This letter forwards proprietary information in accordance with 10 CFR 2.390. The balance of this letter may be considered non-proprietary upon removal of Attachment (7).

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Constellation Energy
Nine Mile Point Nuclear Station

March 30, 2007

U. S. Nuclear Regulatory Commission
Washington, DC 20555-0001

ATTENTION: Document Control Desk

SUBJECT: Nine Mile Point Nuclear Station
Unit No. 2; Docket No. 50-410

License Amendment Request Pursuant to 10 CFR 50.90:
Implementation of ARTS/MELLLA

REFERENCE: (a) Letter from B. R. Sylvia (Niagara Mohawk Power Corporation) to Document Control Desk (NRC), dated October 31, 1997, License Amendment Request to Use NUMAC Power Range Neutron Monitor System (PRNM)

Pursuant to 10 CFR 50.90, Nine Mile Point Nuclear Station, LLC, (NMPNS) hereby requests an amendment to Nine Mile Point Unit 2 (NMP2) Renewed Operating License NPF-69. The proposed amendment would reflect an expanded operating domain resulting from the implementation of Average Power Range Monitor/Rod Block Monitor/Technical Specifications/Maximum Extended Load Line Limit Analysis (ARTS/MELLLA). The Average Power Range Monitor (APRM) flow-biased simulated thermal power scram Allowable Value would be revised to permit operation in the MELLLA region. The current flow-biased Rod Block Monitor (RBM) would also be replaced by a power dependent RBM which also would require new Allowable Values. In addition, the flow-biased APRM simulated thermal power setdown requirement would be replaced by more direct power and flow dependent thermal limits to reduce the need for manual APRM gain adjustments and to provide more direct thermal limits administration during operation at other than rated conditions.

Operation in the MELLLA region will provide improved power ascension capability by extending plant operation at rated power with less than rated flow. Operation in the MELLLA region can result in the need for fewer control rod manipulations to maintain rated power during the fuel cycle. Replacement of the flow-biased APRM simulated thermal power setdown requirement will improve reliability and provide more direct protection of plant limits.

The description and technical basis of the proposed changes are contained in Attachment (1) and the other attachments referenced therein. The proposed Technical Specification (TS) changes are shown in the markup in Attachment (2). Associated TS Bases changes are shown in Attachment (3). The TS Bases

A001

This letter forwards proprietary information in accordance with 10 CFR 2.390. The balance of this letter may be considered non-proprietary upon removal of Attachment (7).

changes are provided for information only and will be processed in accordance with the NMP2 Technical Specifications Bases Control Program (TS 5.5.10). Regulatory commitments associated with the proposed changes are described in Attachment (1). Attachment (4) is a revision to the to the Plant-Specific Evaluations, provided in Reference (a), required by the NUMAC PRNM Retrofit Plus Option III Stability Trip Function Topical Report (NEDC-32410P-A) to address ARTS implementation.

The safety analysis in support of the proposed changes (non-proprietary version) is provided in Attachment (5). A proprietary version of the safety analysis is provided in Attachment (7). Attachment (7) is considered by General Electric (GE) to contain proprietary information exempt from disclosure pursuant to 10 CFR 2.390. Therefore, on behalf of GE, NMPNS hereby makes application to withhold this document from public disclosure in accordance with 10 CFR 2.390(b)(1). An affidavit executed by GE detailing the reasons for the request to withhold the proprietary information is provided in Attachment (6).

NMPNS requests approval of this request by February 28, 2008, with implementation within 60 days of receipt of the approved amendment. This implementation period will provide adequate time to complete implementation activities using the appropriate change control processes prior to startup from NMP2 Refueling Outage 11.

Pursuant to 10 CFR 50.91(b)(1), NMPNS has provided a copy of this license amendment request, with attachments, to the appropriate state representative.

Should you have any questions regarding the information in this submittal, please contact M. H. Miller, Licensing Director, at (315) 349-5219.

Very truly yours,

STATE OF NEW YORK :
: **TO WIT:**
COUNTY OF OSWEGO :

I, Kevin J. Nietmann, being duly sworn, state that I am Acting Vice President Nine Mile Point, and that I am duly authorized to execute and file this request on behalf of Nine Mile Point Nuclear Station, LLC. To the best of my knowledge and belief, the statements contained in this document are true and correct. To the extent that these statements are not based on my personal knowledge, they are based upon information provided by other Nine Mile Point employees and/or consultants. Such information has been reviewed in accordance with company practice and I believe it to be reliable.

Subscribed and sworn before me, a Notary Public in and for the State of New York and County of Oswego, this 30 day of March, 2007.

WITNESS my Hand and Notarial Seal:

Notary Public

My Commission Expires:

THERESA S. CHALONE
NOTARY PUBLIC, STATE OF NEW YORK
OSWEGO COUNTY REG. NO. 01CH4711068
MY COMMISSION EXPIRES 12/31/2010

March 30, 2007
Date

KJN/JJD/kms

Attachments:
(1) Technical Basis and No Significant Hazards Determination
(2) Proposed Technical Specification (TS) Changes (Mark-up)
(3) Changes to Technical Specification Bases (Mark-up)
(4) Revisions to Plant-Specific Evaluations Required by NUMAC PRNM Retrofit Plus Option III Stability Trip Function Topical Report (NEDC-32410P-A) for ARTS Implementation
(5) NEDO-33286, Rev. 0, APRM/RBM/Technical Specifications/Maximum Extended Load Line Limit Analysis (ARTS/MELLLA) - Non-Proprietary Version
(6) Affidavit by General Electric
(7) NEDC-33286P, Rev. 0, APRM/RBM/Technical Specifications/Maximum Extended Load Line Limit Analysis (ARTS/MELLLA) - Proprietary Version

cc: S. J. Collins, NRC (without Attachments 6 and 7)
D. V. Pickett, NRC
Resident Inspector, NRC (without Attachments 6 and 7)
J. P. Spath, NYSERDA (without Attachments 6 and 7)

ATTACHMENT (1)

TECHNICAL BASIS AND

NO SIGNIFICANT HAZARDS DETERMINATION

TABLE OF CONTENTS

A1-1. DESCRIPTION

A1-2. PROPOSED CHANGE

A1-3. BACKGROUND

A1-4. TECHNICAL ANALYSIS

A1-5. NO SIGNIFICANT HAZARDS DETERMINATION

A1-6. APPLICABLE REGULATORY REQUIREMENTS/CRITERIA

A1-7. ENVIRONMENTAL ASSESSMENT

A1-8. PRECEDENT

A1-9. REFERENCES

A1-10. REGULATORY COMMITMENTS

A1-1. DESCRIPTION

The proposed amendment would change the Nine Mile Point Unit 2 (NMP2) Technical Specifications (TSs) contained in Appendix A to Renewed Operating License NPF-69 to reflect an expanded operating domain resulting from implementation of Average Power Range Monitor/Rod Block Monitor/Technical Specifications/Maximum Extended Load Line Limit Analysis (ARTS/MELLLA). The Average Power Range Monitor (APRM) flow-biased simulated thermal power Allowable Value would be revised to permit operation in the MELLLA operating domain. The current flow-biased Rod Block Monitor (RBM) would also be replaced by a power dependent RBM which would require new Allowable Values. The flow-biased APRM simulated thermal power setdown requirement would be replaced by more direct power and flow dependent thermal limits administration. The proposed changes to the TSs are described in the following section.

To support Nine Mile Point Nuclear Station, LLC's (NMPNS'), planned schedule for implementation of Extended Power Uprate at NMP2, NMPNS requests approval of this request by February 28, 2008, with implementation within 60 days of receipt of the approved amendment. This implementation period will provide adequate time to complete implementation activities using the appropriate change control processes prior to startup from NMP2 Refueling Outage 11.

A1-2. PROPOSED CHANGE

The proposed license amendment would implement ARTS/MELLLA at NMP2. The proposed TS changes necessary for implementation are described below and are indicated on the mark-up pages provided in Attachment (2). Associated TS Bases changes are shown in Attachment (3). The TS Bases changes are provided for information only and will be processed in accordance with the NMP2 Technical Specifications Bases Control Program (TS Section 5.5.10).

<u>TS Section 3.1.7, Standby Liquid Control (SLC) System</u>

TS Surveillance Requirement (SR) 3.1.7.7 currently specifies the following for each SLC pump:

> "Verify each pump develops a flow rate ≥ 41.2 gpm at a discharge pressure ≥ 1320 psig."

The SLC pump discharge pressure would be raised from 1320 psig to 1325 psig.

<u>TS Section 3.2.4, Average Power Range Monitor (APRM) Gain and Setpoint</u>

This TS section, which includes requirements for flow-biased APRM simulated thermal power setdown, would be deleted. The following additional changes would be made to reflect deletion of TS 3.2.4:

a. The TS Table of Contents would be revised

b. The definition for Maximum Fraction of Limiting Power Density (MFLPD) would be deleted from TS Section 1.1.

c. References to TS Section 3.2.4 would be deleted from SR 3.3.1.1.3.

TS Table 3.3.1.1-1, Reactor Protection System Instrumentation

The Allowable Value for Function 2.b, APRM Flow Biased Simulated Thermal Power – Upscale, would be changed to:

$$\leq .64W + 63.8\% \text{ RTP and} \leq 115.5\% \text{ RTP}^{(b)}$$

The Allowable Value for single loop operation contained in Footnote (b) is not changed.

TS Section 3.3.2.1, Control Rod Block Instrumentation

SR 3.3.2.1.4 would be revised to require verification that the ARTS based power dependent RBM Power Range - Upscale Functions are not bypassed at the appropriate power levels consistent with the standard TSs included in NUREG-1433 (Reference 1).

