564 is still applicable.]

[The traveler and Safety Evaluation discuss the applicable regulatory requirements and guidance, including the 10 CFR 50, Appendix A, General Design Criteria (GDC). [PLANT] was not licensed to the 10 CFR 50, Appendix A, GDC. The [PLANT] equivalents of the referenced GDC are [reference including UFSAR location, if applicable]. [Discuss the equivalence of the referenced plant-specific requirements to the Appendix A GDC as related to the proposed change.] This difference does not alter the conclusion that the proposed change is applicable to [PLANT].]

3.0 REGULATORY ANALYSIS

3.1 No Significant Hazards Consideration Analysis

[LICENSEE] requests adoption of TSTF-564, "Safety Limit MCPR," which is an approved change to the Improved Standard Technical Specifications (ISTS), into the [PLANT NAME, UNIT NOS] Technical Specifications (TS). The proposed change revises the Technical Specification (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 transition boiling 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.

[LICENSEE] 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 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 boiling transition during an anticipated transient instead of the current limit based on ensuring that 99.9% of the fuel rods are not susceptible to transition boiling 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 does 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, [LICENSEE] 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

Technical Specifications Proposed Changes

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

2.1.1.1 With the reactor steam dome pressure < 785 psig or core flow < 10% rated core flow:

THERMAL POWER shall be ≤ 25% RTP.

2.1.1.2 With the reactor steam dome pressure ≥ 785 psig and core flow ≥ 10% rated core flow:

MCPR shall be ≥ [1.07] [for two recirculation loop operation or ≥ [1.08] 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.

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 ≥ 25% 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 < 25% 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 ≥ 25% RTP AND [24 hours thereafter OR In accordance with the Surveillance Frequency Control Program]

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 <u>AND</u> Once within 72 hours after each completion of SR 3.1.4.4

5.6 Reporting Requirements

5.6.3 CORE OPERATING LIMITS REPORT

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:

[The individual specifications that address core operating limits must be referenced here. The $MCPR_{99.9\%}$ value used to calculate the LCO 3.2.2, "MCPR," limit shall be specified in the COLR.]

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:

--------------------------------REVIEWER'S NOTE--
Licensees that have received prior NRC approval to relocate Topical Report revision numbers and dates to licensee control need only list the number and title of the Topical Report, and the COLR will contain the complete identification for each of the Technical Specification referenced Topical Reports used to prepare the COLR (i.e., report number, title, revision, date, and any supplements). See NRC ADAMS Accession No: ML110660285 for details.

[Identify the Topical Report(s) by number, title, date, and NRC staff approval document or identify the staff Safety Evaluation Report for a plant specific methodology by NRC letter and date.]

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

5.6.4 Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT

a. RCS pressure and temperature limits for heat up, cooldown, low temperature operation, criticality, and hydrostatic testing as well as heatup and cooldown rates shall be established and documented in the PTLR for the following:

[The individual specifications that address RCS pressure and temperature limits must be referenced here.]

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 significant 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 [both General Electric Company (GE) and Advanced Nuclear Fuel Corporation (ANF) 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.

----------------------------------- REVIEWER'S NOTE ------------------------------
In the Background and Applicable Safety Analysis sections, select the $SLMCPR_{95/95}$ discussion or the 99.9% of the fuel rods discussion as the applicable SL 2.1.1.2 basis.
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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 *are not susceptible to boiling transition*~~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 to a structurally weaker form. This weaker form may lose its integrity, resulting in an uncontrolled release of activity to the reactor coolant.

APPLICABLE SAFETY ANALYSES

The fuel cladding must not sustain damage as a result of normal operation and AOOs. *[The 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 limit.

2.1.1.1a Fuel Cladding Integrity [General Electric Company (GE) Fuel]

GE critical power correlations are applicable for all critical power calculations at pressures ≥ 785 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 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 25% RTP for reactor pressure < 785 psig is conservative.

