

ATTACHMENT A

NIAGARA MOHAWK POWER CORPORATION
LICENSE NO. NPF-69
DOCKET NO. 50-410

Proposed Change to Technical Specifications

Replace existing pages iii, 2-1 and 3/4 4-1 with the attached revised pages. Replace existing Bases pages B2-1, B2-2, B2-3 and B2-4 with the attached revised pages. These pages have been retyped in their entirety with marginal markings to indicate changes to the text.

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2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

THERMAL POWER, Low Pressure or Low Flow

2.1.1 THERMAL POWER shall not exceed 25% of RATED THERMAL POWER with the reactor vessel steam dome pressure less than 785 psig or core flow less than 10% of rated flow.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With THERMAL POWER exceeding 25% of RATED THERMAL POWER and the reactor vessel steam dome pressure less than 785 psig or core flow less than 10% of rated flow, be in at least HOT SHUTDOWN within 2 hours and comply with the requirements of Specification 6.7.

THERMAL POWER, High Pressure and High Flow

2.1.2 The MINIMUM CRITICAL POWER RATIO (MCPR) shall not be less than 1.09 with two recirculation loop operation and shall not be less than 1.10 with single recirculation loop operation with the reactor vessel steam dome pressure greater than 785 psig and core flow greater than 10% of rated flow.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With MCPR less than 1.09, with two recirculation loop operation or less than 1.10 with single loop operation, the reactor vessel steam dome pressure greater than 785 psig, and core flow greater than 10% of rated flow, be in at least HOT SHUTDOWN within 2 hours and comply with the requirements of Specification 6.7.

REACTOR COOLANT SYSTEM PRESSURE

2.1.3 The reactor coolant system pressure, as measured in the reactor vessel steam dome, shall not exceed 1325 psig.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3, and 4.

ACTION:

With the reactor coolant system pressure as measured in the reactor vessel steam dome above 1325 psig, be in at least HOT SHUTDOWN with reactor coolant system pressure less than or equal to 1325 psig within 2 hours and comply with the requirements of Specification 6.7.

REACTOR VESSEL WATER LEVEL

2.1.4 The reactor vessel water level shall be above the top of the active irradiated fuel.

3/4.4 REACTOR COOLANT SYSTEM

3/4.4.1 RECIRCULATION SYSTEM

RECIRCULATION LOOPS

LIMITING CONDITIONS FOR OPERATION

3.4.1.1 Two reactor coolant system recirculation loops shall be in operation with:

- a. Total core flow greater than or equal to 45% of rated core flow, or
- b. THERMAL POWER within the unrestricted zone of Figure 3.4.1.1-1.

APPLICABILITY: OPERATIONAL CONDITIONS 1* AND 2*.

ACTION:

- a. With one reactor coolant system recirculation loop not in operation:
 - 1. Within four hours:
 - a) Place the recirculation flow control system in the Loop Manual (Position Control) mode, and
 - b) Reduce THERMAL POWER to $\leq 70\%$ of RATED THERMAL POWER, and,
 - c) Increase the MINIMUM CRITICAL POWER RATIO (MCPR) Safety Limit by 0.01 to 1.10 per Specification 2.1.2, and,
 - d) Reduce the Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) limit per Specification 3.2.1, and,
 - e) Reduce the Average Power Range Monitor (APRM) Scram and Rod Block and Rod Block Monitor Trip Setpoints and Allowable Values to those applicable for single recirculation loop operation per Specifications 2.2.1, 3.2.2 and 3.3.6.
 - f) Reduce the volumetric drive flow rate of the operating recirculation loop to $\leq 41,800^{**}$ gpm.

* See Special Test Exception 3.10.4.

** This value represents the volumetric recirculation loop drive flow which produces 100% core flow at 100% THERMAL POWER.

2.1 BASES FOR SAFETY LIMITS

2.1.0 INTRODUCTION

The fuel cladding, reactor pressure vessel, and primary system piping are the principal barriers to the release of radioactive materials to the environs. Safety Limits are established to protect the integrity of these barriers during normal plant operations and anticipated transients. The fuel cladding integrity Safety Limit is set so that no fuel damage is calculated to occur if the limit is not violated. Because fuel damage is not directly observable, a step-back approach is used to establish a Safety Limit so that the MCPR is not less than 1.09 for two recirculation loop operation and 1.10 for single recirculation loop operation. MCPR greater than 1.09 for two recirculation loop operation and 1.10 for single recirculation loop operation 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 that occur from reactor operation significantly above design conditions and the Limiting Safety System Settings. Although 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 Safety Limit is defined with a margin to the conditions that would produce onset of transition boiling, MCPR of 1.0. These conditions represent a significant departure from the condition intended by design for planned operation.