Table 3.3.2.1-1, Control Rod Block Instrumentation, would be revised as follows:

a. Current RBM Functions 1.a, Upscale, and 1.c, Downscale, would be deleted.

b. Current RBM Function 1.b, Inop, would be redesignated Function 1.d.

c. New power dependent RBM Functions 1.a, Low Power Range – Upscale, 1.b, Intermediate Power Range – Upscale, and 1.c, High Power Range – Upscale, would be added. Appropriate requirements for the Applicable Modes or Other Specified Conditions, Required Channels, Surveillance Requirements, and Allowable Value columns of the table would be added for these new Functions.

d. Current note (a) would be deleted.

e. New notes (a) through (e) would be added. These new notes identify the Applicable Modes or Other Specified Conditions for new RBM Functions 1.a through 1.c and for revised Function 1.d.

f. The applicability of SR 3.3.2.1.4 would be deleted for revised Function 1.d.

g. Current notes (b) and (c) would be redesignated (f) and (g), respectively.

h. A new note (h) would be added. This note would specify that the Allowable Values for RBM Functions 1.a, 1.b, and 1.c are identified in the Core Operating Limits Report (COLR).

TS Section 3.4.1, Recirculation Loops Operating

TS Limiting Condition for Operation (LCO) 3.4.1 for single recirculation loop operation would be revised by deleting item d, which resets the Allowable Value for LCO 3.3.2.1, "Control Rod Block Instrumentation," Function 1.a (Rod Block Monitor - Upscale), during single loop operation. Editorial changes would be made to items b and c to reflect the deletion of item d.

TS Section 5.6.5, Core Operating Limits Report (COLR)

Item 5 of Section 5.6.5.a would be revised to specify that the Allowable Values and Minimum Critical Power Ratio (MCPR) conditions for the RBM-Upscale Functions of Specification 3.3.2.1 are to be included in the COLR.

A1-3. BACKGROUND

Many factors restrict the flexibility of a Boiling Water Reactor (BWR) during power ascension from the low-power/low-core flow condition to the high-power/high-core flow condition. Some of the factors at NMP2 that restrict plant flexibility are:

- The current operating power/flow (P/F) map,
- The APRM flow-biased simulated thermal power setdown requirement, and
- The RBM flow-referenced rod block trip.

Once rated power is achieved, periodic control rod and flow adjustments must be made to compensate for reactivity changes due to xenon effects and fuel burnup.

NMP2 currently operates in the Extended Load Line Limit Analysis (ELLLA) region up to approximately 108% rod line based on the current licensed thermal power (CLTP) and Increased Core Flow (ICF) region up to 105% core flow, which results in a core flow window of 87% to 105% at rated thermal power (RTP).

A further expansion of the operating domain (MELLLA) and implementation of ARTS would allow for more efficient and reliable power ascensions and would allow rated power to be maintained over a wider core flow range, thereby reducing the frequency of control rod manipulations that require power maneuvers to implement. Expansion of the operating domain beyond the current P/F map requires changes to the APRM and RBM trip functions described below.

APRM and RBM Allowable Values

The APRM flow-biased simulated thermal power Allowable Value varies as a function of reactor recirculation loop flow, but is clamped such that it is always less than the APRM neutron flux-high Allowable Value.

The flow-biased RBM Allowable Values will be replaced by power dependent Allowable Values. The RBM is designed to prohibit erroneous withdrawal of a control rod during operation at high power levels. This prevents local fuel damage during a single rod withdrawal error.

APRM Allowable Value Setdown Requirement

LCO 3.2.4 currently requires the APRM flow-biased simulated thermal power Allowable Value to be reduced when the Fraction of Rated Thermal Power (FRTP) is less than the MFLPD. The setdown requirement ensures that margins to the fuel cladding safety limit are preserved during operation at other than rated conditions. As an alternative to adjusting the APRM flow-biased simulated thermal power Allowable Value, the APRM gains may be adjusted such that the APRM readings are greater than or

equal to 100% times MFLPD. The NMP2 normal operating practice is to adjust APRM gains when required to meet LCO 3.2.4. Each APRM channel is typically bypassed while the required gain adjustment is made.

The setdown requirement originated from the Hench-Levy Minimum Critical Heat Flux Ratio (MCHFR) thermal limit criterion. Improved methodologies have subsequently been developed to provide more effective alternatives to the setdown requirement.

Previous Power Uprate

NMP2 has performed a Stretch Power Uprate, which increased the CLTP to 3467 MWt or 104.3% of the Original Licensed Thermal Power (OLTP), 3323 MWt (References 2 and 3). The analysis thermal power for the uprate was 1.02 x 3467 MWt or 3536MWt.

A1-4. TECHNICAL ANALYSIS

The proposed changes would reflect an expanded operating domain resulting from implementation of Maximum Extended Load Line Limit Analysis. In addition, the flow-biased simulated thermal power Allowable Value setdown requirement would be replaced by more direct power and flow dependent thermal limits to reduce the need for manual setpoint adjustments and allow more direct thermal limits administration.

Summary of Safety Analyses Included in Attachment (7)

Safety analyses performed in support of the proposed changes are described in Attachment (7). These changes include fuel performance event evaluations (Sections 3.0 and 4.0), an evaluation of vessel overpressure protection (Section 5.0), an evaluation of thermal-hydraulic stability (Section 6.0), an evaluation of the loss-of-coolant accident (Section 7.0), containment response evaluations (Section 8.0), reactor internals integrity evaluations (Section 9.0), an evaluation of an anticipated transient without scram (Section 10.0), an evaluation of steam dryer and separator performance (Section 11.0), and high energy line break evaluations (Section 12.0). A description of planned testing is included in Section 13.0. The following technical analysis summarizes or supplements the information in Attachment (7).

Attachment (7), Section 1.0, Introduction, and Section 2.0, Overall Analysis Approach, provide a description and background for the implementation of ARTS/MELLLA at NMP2. The content of Sections 1.0 and 2.0, relative to fuel dependent evaluations, describes the approach NMPNS is taking to justify and implement the ARTS/MELLLA bases. The assumptions and conclusions described in Section 1.0 and 2.0 for fuel dependent evaluations are based upon the current NMP2 Cycle 11 core design using GE14 and GE11 fuel and in some cases on existing analyses for plants similar to NMP2.

The content of Attachment (7), Sections 1.0 and 2.0, relative to non-fuel dependent evaluations, describes the approach NMPNS is taking to justify and implement the ARTS/MELLLA bases and reflect the NMP2 configuration. The assumptions and conclusions described in Sections 1.0 and 2.0 relative to non-fuel dependent evaluations are applicable for NMP2.

Attachment (7), Sections 3.0 Fuel Thermal Limits, 4.0, Rod Block Monitor System Improvements, 5.0 Vessel Overpressure Protection, and 6.0 Thermal-Hydraulic Stability, describe particular aspects of the

implementation of ARTS/MELLLA for NMP2 Cycle 11. These sections describe fuel dependent evaluations. The content of the sections describes the approach NMPNS is taking to justify and implement the ARTS/MELLLA bases. The assumptions and conclusions for the fuel dependent evaluations are based upon NMP2 Cycle 11 core design using GE14 and GE11 fuel.

Attachment (7), Section 7.0, Loss-of-Coolant Accident Analysis, describes a fuel-dependent evaluation. Analysis in this section is based on a full core of GE14 fuel, which was determined to be conservative with respect to GE11 fuel for Emergency Core Cooling Systems (ECCS)-Loss of Coolant Accident (LOCA) analysis and will be representative of the NMP2 Cycle 12 core. The content of this section describes the approach NMPNS is taking to justify and implement the ARTS/MELLLA bases and reflects the NMP2 plant configuration.

Attachment (7), Section 8.0, Containment Response, describes a non-fuel dependent evaluation. The section describes the approach NMPNS is taking to justify and implement the ARTS/MELLLA bases and reflects the NMP2 plant configuration. The assumptions and conclusions described are applicable for NMP2.

Attachment (7), Section 9.0, Reactor Internals Integrity, describes non-fuel dependent evaluations with the exception of Section 9.1, Reactor Internal Pressure Differences, which contains some fuel-dependent aspects. Section 9.0 describes the approach NMPNS is taking to justify and implement the ARTS/MELLLA bases and reflects the current NMP2 plant configuration. The assumptions and conclusions described are applicable for NMP2. Although Section 9.1 has aspects that are fuel dependent, further fuel dependent evaluation is not required. Section 9.1 indicates that the existing NMP2 ELLLA and ICF bases are bounding relative to the MELLLA application and therefore no specific fuel evaluations are required to justify the ARTS/MELLLA bases.

Attachment (7), Section 10.0, Anticipated Transient Without Scram (ATWS), describes an evaluation that can be considered fuel dependent. The ATWS evaluation described in Section 10.0 is an NMP2 plant specific evaluation using inputs related to the NMP2 Cycle 11 core. The contents of the section describe the approach NMPNS is taking to justify and implement the ARTS/MELLLA bases.

Attachment (7), Sections 11.0, Steam Dryer and Separator Performance, and 12.0, High Energy Line Break, describe non-fuel dependent evaluations relative to the effects of the ARTS/MELLLA bases. The sections describe the approach NMPNS is taking to justify and implement the ARTS/MELLLA bases and reflect the NMP2 plant configuration. The assumptions and conclusions described are applicable for NMP2.

Attachment (7), Section 13.0, Testing, describes the planned testing which will be performed in support of the ARTS/MELLLA implementation.

ARTS/MELLLA Implementation

The expanded operating domain includes changes for ARTS/MELLLA consistent with approved operating domain improvements at other BWRs. The current ELLLA power-flow upper boundary is modified to include the operating region bounded by the rod line which passes through the 100% CLTP and 80% rated core flow point. The power-flow region that is above the current ELLLA boundary is referred to as the MELLLA region.