2.1.1.1b Fuel Cladding Integrity [Advanced Nuclear Fuel Corporation (ANF) Fuel]

The use of the XN-3 correlation is valid for critical power calculations at pressures > 580 psig and bundle mass fluxes > 0.25 x 10^6 lb/hr-ft^2 (Ref. 3). For operation at low pressures or low flows, the fuel cladding integrity SL is established by a limiting condition on core THERMAL POWER, with the following basis:

Provided that the water level in the vessel downcomer is maintained above the top of the active fuel, natural circulation is sufficient to ensure a minimum bundle flow for all fuel assemblies that have a relatively high power and potentially can approach a critical heat flux condition. For the ANF 9x9 fuel design, the minimum bundle flow is > 30 x 10^3 lb/hr. For the

ANF 8x8 fuel design, the minimum bundle flow is > 28 x 10^3 lb/hr. For all designs, the coolant minimum bundle flow and maximum flow area are

BASES

APPLICABLE SAFETY ANALYSES (continued)

such that the mass flux is always > 0.25 x 10^6 lb/hr-ft^2. Full scale critical power tests taken at pressures down to 14.7 psia indicate that the fuel assembly critical power at 0.25 x 10^6 lb/hr-ft^2 is approximately 3.35 MWt. At 25% RTP, a bundle power of approximately 3.35 MWt corresponds to a bundle radial peaking factor of > 3.0, which is significantly higher than the expected peaking factor. Thus, a THERMAL POWER limit of 25% RTP for reactor pressures < 785 psig is conservative.

2.1.1.2a MCPR [GE *and Westinghouse* Fuel]

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 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 $MCPR_{95/95}$.]*

------------------------------------ Reviewer's Note -------------------------------
The $MCPR_{95/95}$ Values by Vendor and Fuel Product Type:

Vendor	***Fuel Type***	***$MCPR_{95/95}$***
Global Nuclear Fuel	*GE14*	*1.05*
Global Nuclear Fuel	*GNF2*	*1.07*
Global Nuclear Fuel	*GNF3*	*1.07*
Westinghouse	*Optima2*	*1.06*
Westinghouse	*Optima3*	*1.06*

[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.]

2.1.1.2b MCPR [ANF Fuel]

The MCPR SL ensures sufficient conservatism in the operating MCPR limit that, in the event of an AOO from the limiting condition of operation, at least 99.9% of the fuel rods in the core would be expected to avoid boiling transition. The margin between calculated boiling transition (i.e., MCPR = 1.00) and the MCPR SL is based on a detailed statistical

APPLICABLE SAFETY ANALYSES (continued)

procedure that considers the uncertainties in monitoring the core operating state. One specific uncertainty included in the SL is the uncertainty inherent in the XN-3 critical power correlation. Reference 3 describes the methodology used in determining the MCPR SL.

The XN-3 critical power correlation is based on a significant body of practical test data, providing a high degree of assurance that the critical power, as evaluated by the correlation, is within a small percentage of the actual critical power being estimated. As long as the core pressure and flow are within the range of validity of the XN-3 correlation, the assumed reactor conditions used in defining the SL introduce conservatism into the limit because bounding high radial power factors and bounding flat local peaking distributions are used to estimate the number of rods in boiling transition. Still further conservatism is induced by the tendency of the XN-3 correlation to overpredict the number of rods in boiling transition. These conservatisms and the inherent accuracy of the XN-3 correlation provide a reasonable degree of assurance that there would be no transition boiling in the core during sustained operation at the MCPR SL. If boiling transition were to occur, there is reason to believe that the integrity of the fuel would not be compromised. Significant test data accumulated by the NRC and private organizations indicate that the use of a boiling transition limitation to protect against cladding failure is a very conservative approach. Much of the data indicate that BWR fuel can survive for an extended period of time in an environment of boiling transition.

2.1.1.3 Reactor Vessel Water Level

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

BASES

SAFETY LIMITS	The reactor core SLs are established to protect the integrity of the fuel clad barrier to the release of radioactive materials to the environs. SL 2.1.1.1 and SL 2.1.1.2 ensure that the core operates within the fuel design criteria. SL 2.1.1.3 ensures that the reactor vessel water level is greater than the top of the active irradiated fuel in order to prevent elevated clad temperatures and resultant clad perforations.
APPLICABILITY	SLs 2.1.1.1, 2.1.1.2, and 2.1.1.3 are applicable in all MODES.
SAFETY LIMIT VIOLATIONS	Exceeding an SL may cause fuel damage and create a potential for radioactive releases in excess of 10 CFR 100, "Reactor Site Criteria," limits (Ref. 4). Therefore, it is required to insert all insertable control rods and restore compliance with the SLs within 2 hours. The 2 hour Completion Time ensures that the operators take prompt remedial action and also ensures that the probability of an accident occurring during this period is minimal.
REFERENCES	1. 10 CFR 50, Appendix A, GDC 10. 2. NEDE-24011-P-A (latest approved revision). 3. XN-NF524(A), Revision 1, November 1983. 4. 10 CFR 100.