2.1.1 THERMAL POWER, Low Pressure or Low Flow

The use of critical power correlations is not valid for all critical power calculations performed at reduced pressures below 785 psig or core flows less than 10% of rated flow. Therefore, the fuel cladding integrity Safety Limit is established by other means. This is done by establishing a limiting condition on core THERMAL POWER with the following basis. 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 greater than 4.5 psi. Analyses 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 greater than 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 of more than 50% of RATED THERMAL POWER. Thus, a THERMAL POWER limit of 25% of RATED THERMAL POWER for reactor pressure below 785 psig is conservative.

BASES FOR SAFETY LIMITS

2.1.2 THERMAL POWER, High Pressure and High Flow

The fuel cladding integrity Safety Limit is set so 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 resulting in a departure from nucleate boiling have been used to mark the beginning of the region in which fuel damage could occur. Although it is recognized that a departure from nucleate 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. 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 Safety Limit is defined as the CPR 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 Safety Limit MCPR is determined using a statistical model that combines all of the uncertainties in operating parameters and the procedures used to calculate critical power. The probability of the occurrence of boiling transition is determined using an approved critical power correlation. The critical power correlation is valid over the range of conditions used in the tests of the data used to develop the correlation. Details of the fuel cladding integrity Safety Limit calculation are given in Reference 1. Reference 1 also includes a tabulation of the uncertainties used in the determination of the Safety Limit MCPR. The plant specific values of the parameters used in the Safety Limit MCPR statistical analysis are found in the cycle specific analysis.

References:

1. General Electric Standard Application for Reactor Fuel, NEDE-24011-P-A (latest approved revision).

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ATTACHMENT B

NIAGARA MOHAWK POWER CORPORATION LICENSE NO. NPF-69 DOCKET NO. 50-410

Supporting Information and No Significant Hazards Consideration Analysis

INTRODUCTION

On May 24, 1996, GE notified the NRC of a reportable condition involving the generic safety limit calculational methodology. GE determined that the generic Safety Limit MCPR might be non-conservative when applied to some cycle specific core and fuel designs. As a result of this error, GE performed a cycle specific Safety Limit calculation. NMPC submitted NMP2 LER 96-06 on June 3, 1996, and provided additional information regarding the impact of the nonconservative values. NMPC concluded that neither the Safety Limit nor the Operating Limit would have been exceeded for any analyzed plant transient, based on the increased Safety Limit value and the core performance up to that point in the operating cycle. The Supplemental Reload Licensing Report, USAR and COLR have been revised to include the correct Safety Limit Minimum Critical Power Ratio (MCPR) for the current operating cycle. This change to the TS completes the corrective actions described in LER 96-06. NMPC and GE have completed an analysis to determine the necessary Safety Limit MCPR for the upcoming operating cycle (Cycle 7). This analysis shows that the required Safety Limit MCPR will decrease from the current cycle's value based on the cycle specific Safety Limit MCPR calculation. As a result, this TS change revises the Safety Limit MCPR from 1.07 to 1.09 for two recirculation loop operation and from 1.08 to 1.10 for single loop operation to account for these changes.

A cycle specific Safety Limit calculation will be performed for future core reloads using cycle specific core loading patterns and power distributions, pending a long term solution. Thus, each core reload will be evaluated to ensure that the Safety Limit MCPR conservatively bounds the respective reload and operating cycle. NMPC will utilize its administrative control process to ensure that a cycle specific analysis has been completed prior to each startup from a refueling outage. This practice will be maintained as long as cycle specific calculations are required.

In addition, Bases Section 2.1 is revised for consistency on page B2-1 and to delete some of the detailed information and refer instead to General Electric Standard Application for Reactor Fuel (GESTAR II), NEDE-24011-P-A and to the cycle specific analysis (Supplemental Reload Licensing Report). This level of detail is consistent with that in the Improved Standard Technical Specifications (NUREG-1434). Some of the wording has been revised to more accurately reflect current practices.