NMP2 currently uses a digital NUMAC power range neutron monitoring system (PRNMS) (Reference 4). As part of ARTS/MELLLA, the current flow-biased RBM would be replaced by a power dependent RBM. The change to the power dependent RBM can be accomplished with the current NUMAC PRNMS hardware. The change from the flow-biased RBM to the power dependent RBM would also require new Allowable Values. Additionally, the change to the power dependent RBM eliminates the need to maintain flow dependent RBM – Upscale Allowable Values for two loop and single recirculation loop operation; thereby removing the LCO 3.4.1 restriction to reset the RBM – Upscale Allowable Value when entering single loop operation.

The ARTS/MELLLA application is evaluated on a plant-specific basis via a safety and system response analysis for meeting thermal and reactivity margins for BWR plants. When compared to the existing power/flow operating domain, operation in the MELLLA region results in plant operation along a higher rod line, which at off-rated operation allows for higher core power at a given core flow. This increases the fluid subcooling in the downcomer region of the reactor vessel and alters the power distribution in the core in a manner that can potentially affect steady-state operating thermal limit and transient/accident performances. The effect of this operating mode relative to fuel dependent analyses has been evaluated to confirm compliance with the required fuel thermal margins during plant operation. For subsequent reload cycles, NMPNS will include the ARTS/MELLLA operating condition in the reload analysis. Attachment (7) presents the results of the safety analyses and system response evaluations for the non-fuel dependent tasks and the assumptions and conclusions that will be validated or updated for the fuel dependent tasks performed for operation of NMP2 in the region above the current ELLLA and up to the MELLLA boundary line.

With the proposed power/flow map expansion to include the MELLLA region, the upper boundary of the operating domain would be extended to approximately the 115.8% rod line for two loop operation. To accommodate this expanded operating domain, the APRM flow biased simulated thermal power Allowable Value would be revised. The APRM clamp will be unchanged. The MELLLA region would not be used for single loop operation.

Although it is part of the NMP2 design configuration and Technical Specifications, the APRM flow-biased simulated thermal power Allowable Value is not credited in any specific NMP2 safety analysis. The proposed Allowable Value change would permit operation in the MELLLA region for operational flexibility purposes.

Representative results of the Rod Withdrawal Error (RWE) event (with the ARTS based power dependent RBM hardware) demonstrate that the MCPR Safety Limit (SL) and fuel thermal-mechanical design limits are not exceeded, when appropriate power dependent trip setpoints are used in the RBM.

One objective of the ARTS/MELLLA APRM improvements is to justify removal of the APRM trip setdown requirement (TS 3.2.4, APRM Gain and Setpoint) using the following criteria:

- MCPR SL shall not be violated as a result of any Anticipated Operational Occurrence (AOO).
- All fuel thermal-mechanical design bases shall remain within the licensing limits.
- Peak cladding temperature and maximum cladding oxidation fraction following a LOCA shall remain within the limits defined in 10 CFR 50.46.

Power and flow dependent MCPR adjustments to the MCPR and linear heat generation rate (LHGR) thermal limits will be determined using NRC approved analytical methods identified in TS 5.6.5. These

adjustments will ensure that the above three criteria are met during operation at other than rated conditions without the APRM trip setdown.

ATWS Analysis

Attachment (7), Section 10.0, discusses the results of the ATWS analysis performed for ARTS/MELLLA conditions. The ATWS analysis resulted in a peak upper plenum pressure that is 5 psi greater than the current analysis. The increased upper plenum pressure results in a corresponding 5 psi increase in the required SLC pump discharge pressure (which is specified in SR 3.1.7.7).

The increase in peak upper plenum pressure is not due to implementation of MELLLA, but rather to differences in the modeling assumptions used in the revised ATWS analysis. The 5 psi difference is not unexpected due to the following reasons:

- A new model (ODYN) was used in the analysis. The ODYN methods calculate a slightly different pressure drop from the upper plenum to the safety/relief valve (S/RV). In addition, the ODYN modeling of the S/RV's sensing location for the safety mode conservatively uses the relief mode sensing location (dome), and the corresponding relief mode opening delay and an opening stroke time. These modeling changes increase the pressure by approximately 3 psi.

- The revised analysis reports the peak pressure from different reference locations. For ARTS/MELLLA, the peak pressure corresponds to the peak upper plenum pressure (high pressure core spray injection point). For the previous ATWS analysis, the peak pressure was approximated from the peak dome pressure. This change increases the pressure by approximately 2 psi.

The current and proposed changes to the SLC system parameters are shown below:

	Current (psig)	Proposed (psig)
SLC Discharge Pressure	1320	1325
SLC Relief Valve Setpoint	1394	1400
System Design Pressure	1400	1400 (no change)
Discharge/Relief Valve Margin	74	75

In order to preserve the margin between the SLC pump discharge pressure and the relief valve setpoint, the relief valve setpoint will be raised to 1400 psig. The proposed margin of 75 psig between the TS required discharge pressure and the relief valve setpoint includes 30 psig to accommodate for pressure fluctuations due to pump pulsation, and 42 psig to accommodate for set pressure tolerance, a value of 3%. Three psig remain for overall margin. The 75 psig margin reflects the margin value recommended generically by General Electric (GE). The revised relief valve setpoint of 1400 psig will continue to ensure compliance with Section III of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code.

Additional Discussion of SR 3.3.2.1.4 Changes

The surveillance requirement would be modified from that shown in NUREG-1433 (Reference 1). The revised SR has been written based on APRM simulated thermal power input, the digital signal that is

actually used in the NUMAC RBM, not thermal power as in the NUREG. Additionally, the exception for peripheral control rods included in the current SR is maintained since the RBM will continue to be automatically bypassed if a peripheral control rod is selected. These surveillance and operability requirement clarifications result in no functional changes in the equipment performance or operational limits.

Additional Discussion of TS Table 3.3.2.1-1 Changes

This table would be modified to change from a flow-biased RBM to a power dependent RBM consistent with NUREG-1433. Changes from the NUREG would include specifying that the RBM Allowable Values for Low Power Range - Upscale, Intermediate Power Range - Upscale, and High Power Range - Upscale would be in the COLR. Also, the MCPR limits applicable to the operability of the RBM would be specified in the COLR. The RBM power range Allowable Values and MCPR values are calculated on a cycle specific basis. These changes are similar to ones previously approved for Peach Bottom Units 2 and 3.

The current exception in the applicable Mode or other specified conditions note (a) of Table 3.3.2.1-1 for the RBM that excludes operability when a peripheral control rod is selected will be maintained in the new applicability notes (a) through (e) for the RBM Functions. The RBM will continue to be automatically bypassed if a peripheral control rod is selected. This exception is consistent with the ARTS based RBM applicability notes previously approved for Cooper. Additionally, notes (a) through (e) have been written based on APRM simulated thermal power input, the digital signal that is actually used in the NUMAC RBM, not thermal power as in the NUREG.

The RBM downscale function would also be deleted. This deletion is intended to simplify the TSs by deleting a Function that has no significant value due to differences between an analog system and a digital system. Further justification is provided in Attachment (4).

Deletion of Applicability of SR 3.3.2.1.4 to RBM Inop Function in TS Table 3.3.2.1-1

The RBM Inop Function inserts a rod block when too few Local Power Range Monitors (LPRMs) are available as discussed in Section 4.2 of Attachment (7). The RBM Inop Function is not affected by the proposed implementation of ARTS/MELLLA.

The current NMP2 TS, Table 3.3.2.1-1 note (a), requires the RBM Inop Function 1.b to be Operable when thermal power is ≥ 30% RTP and no peripheral control rod is selected. The current SR 3.3.2.1.4 requires verification that the RBM Inop Function is not bypassed at ≥ 30% RTP and a peripheral control rod is not selected. This SR is duplicative to the LCO applicability for the RBM Inop Function. If the RBM Inop Function is bypassed, the RBM is not capable of performing its function as described in the TS and thus is not Operable.

The NMP2 ARTS/MELLLA application proposes a revised SR 3.3.2.1.4 and revised Table 3.3.2.1-1 function applicability notes. The revised SR is worded such that that the SR excludes the RBM Inop Function. The revised function applicability notes in Table 3.3.2.1-1 require the RBM Inop Function to be applicable at APRM Simulated Thermal Power ≥28% RTP with MCPR less than the limits specified in the COLR and no peripheral control rod selected. The deletion of the applicability of SR 3.3.2.1.4 to the RBM Inop Function is consistent with the standard technical specifications presented in Reference 1.

Limiting Safety System Setting (LSSS) Determination

The setpoints removed or altered by this license amendment request are as follows:

- Existing TS 3.2.4, Average Power Range Monitor (APRM) Gain and Setpoint, is being deleted by the proposed change. This specification allows adjustment of the APRM Flow Biased Simulated Thermal Power – Upscale Function Allowable Value when operating under conditions of excessive power peaking to maintain acceptable margin to the fuel cladding integrity safety limit and the fuel cladding 1% plastic strain limit. This specification is no longer needed since improved methodologies provide more effective alternates to the requirement.

- The Allowable Value for two-loop operation specified in TS Table 3.3.1.1-1 for Function 2.b, Flow Biased Simulated Thermal Power – Upscale, is being revised. As described in the TS Bases for Specification 3.3.1.1, no specific safety analyses take direct credit for the APRM Flow Biased Simulated Thermal Power – Upscale Function. Originally, the clamped Allowable Value was based on analyses that took credit for the APRM Flow Biased Simulated Thermal Power – Upscale Function for the mitigation of the loss of feedwater heater event. However, the current methodology for this event is based on a steady state analysis that allows power to increase beyond the clamped Allowable Value. Therefore, applying the current clamped Allowable Value is conservative. The TS Bases for this specification also state that functions not specifically credited in the accident analysis are retained for overall redundancy and diversity of the reactor protection system (RPS) as required by the NRC approved licensing basis. Therefore, this function is part of the RPS and is included in the TS since it is part of the RPS design and is part of the existing licensing basis.