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 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

---------------------------------- REVIEWER'S NOTE ------------------------------
Incorporate the MCPR$_{95/95}$ discussion if applicable.

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 *[SL]* 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.]

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 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.

BASES

APPLICABLE SAFETY ANALYSES (continued)

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 25% RTP and the previously mentioned bypass power level.

The MCPR satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

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 25% RTP, the reactor is operating at a minimum 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 $MCPR_{[99.9\%]}$ ~~SL~~ is not exceeded even if a limiting transient occurs.

Statistical analyses indicate that the nominal value of the initial MCPR expected at 25% 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 25% 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 < 25% 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.

BASES

ACTIONS (continued)

B.1

If the MCPR cannot be restored to within its required limits within the associated Completion Time, the plant must be brought to a MODE or other specified condition in which the LCO does not apply. To achieve this status, THERMAL POWER must be reduced to < 25% RTP within 4 hours. The allowed Completion Time is reasonable, based on operating experience, to reduce THERMAL POWER to < 25% RTP in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.2.2.1

The MCPR is required to be initially calculated within 12 hours after THERMAL POWER is ≥ 25% 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 ≥ 25% RTP is achieved is acceptable given the large inherent margin to operating limits at low power levels. [The 24 hour Frequency is based on both engineering judgment and recognition of the slowness of changes in power distribution during normal operation.

OR

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

----------------------------------REVIEWER'S NOTE----------------------------------
Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.
--]

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

BASES

SURVEILLANCE REQUIREMENTS (continued)

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. NEDO-24011-P-A, "General Electric Standard Application for Reactor Fuel" (latest approved version).
3. FSAR, Chapter [4].
4. FSAR, Chapter [6].
5. FSAR, Chapter [15].
6. [Plant specific single loop operation].
7. [Plant specific load line limit analysis].
8. [Plant specific Average Power Range Monitor, Rod Block Monitor and Technical Specification Improvements (ARTS) Program].
9. NEDO-30130-A, "Steady State Nuclear Methods," May 1985.
10. NEDO-24154, "Qualification of the One-Dimensional Core Transient Model for Boiling Water Reactors," October 1978.

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

2.1.1.1 With the reactor steam dome pressure < 785 psig or core flow < 10% rated core flow:

THERMAL POWER shall be ≤ 25% RTP.

2.1.1.2 With the reactor steam dome pressure ≥ 785 psig and core flow ≥ 10% rated core flow:

MCPR shall be ≥ [1.07] *[*for two recirculation loop operation or ≥ [1.08] 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.

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 ≥ 25% 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 < 25% 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 ≥ 25% RTP <u>AND</u> [24 hours thereafter <u>OR</u> In accordance with the Surveillance Frequency Control Program]

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:

[The individual specifications that address core operating limits must be referenced here. *The $MCPR_{99.9\%}$ value used to calculate the LCO 3.2.2, "MCPR," limit shall be specified in the COLR.*]

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:

--------------------------------REVIEWER'S NOTE--
Licensees that have received prior NRC approval to relocate Topical Report revision numbers and dates to licensee control need only list the number and title of the Topical Report, and the COLR will contain the complete identification for each of the Technical Specification referenced Topical Reports used to prepare the COLR (i.e., report number, title, revision, date, and any supplements). See NRC ADAMS Accession No: ML110660285 for details.

[Identify the Topical Report(s) by number, title, date, and NRC staff approval document or identify the staff Safety Evaluation Report for a plant specific methodology by NRC letter and date.]

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

5.6.4 Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT

a. RCS pressure and temperature limits for heatup, cooldown, low temperature operation, criticality, and hydrostatic testing as well as heatup and cooldown rates shall be established and documented in the PTLR for the following:

[The individual specifications that address RCS pressure and temperature limits must be referenced here.]

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 significant 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 [both General Electric Company (GE) and Advanced Nuclear Fuel Corporation (ANF) 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.