A footnote on page 3/4 4-1 is also deleted. This footnote only applied to the first operating cycle and is therefore no longer applicable or necessary.

ANALYSIS

The reload analyses and evaluations are performed based on General Electric Standard Application for Reactor Fuel, NEDE 24011-P-A-13 and NEDE 24011-P-A-13-US (GESTAR II, latest approved revisions). This document describes the fuel licensing acceptance criteria; the fuel thermal-mechanical, nuclear, and thermal-hydraulic analyses bases; and the safety analysis methodology. The evaluations included transients and accidents likely to limit operation because of MCPR considerations, overpressurization events, loss of coolant accident, and stability analysis.

Core operating limits are established to ensure that the Safety Limits are not exceeded, the limits of 10CFR50.46 are satisfied, and that other fuel licensing acceptance criteria specified in GESTAR II are met. The fuel cladding is one of the principal barriers to the release of radioactive materials to the environment. Safety Limits are established to protect the integrity of this barrier during normal plant operation and anticipated transients. The Safety Limit MCPR is applied to ensure fuel cladding integrity is not lost as a result of over-heating. Compliance with the Safety Limit MCPR will assure that 99.9 percent of the fuel rods would not be expected to experience transition boiling during the most limiting anticipated operational occurrence.

Consistent with the basis of GESTAR II, the limiting transient events are reanalyzed for each reload. These analyses include those transients which could result in a significant reduction in MCPR. As a result of GE's work on cycle specific safety limits, a nonconservative Safety Limit MCPR was identified and a Part 21 notification was made to the NRC. As documented in the Supplemental Reload Licensing Report for Nine Mile Point Nuclear Power Station Unit 2, GENE 24A5174, Rev. 1, June 1996, the Safety Limit MCPR was increased from 1.07 for two recirculation loop operation and 1.08 for single loop operation to 1.10 and 1.12 respectively. This change was implemented in the COLR via an increase in the Operating Limit MCPR to maintain the existing margin of safety. The COLR contains the cycle specific parameters that were removed from the TS in TS Amendment No. 17, dated June 19, 1990.

The upcoming operating cycle (Cycle 7) has been analyzed in accordance with the NRC approved methods (described in GESTAR II) and subsequent commitments made in GE's letter to the NRC dated May 24, 1996, regarding the 10CFR Part 21 reportable condition relating to the Safety Limit MCPR calculation. The methodology used for calculating the cycle specific Safety Limit MCPR is as described in Amendment 25 to GESTAR II, which was submitted by GE to the NRC on December 13, 1996. Additional information can be found in Attachment D. As a result of the cycle specific calculation, the Safety Limit MCPR for Cycle 7 will be 1.09 for two recirculation loop operation and 1.10 for single loop operation. Thus, this TS change revises the Safety Limit MCPR to 1.09 for two recirculation loop operation and 1.1 for single loop operation to account for these changes.

The deletion of the TS Bases tables will not affect the methodology used to calculate the fuel cladding safety limit. As described in the proposed wording in the Bases, GESTAR II includes the appropriate uncertainties. The nominal values are replaced with cycle specific

values, which are contained in the cycle specific analysis (Supplemental Reload Licensing Report). Thus, the NRC approved methods will still be used to calculate the safety limit using the appropriate input.

The deletion of the footnote is an administrative change only, and has no impact on plant operation. This footnote only applied to the first operating cycle and is therefore no longer necessary.

CONCLUSIONS

A cycle specific safety limit calculation was performed for NMP2. The methodologies used to determine the limits were based on those approved by the NRC in GESTAR II and subsequent commitments made in GE's letter to the NRC dated May 24, 1996, regarding the 10CFR Part 21 reportable condition. These methodologies will continue to assure that the fuel licensing acceptance criteria are met. Based on the evaluations and analyses described, NMP2 can be safely operated with the revised Safety Limit MCPR.

NO SIGNIFICANT HAZARDS CONSIDERATION ANALYSIS

10CFR50.91 requires that at the time a licensee requests an amendment, it must provide to the Commission its analysis, using the standards in 10CFR50.92 concerning the issue of no significant hazards consideration. Therefore, in accordance with 10CFR50.91, the following analysis has been performed with respect to the requested change.