- The flow dependent Allowable Value specified in TS Table 3.3.2.1-1 for Function 1.a, Rod Block Monitor – Upscale is being replaced by three power dependent Allowable Values. The three power dependent Allowable Values are for new Function 1.a, Rod Block Monitor, Low Power Range – Upscale, Function 1.b, Rod Block Monitor, Intermediate Power Range – Upscale, and Function 1.c, Rod Block Monitor, High Power Range – Upscale. The values for the three power dependent Allowable Values are to be located in the COLR.

- The existing TS Table 3.3.2.1-1 Function 1.c, Rod Block Monitor – Downscale, is proposed to be deleted.

The current TS Bases for Specification 3.3.2.1 state that the RBM is designed to prevent violation of the MCPR SL and the cladding 1% plastic strain fuel design limit that may result from a RWE. As such, the RBM has associated LSSSs. The NMP2 Updated Safety Analysis Report Section 15.4.2 states that the RWE is evaluated for each reload as a potentially limiting event. The current reload analyses do not take credit for the RBM system. Therefore, the RBM currently provides defense in depth.

With implementation of the ARTS/MELLLA license amendment, the rod block function (with three power dependent Allowable Values) will be credited in the transient analysis with protecting the MCPR SL specified in TS 2.1.1.2 and will have associated LSSSs. The RWE will continue to be evaluated each reload as a potentially limiting event.

As discussed in Attachment (4), the RBM Downscale Function will detect substantial reductions in the RBM local flux after a "null" is completed (a "null" occurs after a new rod selection). This function, in combination with the RBM Inop Function, was intended in the original system to detect problems with or

abnormal conditions in the RBM equipment and system. However, no credit is taken for the RBM Downscale Function in the establishment of the RBM Upscale Allowable Values. Therefore, this function is part of the control rod block instrumentation and was initially included in the TSs since it is part of the RBM design and also part of the existing licensing basis. One effect of the digital processing used by the NUMAC RBM is to eliminate the types of failures that can reasonably be detected by a Downscale Function. The Inop Function in the NUMAC RBM uses an automatic self-test and other internal logic to detect failures and abnormal conditions that can occur in digital equipment. Therefore, when utilizing the NUMAC RBM for ARTS, there is no incremental value or benefit provided by the RBM Downscale Function. Consistent with the overall thrust of the Improved Technical Specifications to eliminate "no value" requirements, the RBM Downscale Function is being removed from the TSs. The RBM Inop Function is retained in the TSs.

Therefore, of the TS functions removed or altered by this license amendment change, the RBM power dependent Allowable Values are considered Limiting Safety System Settings.

Setpoint Methodology for LSSSs

Allowable Values and setpoints for the RBM power dependent functions are calculated on a cycle specific basis using GE setpoint methodology. The GE setpoint methodology is described in NEDC-31336 P-A, "General Electric Setpoint Methodology," September 1996 and has been approved by the NRC as documented in the associated Safety Evaluation Report (SER). The new RBM Allowable Values were calculated using this methodology and the results are included in GE document 0000-0053-1006 NMP2 A-M-T506-RBM-Calc-2006, "Instruments Limits Calculation, Constellation Generation Group, Nine Mile Point Nuclear Station Unit 2, Rod Block Monitor (NUMAC ARTS-MELLLA)," Rev. 0, January 2007, which is included in Attachment (7) to this license amendment request.

Generic Issues Related to Setpoint Allowable Values

On September 7, 2005, the NRC issued a letter to the NEI Setpoint Methods Task Force entitled, "Technical Specification for Addressing Issues Related to Setpoint Allowable Values." This letter provides NRC expectations for addressing staff concerns related to technical specification Allowable Values in plant-specific license amendment requests.

For the Allowable Values associated with LSSSs that are proposed to be altered by this license amendment request (Power Dependent Rod Block Functions), NMPNS does not plan to implement the Allowable Value related TSs described in the September 7, 2005, NRC letter. The application of the suggested notes to these instrument functions is unnecessary due to the specific nature of this instrumentation.

The RBM Functions associated with protecting the fuel cladding during the RWE analysis are provided by a digital device. The digital device utilizes a nominal trip setpoint that has no additional conservatisms added to account for testing and calibration error. There are no margins applied to the RBM nominal trip setpoint calculations which could mask RBM degradation. There are no as-left tolerances and no as-found tolerances associated with these digital trip settings.

With the implementation of ARTS/MELLLA, a more direct trip logic than is currently provided is implemented (See Figure 4-4 in Attachment (7)). The RBM takes input from the LPRMs surrounding the rod that is selected for withdrawal and an average of these readings at the time of rod selection is

calculated. A "nulling" operation is then performed which establishes the pre-rod motion value. This value is normalized to 100%. This nulling establishes a fixed reference level (100) identified on Figure 4-4 inputting into the "calibration" box. As the rod is pulled, the LPRM readings increase and subsequent average values from the same set of LPRMs are calculated. The "calibration" box in the figure represents that the value is then divided by the average at the time of nulling and is multiplied by 100 to give the instantaneous RBM readings (signal that is shown exiting the "calibration" box). If this reading exceeds the trip setpoint, a rod block is issued that protects against rod withdrawal errors. Since the RBM reading is a ratio relative to the value just before rod pull, LPRM drift and calibration errors from the previous LPRM calibration are of no real significance because they cancel out when the ratio is taken. The reference level is the level to which the RBM is automatically calibrated upon rod selection.

The RBM trip setpoints (low, intermediate, or high) are enabled at three simulated thermal power levels from the APRM (shown on Figure 4-4 coming into the comparator from the left). The surveillances for enabling simulated thermal power values are covered by the APRM TSs and are not part of the RBM.

The RBM trip setpoints are determined by use of NRC approved setpoint methodology. Using the GE setpoint methodology based on Instrumentation, Systems, and Automation Society (ISA) setpoint calculation method 2, the RBM Allowable Values are determined from the analytical limit, corrected for RBM input signal calibration error, process measurement error, primary element accuracy and instrument accuracy under trip conditions. The error due to the neutron flux measurement is accounted for in the non-linearity error from the LPRM detectors and is referred to in the setpoint calculation as the APRM Primary Element Accuracy. There is both a bias and random component to this error. There is also an error due to tracking and neutron flux noise, and that is labeled as Process Measurement Accuracy (PMA). The RBM trip setpoint has no drift characteristic, with no as-left or as-found tolerances, since it only performs digital calculations on digitized input signals. The Nominal Trip Setpoint (NTSP) includes a drift allowance over the interval from rod selection to rod movement, which is not the surveillance interval. Drift of RBM channel components between surveillance intervals does not apply to the normalized RBM reading.

Surveillance procedures are used to establish operability of the RBM. The surveillance procedures include appropriate steps to ensure the RBM is functioning properly and that the proper setpoint values are established in the hardware. Other self-test functions are performed automatically and routinely in the RBM hardware modules (Central Processing Unit, Power Supplies, etc.) The periodic RBM calibration in the TSs requires a verification of only the trip setting. The trip setpoints are stored in computer memory as fixed numerical values and thus cannot drift due to the nature of the RBM instrument (digital hardware). The calibration method in the TS surveillance procedures ensures that the trip setting is proper. Since the trip setpoint is a numerical value stored in the digital hardware and not subject to drift, the as-found and as-left tolerance values for the setpoint are the same as the setpoint (i.e. there is no tolerance band). The surveillance procedures also perform a channel functional test, which assures the RBM is functioning properly.

The TS Bases for the instruments that have Allowable Values modified by this license amendment request (APRM Flow Biased Simulated Thermal Power – Upscale and RBM Power Ranges – Upscale) indicate that the instrument channels are operable when the actual setpoints are within the Allowable Values, i.e., a channel is inoperable if its setpoint is found to be above its Allowable Value. Additionally, the NMPNS corrective action program requires a condition report to be written to address instruments and equipment found out of calibration or tolerance required to maintain loop or system function within acceptable calibration or tolerance.

Safety/Relief Valve Setpoints

The ATWS, LOCA, and overpressure analyses are performed using the TS SR 3.4.4.1 safety/relief valve (S/RV) lift setpoints. These setpoints include an approximately ± 3% as-found tolerance and a ± 1% as-left tolerance. The TS S/RV settings are:

Number of S/RVs	Setpoint (psig)
2	1165 ± 35.0
4	1175 ± 35.0
4	1185 ± 36.0
4	1195 ± 36.0
4	1205 ± 36.0

Actual historical in-service surveillance test results of S/RV performance are monitored for compliance in accordance with TSs and ASME/ANSI OM 1987 - (Part 1), "Requirements for In-service Performance Testing of Nuclear Power Plant Pressure Relief Devices." Of the 103 as-found S/RV lift setpoint verification tests performed from 1988 to 2006, only two (2) S/RVs have failed to meet the as specified setpoint tolerance, with one (1) S/RV test found to be above and one (1) found to be below the setpoint tolerance. Thus, the in-service surveillance testing of the S/RVs has not shown a significant propensity for high setpoint drift greater than the approximately ± 3% specified in the TSs.

The performance of the S/RVs is unaffected by operation in the MELLLA domain. The results presented in Attachment (7) show that the applicable analysis requirements continue to be met under MELLLA conditions. Therefore, the current S/RV setpoints remain valid.

Conclusion

The proposed changes will increase operating flexibility in power ascension and operation at rated power. Replacement of the APRM setdown requirement with more direct power and flow dependent thermal limits will reduce the need for manual Allowable Value or gain adjustments and allow for more direct thermal limits administration. This will improve the human/machine interface, update thermal limits administration, increase reliability, and provide more direct protection of plant limits.