---------------------------------- REVIEWER'S NOTE -----------------------------
In the Background and Applicable Safety Analysis sections, select the SLMCPR$_{95/95}$ discussion or the 99.9% of the fuel rods discussion as the applicable SL 2.1.1.2 basis.
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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 *are not susceptible to boiling transition*~~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 to a structurally weaker form. This weaker form may lose its integrity, resulting in an uncontrolled release of activity to the reactor coolant.

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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 an MCPR limit is to be established, such that at least 99.9% of the fuel rods in the core would not be expected to experience the onset of transition boiling.*]*

The Reactor Protection System setpoints (LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation"), in combination with 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.

2.1.1.1a Fuel Cladding Integrity [General Electric Company (GE) Fuel]

GE critical power correlations are applicable for all critical power calculations at pressures ≥ 785 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 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 25% RTP for reactor pressure < 785 psig is conservative.

2.1.1.1b Fuel Cladding Integrity [Advanced Nuclear Fuel Corporation (ANF) Fuel]

The use of the XN-3 correlation is valid for critical power calculations at pressures > 580 psig and bundle mass fluxes > 0.25 x 10^6 lb/hr-ft^2 (Ref. 3). For operation at low pressures or low flows, the fuel cladding integrity SL is established by a limiting condition on core THERMAL POWER, with the following basis:

> Provided that the water level in the vessel downcomer is maintained above the top of the active fuel, natural circulation is sufficient to ensure a minimum bundle flow for all fuel assemblies that have a

relatively high power and potentially can approach a critical heat flux condition. For the ANF 9x9 fuel design, the minimum bundle flow

BASES

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is > 30 x 10^3 lb/hr. For the ANF 8x8 fuel design, the minimum bundle flow is > 28 x 10^3 lb/hr. For all designs, the coolant minimum bundle flow and maximum flow area are such that the mass flux is always > 0.25 x 10^6 lb/hr-ft^2. Full scale critical power tests taken at pressures down to 14.7 psia indicate that the fuel assembly critical power at 0.25 x 10^6 lb/hr-ft^2 is approximately 3.35 MWt. At 25% RTP, a bundle power of approximately 3.35 MWt corresponds to a bundle radial peaking factor of > 3.0, which is significantly higher than the expected peaking factor. Thus, a THERMAL POWER limit of 25% RTP for reactor pressures < 785 psig is conservative.

2.1.1.2a MCPR [GE *and Westinghouse* Fuel]

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 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 will not go into boiling transition, referred to as $MCPR_{95/95}$.]*

------------------------------------- Reviewer's Note -------------------------------

The $MCPR_{95/95}$ Values by Vendor and Fuel Product Type:

Vendor	***Fuel Type***	***$MCPR_{95/95}$***
Global Nuclear Fuel	*GE14*	*1.05*
Global Nuclear Fuel	*GNF2*	*1.07*
Global Nuclear Fuel	*GNF3*	*1.07*
Westinghouse	*Optima2*	*1.06*
Westinghouse	*Optima3*	*1.06*

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[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.]

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APPLICABLE SAFETY ANALYSES (continued)

2.1.1.2b MCPR [ANF Fuel]

The MCPR SL ensures sufficient conservatism in the operating MCPR limit that, in the event of an AOO from the limiting condition of operation, at least 99.9% of the fuel rods in the core would be expected to avoid boiling transition. The margin between calculated boiling transition (i.e., MCPR = 1.00) and the MCPR SL is based on a detailed statistical procedure that considers the uncertainties in monitoring the core operating state. One specific uncertainty included in the SL is the uncertainty inherent in the XN-3 critical power correlation. Reference 3 describes the methodology used in determining the MCPR SL.

The XN-3 critical power correlation is based on a significant body of practical test data, providing a high degree of assurance that the critical power, as evaluated by the correlation, is within a small percentage of the actual critical power being estimated. As long as the core pressure and flow are within the range of validity of the XN-3 correlation, the assumed reactor conditions used in defining the SL introduce conservatism into the limit because bounding high radial power factors and bounding flat local peaking distributions are used to estimate the number of rods in boiling transition. Still further conservatism is induced by the tendency of the XN-3 correlation to overpredict the number of rods in boiling transition. These conservatisms and the inherent accuracy of the XN-3 correlation provide a reasonable degree of assurance that there would be no transition boiling in the core during sustained operation at the MCPR SL. If boiling transition were to occur, there is reason to believe that the integrity of the fuel would not be compromised. Significant test data accumulated by the NRC and private organizations indicate that the use of a boiling transition limitation to protect against cladding failure is a very conservative approach. Much of the data indicate that BWR fuel can survive for an extended period of time in an environment of boiling transition.