The operation of Nine Mile Point Unit 2, in accordance with the proposed amendment, will not involve a significant increase in the probability or consequences of an accident previously evaluated.

The derivation of the revised Safety Limit MCPR was performed using the NRC approved methodology in GESTAR II. The Safety Limit MCPR is a TS numerical value that cannot initiate an event. Maintaining compliance with this limit will assure that 99.9 percent of the fuel rods will not experience transition boiling during transient events. The deletion of the footnote that is no longer necessary and the revision to the Bases information are administrative only. The proposed change does not modify any of the accident initiators described in the USAR. No equipment malfunctions or procedural errors are created as a result of this change, therefore, no accidents are affected by it. The change does not adversely impact the integrity of the fuel cladding, which is the first barrier to the release of radioactivity to the environment. The change does not affect the operation of any systems necessary to mitigate the radiological consequences of an accident or to safely shutdown the plant. Therefore, this change will not involve a significant increase in the probability or consequences of an accident previously evaluated.

The operation of Nine Mile Point Unit 2, in accordance with the proposed amendment, will not create the possibility of a new or different kind of accident from any accident previously evaluated.

The Safety Limit MCPR is a TS numerical value designed to prevent fuel damage from transition boiling. It cannot create the possibility of a transient or accident. The deletion of the footnote that is no longer necessary and the revision to the Bases information are administrative only. The proposed change does not directly impact the operation of any

systems or equipment important to safety. The analyses show that all fuel licensing acceptance criteria are met. The fuel cladding, reactor vessel, and reactor coolant system integrity will be maintained. Therefore, this change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

The operation of Nine Mile Point Unit 2, in accordance with the proposed amendment, will not involve a significant reduction in a margin of safety.

The Safety Limit MCPR calculation was performed using the NRC approved methodology in GESTAR II. Analyses of limiting USAR transients establish Operating Limit MCPR values that ensure that the Safety Limit MCPR is not violated. The revised cycle specific Safety Limit MCPR preserves the existing margin of safety and will continue to assure that 99.9 percent of the fuel rods will not experience transition boiling during transient events. The deletion of the footnote that is no longer necessary and the revision to the Bases information are administrative only. Thus, the margin of safety to fuel cladding failure due to insufficient cladding heat transfer during transient events is not reduced. Therefore, this change will not involve a significant reduction in a margin of safety.

ATTACHMENT C

NIAGARA MOHAWK POWER CORPORATION
LICENSE NO. NPF-69
DOCKET NO. 50-410

Marked-Up Copy of Proposed Change to Current Technical Specifications

The current version of NMP2 Technical Specifications pages iii, 2-1, 3/4 4-1, B2-1, B2-2, B2-3 and B2-4 have been hand marked-up to reflect the proposed changes.

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2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

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2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

THERMAL POWER, Low Pressure or Low Flow

2.1.1 THERMAL POWER shall not exceed 25% of RATED THERMAL POWER with the reactor vessel steam dome pressure less than 785 psig or core flow less than 10% of rated flow.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With THERMAL POWER exceeding 25% of RATED THERMAL POWER and the reactor vessel steam dome pressure less than 785 psig or core flow less than 10% of rated flow, be in at least HOT SHUTDOWN within 2 hours and comply with the requirements of Specification 6.7.

THERMAL POWER, High Pressure and High Flow

2.1.2 The MINIMUM CRITICAL POWER RATIO (MCPR) shall not be less than ^{1.10}~~1.07~~ with two recirculation loop operation and shall not be less than ^{1.09}~~1.08~~ with single recirculation loop operation with the reactor vessel steam dome pressure greater than 785 psig and core flow greater than 10% of rated flow.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With MCPR less than ^{1.09}~~1.07~~, with two recirculation loop operation or less than ^{1.10}~~1.08~~ with single loop operation, the reactor vessel steam dome pressure greater than 785 psig, and core flow greater than 10% of rated flow, be in at least HOT SHUTDOWN within 2 hours and comply with the requirements of Specification 6.7.

REACTOR COOLANT SYSTEM PRESSURE

2.1.3 The reactor coolant system pressure, as measured in the reactor vessel steam dome, shall not exceed 1325 psig.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3, and 4.