A1-5. NO SIGNIFICANT HAZARDS DETERMINATION

Nine Mile Point Nuclear Station, LLC (NMPNS), is requesting a revision to Renewed Operating License No. NPF-69 for Nine Mile Point Unit 2 (NMP2). The proposed amendment would change the NMP2 Technical Specifications (TSs) to implement the Average Power Range Monitor/Rod Block Monitor/Technical Specifications/Maximum Extended Load Line Limit Analysis (ARTS/MELLLA). Specifically, the average power range monitor (APRM) flow-biased simulated thermal power Allowable Value would be revised to permit operation in the MELLLA region. The current flow-biased rod block monitor (RBM) would also be replaced by a power dependent RBM. The change from the flow-biased RBM to the power-dependent RBM would also require new Allowable Values. In addition, the flow-biased APRM simulated thermal power setdown requirement would be replaced by more direct power and flow-dependent thermal limits to reduce the need for APRM gain adjustments, and to allow more direct thermal limits administration during operation at other than rated conditions.

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

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

   Response: No.

   The proposed change eliminates the APRM flow-biased simulated thermal power setdown requirement and substitutes power and flow dependent adjustments to the Minimum Critical Power Ratio (MCPR) and Linear Heat Generation Rate (LHGR) thermal limits. Thermal limits will be determined using NRC approved analytical methods. The proposed change will have no effect upon any accident initiating mechanism. The power and flow dependent adjustments will ensure that the MCPR safety limit will not be violated as a result of any Anticipated Operational Occurrence (AOO), and that the fuel thermal and mechanical design bases will be maintained.

   The proposed change also expands the power and flow operating domain by relaxing the restrictions imposed by the formulation of the APRM flow-biased simulated thermal power Allowable Value and the replacement of the current flow-biased RBM with a new power dependent RBM. The APRM and RBM are not involved in the initiation of any accident, and the APRM flow-biased simulated thermal power function is not credited in any NMP2 safety analyses. The proposed change will not introduce any initial conditions that would result in NRC approved criteria being exceeded and the APRM and RBM will remain capable of performing their design functions.

   The Standby Liquid Control (SLC) System is provided to mitigate anticipated transients without scram (ATWS) events and, as such, is not considered an initiator of an ATWS event or any other analyzed accident. The revised SLC discharge pump test pressure neither reduces the ability of the SLC system to respond to or mitigate an ATWS event nor increases the likelihood of a system malfunction that could increase the consequences of an accident.

   Based on the above discussion, it is concluded that the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Response: No.

The proposed change eliminates the APRM flow-biased simulated thermal power setdown requirement and substitutes power and flow dependent adjustments to the MCPR and LHGR thermal limits. Because the thermal limits will continue to be met, no analyzed transient event will escalate into a new or different type of accident due to the initial starting conditions permitted by the adjusted thermal limits.

The proposed change also expands the power and flow operating domain by relaxing the restrictions imposed by the formulation of the APRM flow-biased simulated thermal power Allowable Value and the replacement of the current flow-biased RBM with a new power dependent RBM. Changing the formulation for the APRM flow-biased simulated thermal power Allowable Value and changing from a flow-biased RBM to a power dependent RBM does not change their respective functions and manner of operation. The change does not introduce a sequence of events or introduce a new failure mode that would create a new or different type of accident. While not credited, the APRM flow-biased simulated thermal power Allowable Value and associated scram trip setpoint will continue to initiate a scram to protect the MCPR safety limit. The power dependent RBM will prevent rod withdrawal when the power dependent RBM rod block setpoint is reached. No new failure mechanisms, malfunctions, or accident initiators are being introduced by the proposed change. In addition, operating within the expanded power flow map will not require any systems, structures or components to function differently than previously evaluated and will not create initial conditions that would result in a new or different kind of accident from any accident previously evaluated.

The proposed change to the SLC pump test discharge pressure is consistent with the functional requirements of the ATWS rule (10 CFR 50.62). This proposed change does not involve the installation of any new or different type of equipment, does not introduce any new modes of plant operation, and does not change any methods governing normal plant operation.

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

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

Response: No.

The proposed change eliminates the APRM flow-biased simulated thermal power setdown requirement and substitutes power and flow dependent adjustments to the MCPR and LHGR thermal limits. Replacement of the APRM setdown requirement with power and flow dependent adjustments to the MCPR and LHGR thermal limits will continue to ensure that margins to the fuel cladding Safety Limit are preserved during operation at other than rated conditions. Thermal limits will be determined using NRC approved analytical methods. The power and flow dependent adjustments will ensure that the MCPR safety limit will not be violated as a result of any AOO, and that the fuel thermal and mechanical design bases will be maintained.

The proposed change also expands the power and flow operating domain by relaxing the restrictions imposed by the formulation of the APRM flow-biased simulated thermal power

Allowable Value and the replacement of the current flow-biased RBM with a new power dependent RBM. The APRM flow-biased simulated thermal power Allowable Value and associated scram trip setpoint will continue to initiate a scram to protect the MCPR safety limit. The RBM will continue to prevent rod withdrawal when the power dependent RBM rod block setpoint is reached. The MCPR and LHGR thermal limits will be developed to ensure that fuel thermal mechanical design bases remain within the licensing limits during a control rod withdrawal error event and to ensure that the MCPR safety limit will not be violated as a result of a control rod withdrawal error event. Operation in the expanded operating domain will not alter the manner in which safety limits, limiting safety system settings, or limiting conditions for operation are determined. AOOs and postulated accidents within the expanded operating domain will continue to be evaluated using NRC approved methods. The 10 CFR 50.46 acceptance criteria for the performance of the ECCS following postulated LOCAs will continue to be met.

The proposed change to the SLC pump discharge test pressure does not alter the results of any accident analyses. The proposed change is consistent with the functional requirements of the ATWS rule (10 CFR 50.62). The ability of the SLC system to respond to and mitigate an ATWS event is not affected.

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

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

A1-6. APPLICABLE REGULATORY REQUIREMENTS/CRITERIA

Analysis

10 CFR 50, Appendix A, General Design Criterion (GDC) 10 requires that the reactor core and associated coolant, control, and protection systems be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences. The assumptions and conclusions relative to fuel dependent calculations will be validated on a cycle specific basis to ensure the requirements of GDC 10 continue to be met.

10 CFR 50, Appendix A, GDC 12 requires that the reactor core and associated coolant, control, and protection systems shall be designed to assure that power oscillations which can result in conditions exceeding specified acceptable fuel design limits are not possible or can be reliably and readily detected and suppressed. The assumptions and conclusions relative to fuel dependent calculations will be validated on a cycle specific basis to ensure the requirements of GDC 12 continue to be met.

10 CFR 50, Appendix A, GDC 50 requires that the reactor containment structure be designed so that the containment structure and its internal compartments can accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from a LOCA. Evaluations described in Attachment (7), Section 8.0 demonstrate that all containment parameters stay within their design limits.

10 CFR 50.46 sets forth acceptance criteria for the performance of the ECCS following postulated LOCAs. 10 CFR 50 Appendix K describes required and acceptable features of the evaluation models

used to calculate ECCS performance. The plant specific LOCA analysis in Section 7.0 of Attachment (7) demonstrates that the requirements of 10 CFR 50.46 continue to be met.

10 CFR 50.49 establishes requirements for environmental qualification of electric equipment important to safety for nuclear power plants. Evaluations described in Attachment (7), Section 12.0 demonstrate acceptable results for the analyzed high energy line breaks.

10 CFR 50.62, in part, specifies the equivalent flow rate, level of boron concentration and boron-10 isotope enrichment required for BWR standby liquid control systems. The analyses described in Attachment (7), Section 10.0, confirm that the key performance parameters (reactor vessel pressure, peak cladding temperature, suppression pool temperature and containment pressure) remain within acceptable limits.

Conclusion

Based on the considerations discussed above and detailed in the attachments to this submittal, (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 requested license amendment will not be inimical to the common defense and security or to the health and safety of the public.

A1-7. ENVIRONMENTAL ASSESSMENT

A review has determined that the proposed amendment 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 amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment 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 amendment.

A1-8. PRECEDENT

Plants with full ARTS/MELLLA implementation, including ICF, include Hatch Units 1 and 2, Duane Arnold (no ICF), Cooper, Pilgrim, Fermi, Monticello, Brunswick Units 1 and 2, Peach Bottom Units 2 and 3, Browns Ferry Units 1, 2 and 3, and Susquehanna Units 1 and 2. Plants with partial ARTS/MELLLA implementation (RBM is not modified to be power dependent), including ICF, include Dresden Units 2 and 3, Quad Cities Units 1 and 2, Vermont Yankee and Hope Creek. FitzPatrick has a partial ARTS submittal currently under review with the NRC.

A1-9. REFERENCES

1. NUREG-1433, Standard Technical Specifications – General Electric Plants, BWR/4, Revision 3

2. NEDC-31994P, Revision 1, Nine Mile Point Nuclear Station Unit 2, Power Uprate Licensing Evaluation for Power Uprate Nine Mile Point Nuclear Station Unit 2, May 1993, Including E&A No. 1 September 1994 and E&A No. 2 November 1994

3. Letter from G. E. Edison (NRC) to B. R. Sylvia (Niagara Mohawk Power Corporation) dated April 28, 1995, Issuance of Amendment for Nine Mile Point Nuclear Station, Unit 2 (TAC No. M87088)

4. Letter from D. S. Hood, (NRC) to J. H. Mueller (Niagara Mohawk Power Corporation) dated March 31, 1998, Issuance of Amendment for Nine Mile Point Nuclear Station, Unit No. 2 (TAC No. MA0150)

A1-10. REGULATORY COMMITMENTS

The following table identifies those actions committed to by NMPNS in this submittal. Any other statements in this submittal are provided for information purposes and are not considered to be regulatory commitments.