2.1.1.3 Reactor Vessel Water Level

During MODES 1 and 2, the reactor vessel water level is required to be above the top of the active fuel to provide core cooling capability. With fuel in the reactor vessel during periods when the reactor is shut down, consideration must be given to water level requirements due to the effect of decay heat. If the water level should drop below the top of the active irradiated fuel during this period, the ability to remove decay heat is reduced. This reduction in cooling capability could lead to elevated

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cladding temperatures and clad perforation in the event that the water level becomes < 2/3 of the core height. The reactor vessel water level SL has been established at the top of the active irradiated fuel to provide a point that can be monitored and to also provide adequate margin for effective action.

SAFETY LIMITS

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

APPLICABILITY

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

SAFETY LIMIT VIOLATIONS

Exceeding an SL may cause fuel damage and create a potential for radioactive releases in excess of 10 CFR 100, "Reactor Site Criteria," limits (Ref. 4). 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, (latest approved revision).
3. XN-NF524(A), Revision 1, November 1983.
4. 10 CFR 100.

B 3.2 POWER DISTRIBUTIONS 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 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. 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

----------------------------------- REVIEWER'S NOTE -------------------------------
Incorporate the $MCPR_{95/95}$ discussion if applicable.

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 2, 3, 4, and 5. 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 *[SL]*$MCPR_{[99.9\%]}$ ~~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.]

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 state ($MCPR_f$ and $MCPR_p$, respectively) to ensure adherence to fuel design limits during the worst transient that occurs with moderate frequency (Refs. 3, 4, and 5). Flow dependent MCPR limits are determined by steady state thermal hydraulic methods using the three dimensional BWR simulator code (Ref. 6) and the multichannel thermal hydraulic code (Ref. 7). $MCPR_f$

APPLICABLE SAFETY ANALYSES (continued)

curves are provided based on the maximum credible flow runout transient for Loop Manual and Non Loop Manual operation. The result of a single failure or single operator error during Loop Manual operation is the runout of only one loop because both recirculation loops are under independent control. Non Loop Manual operational modes allow simultaneous runout of both loops because a single controller regulates core flow.

Power dependent MCPR limits ($MCPR_p$) are determined by *approved transient analysis models*~~the three dimensional BWR simulator code and the one dimensional transient code~~ (Ref. 8). 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 10 CFR 50.36(c)(2)(ii).

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 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 $MCPR_{[99.9\%]}$ ~~SL~~ is not exceeded even if a limiting transient occurs.

Statistical analyses documented in Reference 9 indicate that the nominal value of the initial MCPR expected at 25% 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 (Ref. 5) 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 25% 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 (IRM) provides rapid scram initiation for any significant power increase transient, which effectively eliminates any MCPR compliance concern. Therefore, at THERMAL POWER levels < 25% 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 < 25% RTP within 4 hours. The allowed Completion Time is reasonable, based on operating experience, to reduce THERMAL POWER to < 25% RTP in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.2.2.1

The MCPR is required to be initially calculated within 12 hours after THERMAL POWER is ≥ 25% 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 ≥ 25% RTP is acceptable given the large inherent margin to operating limits at low power levels. [The 24 hour Frequency is based on both engineering judgment and recognition of the slowness of changes in power distribution during normal operation.

OR

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

-----------------------------------REVIEWER'S NOTE-----------------------------------
Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.
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BASES

REFERENCES

1. NUREG-0562, June 1979.

2. [Plant specific current cycle safety analysis].

3. FSAR, [Appendix 15B].

4. FSAR, [Appendix 15C].

5. FSAR, [Appendix 15D].

6. XN-NF-80-19(P)(A), "Exxon Nuclear Methodology for Boiling Water Reactors, Neutronics Methods for Design and Analysis," Volume 1 (as supplemented).

7. XN-NF-80-19(P)(A), "Exxon Nuclear Methodology for Boiling Water Reactors, THERMEX Thermal Limits Methodology Summary Description," Volume 3, Revision 2, January 1987.

8. XN-NF-79-71(P), "Exxon Nuclear Plant Methodology for Boiling Water Reactors," Revision 2, November 1981.

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