ACTION:

With the reactor coolant system pressure as measured in the reactor vessel steam dome above 1325 psig, be in at least HOT SHUTDOWN with reactor coolant system pressure less than or equal to 1325 psig within 2 hours and comply with the requirements of Specification 6.7.

REACTOR VESSEL WATER LEVEL

2.1.4 The reactor vessel water level shall be above the top of the active irradiated fuel.

3/4.4 REACTOR COOLANT SYSTEM

3/4.4.1 *RECIRCULATION SYSTEM

RECIRCULATION LOOPS

LIMITING CONDITIONS FOR OPERATION

3.4.1.1 Two reactor coolant system recirculation loops shall be in operation with:

- a. Total core flow greater than or equal to 45% of rated core flow, or
- b. THERMAL POWER within the unrestricted zone of Figure 3.4.1.1-1.

APPLICABILITY: OPERATIONAL CONDITIONS 1* and 2*.

ACTION:

- a. With one reactor coolant system recirculation loop not in operation:
 1. Within four hours:
 - a) Place the recirculation flow control system in the Loop Manual (Position Control) mode, and
 - b) Reduce THERMAL POWER to $\leq 70\%$ of RATED THERMAL POWER, and,
 - c) Increase the MINIMUM CRITICAL POWER RATIO (MCPR) Safety Limit by 0.01 to ~~1.08***~~ ^{1.10} per Specification 2.1.2, and,
 - d) Reduce the Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) limit per Specification 3.2.1, and,
 - e) Reduce the Average Power Range Monitor (APRM) Scram and Rod Block and Rod Block Monitor Trip Setpoints and Allowable Values to those applicable for single recirculation loop operation per Specifications 2.2.1, 3.2.2 and 3.3.6.
 - f) Reduce the volumetric drive flow rate of the operating recirculation loop to $\leq 41,800^{**}$ gpm.

* See Special Test Exception 3.10.4.

** This value represents the volumetric recirculation loop drive flow which produces 100% core flow at 100% THERMAL POWER.

~~*** The MCPR Safety Limit of 1.07 will be used through the first operating cycle.~~

2.1 BASES FOR SAFETY LIMITS

2.1.0 INTRODUCTION

The fuel cladding, reactor pressure vessel, and primary system piping are the principal barriers to the release of radioactive materials to the environs. Safety Limits are established to protect the integrity of these barriers during normal plant operations and anticipated transients. The fuel cladding integrity Safety Limit is set so that no fuel damage is calculated to occur if the limit is not violated. Because fuel damage is not directly observable, a step-back approach is used to establish a Safety Limit so that the MCPR is not less than ~~1.07~~ for two recirculation loop operation and ~~1.08~~ for single recirculation loop operation. MCPR greater than ~~1.07~~ for two recirculation loop operation and ~~1.08~~ for single recirculation loop operation 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 that occur from reactor operation significantly above design conditions and the Limiting Safety System Settings. Although 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 Safety Limit is defined with a margin to the conditions that would produce onset of transition boiling, MCPR of 1.0. These conditions represent a significant departure from the condition intended by design for planned operation.

2.1.1 THERMAL POWER, Low Pressure or Low Flow

The use of critical power correlations is not valid for all critical power calculations performed at reduced pressures below 785 psig or core flows less than 10% of rated flow. Therefore, the fuel cladding integrity Safety Limit is established by other means. This is done by establishing a limiting condition on core THERMAL POWER with the following basis. 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 greater than 4.5 psi. Analyses 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 greater than 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 of more than 50% of RATED THERMAL POWER. Thus, a THERMAL POWER limit of 25% of RATED THERMAL POWER for reactor pressure below 785 psig is conservative.

BASES FOR SAFETY LIMITS

2.1.2 THERMAL POWER, High Pressure and High Flow

The fuel cladding integrity Safety Limit is set so 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 resulting in a departure from nucleate boiling have been used to mark the beginning of the region in which fuel damage could occur. Although it is recognized that a departure from nucleate 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. 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 Safety Limit is defined as the CPR 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 Safety Limit MCPR is determined using a statistical model that combines all of the uncertainties in operating parameters and the procedures used to calculate critical power. The probability of the occurrence of boiling transition is determined using an approved critical power correlation. The critical power correlation is valid over the range of conditions used in the tests of the data used to develop the correlation.