REGULATORY COMMITMENT	DUE DATE
Raise the standby liquid control system pump discharge relief valve set pressure to 1400 psig.	60 days following NRC approval of the license amendment request

ATTACHMENT (2)

PROPOSED TECHNICAL SPECIFICATION (TS) CHANGES (MARK-UP)

TS Pages

i
1.1-4
3.1.7-3
3.2.4-1
3.2.4-2
3.3.1.1-4
3.3.1.1-8
3.3.2.1-4
3.3.2.1-6
3.4.1-1
5.6-3

TABLE OF CONTENTS

1.0 USE AND APPLICATION
1.1 Definitions 1.1-1
1.2 Logical Connectors 1.2-1
1.3 Completion Times 1.3-1
1.4 Frequency 1.4-1

2.0 SAFETY LIMITS (SLs)
2.1 SLs 2.0-1
2.2 SL Violations 2.0-1

3.0 LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY . . . 3.0-1
3.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY 3.0-4

3.1 REACTIVITY CONTROL SYSTEMS
3.1.1 SHUTDOWN MARGIN (SDM) 3.1.1-1
3.1.2 Reactivity Anomalies 3.1.2-1
3.1.3 Control Rod OPERABILITY 3.1.3-1
3.1.4 Control Rod Scram Times 3.1.4-1
3.1.5 Control Rod Scram Accumulators 3.1.5-1
3.1.6 Rod Pattern Control 3.1.6-1
3.1.7 Standby Liquid Control (SLC) System 3.1.7-1
3.1.8 Scram Discharge Volume (SDV) Vent and Drain Valves 3.1.8-1

3.2 POWER DISTRIBUTION LIMITS
3.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR) 3.2.1-1
3.2.2 MINIMUM CRITICAL POWER RATIO (MCPR) 3.2.2-1
3.2.3 LINEAR HEAT GENERATION RATE (LHGR) 3.2.3-1
~~3.2.4 Average Power Range Monitor (APRM) Gain and Setpoint . . . . . 3.2.4-1~~ Delete

3.3 INSTRUMENTATION
3.3.1.1 Reactor Protection System (RPS) Instrumentation 3.3.1.1-1
3.3.1.2 Source Range Monitor (SRM) Instrumentation 3.3.1.2-1
3.3.2.1 Control Rod Block Instrumentation 3.3.2.1-1
3.3.2.2 Feedwater System and Main Turbine High Water Level Trip Instrumentation 3.3.2.2-1
3.3.3.1 Post Accident Monitoring (PAM) Instrumentation . . 3.3.3.1-1
3.3.3.2 Remote Shutdown System 3.3.3.2-1
3.3.4.1 End of Cycle Recirculation Pump Trip (EOC-RPT) Instrumentation 3.3.4.1-1
3.3.4.2 Anticipated Transient Without Scram Recirculation Pump Trip (ATWS-RPT) Instrumentation 3.3.4.2-1
3.3.5.1 Emergency Core Cooling System (ECCS) Instrumentation 3.3.5.1-1
3.3.5.2 Reactor Core Isolation Cooling (RCIC) System Instrumentation 3.3.5.2-1

(continued)

1.1 Definitions

LEAKAGE (continued)	2. LEAKAGE into the drywell atmosphere from sources that are both specifically located and known either not to interfere with the operation of leakage detection systems or not to be pressure boundary LEAKAGE; b. Unidentified LEAKAGE All LEAKAGE into the drywell that is not identified LEAKAGE; and c. Pressure Boundary LEAKAGE LEAKAGE through a nonisolable fault in a Reactor Coolant System (RCS) component body, pipe wall, or vessel wall.
LINEAR HEAT GENERATION RATE (LHGR)	The LHGR shall be the heat generation rate per unit length of fuel rod. It is the integral of the heat flux over the heat transfer area associated with the unit length.
LOGIC SYSTEM FUNCTIONAL TEST	A LOGIC SYSTEM FUNCTIONAL TEST shall be a test of all required logic components (i.e., all required relays and contacts, trip units, solid state logic elements, etc.) of a logic circuit, from as close to the sensor as practicable up to, but not including, the actuated device, to verify OPERABILITY. The LOGIC SYSTEM FUNCTIONAL TEST may be performed by means of any series of sequential, overlapping, or total system steps so that the entire logic system is tested.
~~MAXIMUM FRACTION OF LIMITING POWER DENSITY (MFLPD)~~	~~The MFLPD shall be the largest value of the fraction of limiting power density (FLPD) in the core. The FLPD shall be the LHGR existing at a given location divided by the specified LHGR limit for that bundle type.~~ Delete
MINIMUM CRITICAL POWER RATIO (MCPR)	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.

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.1.7.7	Verify each pump develops a flow rate ≥ 41.2 gpm at a discharge pressure ≥ ~~1320~~ 1325 psig.	In accordance with the Inservice Testing Program
SR 3.1.7.8	Verify flow through one SLC subsystem from pump into reactor pressure vessel.	24 months on a STAGGERED TEST BASIS
SR 3.1.7.9	Verify all heat traced piping between storage tank and pump suction valve is unblocked.	24 months AND Once within 24 hours after piping temperature is restored to ≥ 70°F
SR 3.1.7.10	Verify sodium pentaborate enrichment is ≥ 25 atom percent B-10.	Prior to addition to SLC tank

3.2 POWER DISTRIBUTION LIMITS

3.2.4 Average Power Range Monitor (APRM) Gain and Setpoint

LCO 3.2.4

a. MFLPD shall be less than or equal to Fraction of RTP (FRTP); or

b. Each required APRM Flow Biased Simulated Thermal Power—Upscale Function Allowable Value shall be modified by ≤ FRTP/MFLPD; or

c. Each required APRM gain shall be adjusted such that the APRM readings are ≥ 100% times MFLPD.

APPLICABILITY: THERMAL POWER ≥ 25% RTP.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. Requirements of the LCO not met.	A.1	Satisfy the requirements of the LCO.	6 hours
B. Required Action and associated Completion Time not met.	B.1	Reduce THERMAL POWER to < 25% RTP.	4 hours

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SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.4.1	----------NOTE---------- Not required to be met if SR 3.2.4.2 is satisfied for LCO 3.2.4.b or LCO 3.2.4.c requirements. ---------- Verify MFLPD is within limits.	Once within 12 hours after ≥ 25% RTP AND 24 hours thereafter
SR 3.2.4.2	----------NOTE---------- Not required to be met if SR 3.2.4.1 is satisfied for LCO 3.2.4.a requirements. ---------- Verify each required: a. APRM Flow Biased Simulated Thermal Power—Upscale Function Allowable Value is modified by ≤ FRTP/MFLPD; or b. APRM gain is adjusted such that the APRM reading is ≥ 100% times MFLPD.	12 hours

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SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.1.3	----------NOTE---------- Not required to be performed until 12 hours after THERMAL POWER ≥ 25% RTP. ---------- Verify the absolute difference between the average power range monitor (APRM) channels and the calculated power ≤ 2% RTP plus any gain adjustment required by LCO 3.2.4, "Average Power Range Monitor (APRM) Gain and Setpoint," while operating at ≥ 25% RTP.	7 days Delete
SR 3.3.1.1.4	----------NOTE---------- For Functions 1.a and 1.b, not required to be performed when entering MODE 2 from MODE 1 until 12 hours after entering MODE 2. ---------- Perform CHANNEL FUNCTIONAL TEST.	7 days
SR 3.3.1.1.5	Verify the source range monitor (SRM) and intermediate range monitor (IRM) channels overlap.	Prior to fully withdrawing SRMs
SR 3.3.1.1.6	----------NOTE---------- Only required to be met during entry into MODE 2 from MODE 1. ---------- Verify the IRM and APRM channels overlap.	7 days

(continued)

Table 3.3.1.1-1 (page 1 of 3)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Intermediate Range Monitors					
a. Neutron Flux – Upscale	2	3	H	SR 3.3.1.1.1 SR 3.3.1.1.4 SR 3.3.1.1.5 SR 3.3.1.1.6 SR 3.3.1.1.13 SR 3.3.1.1.14	≤ 122/125 divisions of full scale
	5(a)	3	I	SR 3.3.1.1.1 SR 3.3.1.1.4 SR 3.3.1.1.13 SR 3.3.1.1.14	≤ 122/125 divisions of full scale
b. Inop	2	3	H	SR 3.3.1.1.4 SR 3.3.1.1.14	NA
	5(a)	3	I	SR 3.3.1.1.4 SR 3.3.1.1.14	NA
2. Average Power Range Monitors					
a. Neutron Flux – Upscale, Setdown	2	3 per logic channel	H	SR 3.3.1.1.2 SR 3.3.1.1.6 SR 3.3.1.1.7 SR 3.3.1.1.10 SR 3.3.1.1.13	≤ 20% RTP
b. Flow Biased Simulated Thermal Power – Upscale	1	3 per logic channel	G	SR 3.3.1.1.2 SR 3.3.1.1.3 SR 3.3.1.1.7 SR 3.3.1.1.10 SR 3.3.1.1.13	≤ .58W + 62% RTP and ≤ 115.5% RTP(b)
c. Fixed Neutron Flux – Upscale	1	3 per logic channel	G	SR 3.3.1.1.2 SR 3.3.1.1.3 SR 3.3.1.1.7 SR 3.3.1.1.10 SR 3.3.1.1.13	≤ 120% RTP
d. Inop	1,2	3 per logic channel	H	SR 3.3.1.1.7 SR 3.3.1.1.10	NA
e. OPRM-Upscale	1	3 per logic channel	F	SR 3.3.1.1.2 SR 3.3.1.1.7 SR 3.3.1.1.10 SR 3.3.1.1.13 SR 3.3.1.1.16	As specified in the COLR
f. 2-Out-Of-4 Voter	1,2	2	H	SR 3.3.1.1.2 SR 3.3.1.1.10 SR 3.3.1.1.14 SR 3.3.1.1.17	NA

(continued)

(a) With any control rod withdrawn from a core cell containing one or more fuel assemblies.