The required input to the statistical model are the uncertainties listed in Bases Table B2.1.2-1 and the nominal values of the core parameters listed in Bases Table B2.1.2-2. The bases for the uncertainties in the core parameters and the basis for the uncertainty in the critical power correlation are given in Reference 1. The power distribution is based on a typical 764 assembly core in which the rod pattern was arbitrarily chosen to produce a skewed power distribution having the greatest number of assemblies at the highest power levels. The worst distribution during any fuel cycle would not be as severe as the distribution used in the analysis.

References:

1. General Electric Standard Application for Reactor Fuel, NEDE-24011-P-A (latest approved revision).

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Details of the fuel cladding integrity Safety Limit calculation are given in Reference 1. Reference 1 also includes a tabulation of the uncertainties used in the determination of the Safety Limit MCPR. The plant specific values of the parameters used in the Safety Limit MCPR statistical analysis are found in the cycle specific analysis.

DASES TABLE B2.1.2-1
UNCERTAINTIES USED IN THE DETERMINATION
OF THE FUEL CLADDING SAFETY LIMIT*

<u>QUANTITY</u>	<u>STANDARD DEVIATION (% OF POINT)</u>
Feedwater Flow	1.76
Feedwater Temperature	0.76
Reactor Pressure	0.5
Core Inlet Temperature	0.2
Core Total Flow	2.5
Channel Flow Area	3.0
Friction Factor Multiplier	10.0
Channel Friction Factor Multiplier	5.0
TIP Readings	8.7
R Factor	1.6
Critical Power	3.0

* The uncertainty analysis used to establish the corewide Safety Limit MCPR is based on the assumption of quadrant power symmetry for the reactor core. The values herein apply to both two recirculation loop operation and single recirculation loop operation.

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BASES TABLE B2.1.2-2

NOMINAL VALUES OF PARAMETERS* USED IN
THE STATISTICAL ANALYSIS OF FUEL CLADDING INTEGRITY SAFETY LIMIT**

<u>PARAMETER</u>	<u>VALUE</u>
THERMAL POWER	3293 MW
Core Flow	102.5 Mlb/hr
Dome Pressure	1005 psig
Bundle Enrichment	3.0 Wt % U-235
R-Factor:	
0 - 10 GWD/ST	0.915
10 - 15 GWD/ST	0.954
> 15 GWD/ST	0.954

* The values in this table are for a representative plant.

** The Statistical analysis has been evaluated and shown to be valid at 3467 MW(t) with GE fuel (References: NEDC 31984P, "Generic Evaluations of GE BWR Power Uprate", Volume 1; NEDC-24011-P-A, GESTAR II; and NEDC-31152P, "GE Fuel Bundle Designs").

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ATTACHMENT F

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Non-Proprietary Version of Information Regarding GE's
Methodology for Cycle Specific Safety Limit MCPR Calculations

ATTACHED

References

- [1] *General Electric BWR Thermal Analysis Basis (GETAB): Data, Correlation and Design Application*, NEDO-10958-A, January 1977.
- [2] *General Electric Standard Application for Reactor Fuel (GESTAR II)*, NEDE-24011-P-A-11, November 1995.
- [3] *General Electric Standard Application for Reactor Fuel (GESTAR II)*, NEDE-24011-P-A-13, August 1996.
- [4] *General Electric Fuel Bundle Designs*, NEDE-31152-P, Revision 6, April 1997.
- [5] *Methodology and Uncertainties for Safety Limit MCPR Evaluations*, NEDC-32601P, Class III, December 1996.
- [6] *R-Factor Calculation Method for GE11, GE12 and GE13 Fuel*, NEDC-32505P, November 1995.

Comparison of NMP-2 Cycle 7 SLMCPR versus the Cycle 6 SLMCPR

Table 1 summarizes the relevant input parameters and results of the SLMCPR determination for both the NMP-2 Cycle 6 and Cycle 7 cores. Both cycle-specific evaluations were performed using the methods described in GETAB [1]. The evaluations yield different calculated SLMCPR values because the inputs that are used are different. The quantities that have been shown to have some impact on the determination of the safety limit MCPR (SLMCPR) are provided. Much of this information is redundant but is provided in this case because it has been provided previously to the NRC to assist them in understanding the differences between plant/cycle specific SLMCPR evaluations. [[]]

Prior to 1996, GESTAR II [2] stipulated that the SLMCPR analysis for a new fuel design be performed for a large high power density plant assuming a bounding equilibrium core. The GE11 product line generic SLMCPR value of 1.07 was determined according to this specification. Later revisions to GESTAR II [3] have been submitted to the NRC to describe how plant/cycle specific SLMCPR analyses are used to confirm the calculated SLMCPR value on a plant/cycle specific basis using the uncertainties defined in Reference [4].