(b) Allowable Value is .58(W - 5%) + 62% RTP when reset for single loop operation per LCO 3.4.1, "Recirculation Loops Operating."

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.2.1.2	----------NOTE---------- Not required to be performed until 1 hour after THERMAL POWER is ≤ 10% RTP in MODE 1. ---------- Perform CHANNEL FUNCTIONAL TEST.	92 days
SR 3.3.2.1.3	Perform CHANNEL FUNCTIONAL TEST.	184 days
SR 3.3.2.1.4	----------NOTE---------- Neutron detectors are excluded. ---------- Delete Insert 1 → ~~Verify the RBM is not bypassed when THERMAL POWER is ≥ 30% RTP and a peripheral control rod is not selected.~~	24 months
SR 3.3.2.1.5	Verify the RWM is not bypassed when THERMAL POWER is ≤ 10% RTP.	24 months
SR 3.3.2.1.6	----------NOTE---------- Not required to be performed until 1 hour after reactor mode switch is in the shutdown position. ---------- Perform CHANNEL FUNCTIONAL TEST.	24 months

(continued)

INSERT 1

Verify the RBM:

a. Low Power Range - Upscale Function is not bypassed when APRM Simulated Thermal Power is $\geq$ 28% and < 63% RTP and a peripheral control rod is not selected.

b. Intermediate Power Range - Upscale Function is not bypassed when APRM Simulated Thermal Power is $\geq$ 63% and < 83% RTP and a peripheral control rod is not selected.

c. High Power Range - Upscale Function is not bypassed when APRM Simulated Thermal Power is $\geq$ 83% RTP and a peripheral control rod is not selected.

Table 3.3.2.1-1 (page 1 of 1)
Control Rod Block Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Rod Block Monitor				
Insert 2 → a. Upscale	(a)	2	SR 3.3.2.1.3 SR 3.3.2.1.4 SR 3.3.2.1.7	As specified in the COLR
d ~~b~~. Inop	~~(a)~~ (d) (e)	2	SR 3.3.2.1.3 ~~SR 3.3.2.1.4~~	NA
c. Downscale	(a)	2	SR 3.3.2.1.3 SR 3.3.2.1.4 SR 3.3.2.1.7	≥ 3% RTP
2. Rod Worth Minimizer	1 ~~(b)~~ (f), 2 ~~(b)~~ (f)	1	SR 3.3.2.1.1 SR 3.3.2.1.2 SR 3.3.2.1.5 SR 3.3.2.1.8	NA
3. Reactor Mode Switch – Shutdown Position	~~(c)~~ (g)	2	SR 3.3.2.1.6	NA

Delete

Insert 3 → (a) THERMAL POWER ≥ 30% RTP and no peripheral control rod selected. Delete

(f) ~~(b)~~ With THERMAL POWER ≤ 10% RTP.

(g) ~~(c)~~ Reactor mode switch in the shutdown position.

Insert 4 →

INSERT 2

a.	Low Power Range - Upscale	(a)	2	SR 3.3.2.1.3 SR 3.3.2.1.4 SR 3.3.2.1.7	(h)
b.	Intermediate Power Range - Upscale	(b)	2	SR 3.3.2.1.3 SR 3.3.2.1.4 SR 3.3.2.1.7	(h)
c.	High Power Range - Upscale	(c)(d)	2	SR 3.3.2.1.3 SR 3.3.2.1.4 SR 3.3.2.1.7	(h)

INSERT 3

(a) APRM Simulated Thermal Power is ≥ 28% and < 63% RTP and MCPR < limit specified in the COLR and no peripheral control rod selected.

(b) APRM Simulated Thermal Power is ≥ 63% and < 83% RTP and MCPR < limit specified in the COLR and no peripheral control rod selected.

(c) APRM Simulated Thermal Power is ≥ 83% and < 90% RTP and MCPR < limit specified in the COLR and no peripheral control rod selected.

(d) APRM Simulated Thermal Power is ≥ 90% RTP and MCPR < limit specified in the COLR and no peripheral control rod selected.

(e) APRM Simulated Thermal Power is ≥ 28% RTP and < 90% RTP and MCPR < limit specified in the COLR and no peripheral control rod is selected.

INSERT 4

(h) Allowable Value specified in the COLR.

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.1 Recirculation Loops Operating

LCO 3.4.1 Two recirculation loops with matched flows shall be in operation,

OR

One recirculation loop shall be in operation with the following limits applied when the associated LCO is applicable:

a. LCO 3.2.1, "AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)," single loop operation limits specified in the COLR;

b. LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)," single loop operation limits specified in the COLR; and

c. LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation," Function 2.b (Average Power Range Monitors Flow Biased Simulated Thermal Power—Upscale), Allowable Value of Table 3.3.1.1-1 is reset for single loop operation. ~~and~~

Delete

~~d. LCO 3.3.2.1, "Control Rod Block Instrumentation," Function 1.a (Rod Block Monitor—Upscale), Allowable Value of Table 3.3.2.1-1 is reset for single loop operation.~~

APPLICABILITY: MODES 1 and 2.

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

1. The APLHGR for Specification 3.2.1.

2. The MCPR for Specification 3.2.2.

3. The LHGR for Specification 3.2.3.

4. Reactor Protection System Instrumentation Setpoint for the OPRM – Upscale Function Allowable Value for Specification 3.3.1.1.

Insert 5

Delete

5. ~~Control Rod Block Instrumentation Setpoint for the Rod Block Monitor – Upscale Function Allowable Value for Specification 3.3.2.1.~~

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

1. NEDE-24011-P-A-US, "General Electric Standard Application for Reactor Fuel," U.S. Supplement, (NRC approved version specified in the COLR).

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.

(continued)

INSERT 5

5. The Allowable Values and MCPR conditions for the Rod Block Monitor - Upscale Functions for Specification 3.3.2.1.

ATTACHMENT (3)

CHANGES TO TECHNICAL SPECIFICATION BASES (MARK-UP)

The current versions of the following Technical Specifications Bases pages have been marked-up by hand to reflect the proposed changes. These Bases pages are provided for information only and do not require NRC approval.

B i
B 3.1.7-5
B 3.2.2-1
B 3.2.2-2
B 3.2.2-4
B 3.2.3-1
B 3.2.3-3
B 3.2.4-1
B 3.2.4-2
B 3.2.4-3
B 3.2.4-4
B 3.2.4-5
B 3.2.4-6
B 3.3.1.1-28
B 3.3.2.1-1
B 3.3.2.1-2
B 3.3.2.1-3
B 3.3.2.1-4
B 3.3.2.1-9
B 3.3.2.1-12
B 3.4.1-3
B 3.4.1-4

TABLE OF CONTENTS

B 2.0 SAFETY LIMITS (SLs)
B 2.1.1 Reactor Core SLs ..... B 2.0-1
B 2.1.2 Reactor Coolant System (RCS) Pressure SL ..... B 2.0-6

B 3.0 LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY ..... B 3.0-1
B 3.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY ..... B 3.0-12

B 3.1 REACTIVITY CONTROL SYSTEMS
B 3.1.1 SHUTDOWN MARGIN (SDM) ..... B 3.1.1-1
B 3.1.2 Reactivity Anomalies ..... B 3.1.2-1
B 3.1.3 Control Rod OPERABILITY ..... B 3.1.3-1
B 3.1.4 Control Rod Scram Times ..... B 3.1.4-1
B 3.1.5 Control Rod Scram Accumulators ..... B 3.1.5-1
B 3.1.6 Rod Pattern Control ..... B 3.1.6-1
B 3.1.7 Standby Liquid Control (SLC) System ..... B 3.1.7-1
B 3.1.8 Scram Discharge Volume (SDV) Vent and Drain Valves ..... B 3.1.8-1

B 3.2 POWER DISTRIBUTION LIMITS
B 3.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR) ..... B 3.2.1-1
B 3.2.2 MINIMUM CRITICAL POWER RATIO (MCPR) ..... B 3.2.2-1
B 3.2.3 LINEAR HEAT GENERATION RATE (LHGR) ..... B 3.2.3-1
~~B 3.2.4 Average Power Range Monitor (APRM) Gain and Setpoint ..... B 3.2.4-1~~ Delete

B 3.3 INSTRUMENTATION
B 3.3.1.1 Reactor Protection System (RPS) Instrumentation ..... B 3.3.1.1-1
B 3.3.1.2 Source Range Monitor (SRM) Instrumentation ..... B 3.3.1.2-1
B 3.3.2.1 Control Rod Block Instrumentation ..... B 3.3.2.1-1
B 3.3.2.2 Feedwater System and Main Turbine High Water Level Trip Instrumentation ..... B 3.3.2.2-1
B 3.3.3.1 Post Accident Monitoring (PAM) Instrumentation ..... B 3.3.3.1-1
B 3.3.3.2 Remote Shutdown System ..... B 3.3.3.2-1
B 3.3.4.1 End of Cycle Recirculation Pump Trip (EOC-RPT) Instrumentation ..... B 3.3.4.1-1
B 3.3.4.2 Anticipated Transient Without Scram Recirculation Pump Trip (ATWS-RPT) Instrumentation ..... B 3.3.4.2-1
B 3.3.5.1 Emergency Core Cooling System (ECCS) Instrumentation ..... B 3.3.5.1-1
B 3.3.5.2 Reactor Core Isolation Cooling (RCIC) System Instrumentation ..... B 3.3.5.2-1
B 3.3.6.1 Primary Containment Isolation Instrumentation ..... B 3.3.6.1-1
B 3.3.6.2 Secondary Containment Isolation Instrumentation ..... B 3.3.6.2-1

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.1.7.7

Demonstrating each SLC System pump develops a flow rate ≥ 41.2 gpm at a discharge pressure ≥ ~~1320~~ 1325 psig ensures that pump performance has not degraded during the fuel cycle. This minimum pump flow rate requirement ensures that, when combined with the sodium pentaborate solution concentration requirements, the rate of negative reactivity insertion from the SLC System will adequately compensate for the positive reactivity effects encountered during power reduction, cooldown of the moderator, and xenon decay. This test confirms one point on the pump design curve, and is indicative of overall performance. Such inservice tests confirm component OPERABILITY and detect incipient failures by indicating abnormal performance. The Frequency of this Surveillance is in accordance with the Inservice Testing Program.