In comparing the NMP-2 Cycle 6 and Cycle 7 SLMCPR values it is important to note that although GE11 dominates both cores, neither core is an equilibrium core. C6 was a mixed core with GE11 and GE9 fuel. In both cores, the fresh bundle enrichment increases relative to the previous cycle fresh fuel. The freshest fuel is the latest batch of GE11 that comprises [[]] of the bundles in the core. Also, this fresh batch of GE11 has the highest enrichment [[]], as compared to a core average enrichment of [[]], as shown in Table 1. By way of comparison, the Cycle 6 core has a smaller batch of fresh GE11 [[]] and a lower core average enrichment of [[]]. Higher enrichment in the fresh fuel for the NMP-2 Cycle 7 core (compared to the rest of the core) produces higher power in the fresh bundles relative to the rest of the core. These enrichment differences result in the GE11 fresh fuel producing a higher relative share of the number of fuel rods calculated to be susceptible to boiling transition (NRSBT). [[]]

[[]] The NMP-2 Cycle 7 core has a somewhat flatter core MCPR distribution than the Cycle 6 core and the fresh Cycle 7 bundle R-factor distributions are somewhat more peaked than the fresh Cycle 6 bundles.

[[]]

[[]]

[[]] one is led to the conclusion that the core MCPR distribution for NMP-2 Cycle 7 is slightly flatter than the distribution evaluated for Cycle 6.

The uncontrolled bundle pin-by-pin power distributions were compared between the NMP-2 Cycle 6 and Cycle 7 bundles used for the SLMCPR analyses. Pin-by-pin power distributions are characterized in terms of R-factors using the methodology defined in Reference [6]. For the NMP-2 Cycle 7 bundles, there is a somewhat more peaked distribution of uncontrolled R-factors for the highest power rods in each bundle, which in the calculation are the rods most likely to be susceptible to boiling transition. This fact is suggested by a graphical comparison of the bundle R-factor distributions but is difficult to quantify graphically since the relative flatnesses are similar and the rods that have an R-factor closer to the R-factor for the lead rod are statistically worth much more than those that have R-factors that are further away. [[]]

The flatness of the pin R-factor distribution within a particular bundle is characterized [[]]

[[]] Thus this supports the conclusion that the lower SLMCPR value for NMP-2 Cycle 7 is at least in part due to the more peaked bundle R-factors relative to those used for the Cycle 6 SLMCPR evaluation.

Table 1 Comparison of NMP-2 Cycle 6 and Cycle 7 Core and Bundle Quantities that Impact the SLMCPR [[]]

Summary

The calculated nominal 1.09 Monte Carlo SLMCPR for NMP-2 Cycle 7 is consistent with what one would expect [[]] the 1.09 SLMCPR value is appropriate.

Various quantities [[]] have been used over the last year to compare quantities that impact the calculated SLMCPR value. These other quantities have been provided to the NRC previously for other plant/cycle specific analyses using a format such as that given in Table 1. These other quantities have also been compared for this core/cycle [[]]. The key parameters in Table 1 support the conclusions that the NMP-2 Cycle 7 core/cycle has a slightly flatter core MCPR distribution and somewhat more peaked in-bundle power distributions [[]] than what was used to perform the Cycle 6 SLMCPR evaluation. The more peaked bundle R-factor distribution slightly outweighs the flatter core MCPR distribution, resulting in the lower calculated Cycle 7 SLMCPR relative to the Cycle 6 SLMCPR evaluation.

Based on all of the facts, observations and arguments presented above, it is concluded that the calculated SLMCPR value of 1.09 for the NMP-2 Cycle 7 core is appropriate. It is reasonable that this value is 0.01 lower than the 1.10 value calculated for Cycle 6.

For single loop operations (SLO) the safety limit MCPR is 0.01 greater than the two-loop value [[]]