SR 3.1.7.8 and SR 3.1.7.9

These Surveillances ensure that there is a functioning flow path from the boron solution storage tank to the RPV, including the firing of an explosive valve. The replacement charge for the explosive valve shall be from the same manufactured batch as the one fired or from another batch that has been certified by having one of that batch successfully fired. The pump and explosive valve tested should be alternated such that both complete flow paths are tested every 48 months, at alternating 24 month intervals. The Surveillance may be performed in separate steps to prevent injecting boron into the RPV. An acceptable method for verifying flow from the pump to the RPV is to pump demineralized water from a test tank through one SLC subsystem and into the RPV. While these Surveillances can be performed with the reactor at power, operating experience has shown these components usually pass the Surveillances when performed at the 24 month Frequency, which is based on the refueling cycle. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

Demonstrating that all heat traced piping between the boron solution storage tank and the suction valve to the injection pumps is unblocked ensures that there is a functioning flow

(continued)

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.2 MINIMUM CRITICAL POWER RATIO (MCPR)

BASES

BACKGROUND

MCPR is a ratio of the fuel assembly power that would result in the onset of boiling transition to the actual fuel assembly power. The MCPR Safety Limit (SL) is set such that 99.9% of the fuel rods are expected to avoid boiling transition if the limit is not violated (refer to the Bases for SL 2.1.1.2). The operating limit MCPR is established to ensure that no fuel damage results during anticipated operational occurrences (AOOs). Although fuel damage does not necessarily occur if a fuel rod actually experiences boiling transition (Ref. 1), the critical power at which boiling transition is calculated to occur has been adopted as a fuel design criterion.

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

APPLICABLE SAFETY ANALYSES

The analytical methods and assumptions used in evaluating the AOOs to establish the operating limit MCPR are presented in the USAR, Chapters 4, 6, 15 and Appendix A, and References ~~2 and 3~~. To ensure that the MCPR SL is not exceeded during any transient event that occurs with moderate frequency, limiting transients have been analyzed to determine the largest reduction in critical power ratio (CPR). The types of transients evaluated are loss of flow, increase in pressure and power, positive reactivity insertion, and coolant temperature decrease. The limiting transient yields the largest change in CPR (ΔCPR). When the largest ΔCPR is added to the MCPR SL, the required operating limit MCPR is obtained.

2, 3 and 6

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The MCPR operating limits derived from the transient analysis are dependent on the operating core flow to ensure adherence to fuel design limits during the worst transient that occurs with moderate frequency as identified in USAR, Chapter 15B. Insert B1 Insert B2

Insert B3 → ~~Flow dependent MCPR limits are determined by steady state thermal hydraulic methods using the three dimensional BWR simulator code and the multichannel thermal hydraulic code (Ref. 3). The worst flow increase transient results from recirculation flow controller failure. The K_f curve is derived assuming both recirculation loop controllers fail. This condition produces the maximum possible power increase and hence maximum ΔCPR for transients initiated from less than rated power and flow.~~ Delete

The MCPR satisfies Criterion 2 of Reference 4.

LCO

The MCPR operating limits specified in the COLR are the result of the Design Basis Accident (DBA) and transient analysis. The MCPR operating limit is determined by Insert B4 ~~multiplying the normal MCPR operating limit (100% core flow) by a correction factor based on actual core flow.~~ Delete

---

APPLICABILITY

The MCPR operating limits are primarily derived from transient analyses that are assumed to occur at high power levels. Below 25% RTP, the reactor is operating at a slow recirculation pump speed and the moderator void ratio is small. Surveillance of thermal limits below 25% RTP is unnecessary due to the large inherent margin that ensures that the MCPR SL is not exceeded even if a limiting transient occurs.

Statistical analyses documented in Reference 5 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

(continued)

INSERT B1

and power state ($MCPR_f$ and $MCPR_p$, respectively)

INSERT B2

The determination of MCPR limits is discussed in Reference 6.

INSERT B3

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

INSERT B4

the larger of the $MCPR_f$ and $MCPR_p$ limits.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.2.2.2

Because the transient analysis takes credit for conservatism in the scram speed performance, it must be demonstrated that the specific scram speed distribution is consistent with that used in the transient analysis. SR 3.2.2.2 determines the value of τ, which is a measure of the actual scram speed distribution compared with the assumed distribution. The MCPR operating limit is then determined based on an interpolation between the applicable limits for Option A (scram times of LCO 3.1.4, "Control Rod Scram Times") and Option B (realistic scram times) analyses. The parameter τ must be determined once within 72 hours after each set of scram time tests required by SR 3.1.4.1, SR 3.1.4.2, and SR 3.1.4.4 because the effective scram speed distribution may change during the cycle or after maintenance that could affect scram times. The 72 hour Completion Time is acceptable due to the relatively minor changes in τ expected during the fuel cycle.

REFERENCES

1. NUREG-0562, June 1979.

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

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

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

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

Insert B5 ⟶

INSERT B5

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

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.3 LINEAR HEAT GENERATION RATE (LHGR)

BASES

BACKGROUND

The LHGR is a measure of the heat generation rate of a fuel rod in a fuel assembly at any axial location. Limits on the LHGR are specified to ensure that fuel design limits are not exceeded anywhere in the core during normal operation, including anticipated operational occurrences (AOOs). Exceeding the LHGR limit could potentially result in fuel damage and subsequent release of radioactive materials. Fuel design limits are specified to ensure that fuel system damage, fuel rod failure or inability to cool the fuel does not occur during the anticipated operating conditions identified in Reference 1.

APPLICABLE SAFETY ANALYSES

The analytical methods and assumptions used in evaluating the fuel system design are presented in References 1 and 2. 1,2, and 5. The fuel assembly is designed to ensure (in conjunction with the core nuclear and thermal hydraulic design, plant equipment, instrumentation, and protection system) that fuel damage will not result in the release of radioactive materials in excess of the guidelines of 10 CFR, Parts 20, 50, and 100. The mechanisms that could cause fuel damage during operational transients and that are considered in fuel evaluations are:

a. Rupture of the fuel rod cladding caused by strain from the relative expansion of the UO_2 pellet; and

b. Severe overheating of the fuel rod cladding caused by inadequate cooling.

A value of 1% plastic strain of the fuel cladding has been defined as the limit below which fuel damage caused by overstraining of the fuel cladding is not expected to occur (Ref. 3).

Fuel design evaluations have been performed and demonstrate that the 1% fuel cladding plastic strain design limit is not exceeded during continuous operation with LHGRs up to the operating limit specified in the COLR. The analysis also includes allowances for short term transient operation above the operating limit to account for AOOs, plus an allowance for densification power spiking.

Insert B6 →

(continued)

INSERT B6

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

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.2.3.1

The LHGRs are required to be initially calculated within 12 hours after THERMAL POWER is ≥ 25% RTP and then every 24 hours thereafter. They are compared with the specified limits in the COLR to ensure that the reactor is operating within the assumptions of the safety analysis. The 24 hour Frequency is based on both engineering judgment and recognition of the slowness of changes in power distribution under normal conditions. The 12 hour allowance after THERMAL POWER ≥ 25% RTP is achieved is acceptable given the large inherent margin to operating limits at lower power levels.

REFERENCES

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

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

3. NUREG-0800, Section II A.2(g), Revision 2, July 1981.

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

Insert B7 →

INSERT B7

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

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.4 Average Power Range Monitor (APRM) Gain and Setpoint

BASES

BACKGROUND

The OPERABILITY of the APRMs and their setpoints is an initial condition of all safety analyses that assume rod insertion upon reactor scram. Applicable GDCs are GDC 10, "Reactor Design"; GDC 13, "Instrumentation and Control"; GDC 20, "Protection System Functions"; and GDC 29, "Protection against Anticipated Operation Occurrences" (Ref. 1). This LCO is provided to require the APRM gain or APRM Flow Biased Simulated Thermal Power—Upscale Function Allowable Value (LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation," Function 2.b) to be adjusted when operating under conditions of excessive power peaking to maintain acceptable margin to the fuel cladding integrity Safety Limit (SL) and the fuel cladding 1% plastic strain limit.

Delete Entire Page

The condition of excessive power peaking is determined by the ratio of the actual power peaking to the limiting power peaking at RTP. This ratio is equal to the ratio of the core limiting MFLPD to the Fraction of RTP (FRTP) where FRTP is the measured THERMAL POWER divided by the RTP. Excessive power peaking exists when:

$$\frac{MFLPD}{FRTP} > 1,$$

indicating that MFPLD is not decreasing proportionately to the overall power reduction, or conversely, that power peaking is increasing. To maintain margins similar to those at RTP conditions, the excessive power peaking is compensated by gain adjustment on the APRMs or adjustment of the APRM Flow Biased Simulated Thermal Power—Upscale Function Allowable Value. Either of these adjustments has effectively the same result as maintaining MFLPD less than or equal to FRTP and thus maintains RTP margins for APLHGR, MCPR, and LHGR (Ref. 3).

The normally selected APRM Flow Biased Simulated Thermal Power—Upscale Function Allowable Value positions the scram above the upper bound of the normal power/flow operating region that has been considered in the design of the fuel rods. The Allowable Value is flow biased with a slope that approximates the upper flow control line, such that an

(continued)

BASES

BACKGROUND (continued)

approximately constant margin is maintained between the flow biased trip level and the upper operating boundary for core flows in excess of about 45% of rated core flow. In the range of infrequent operations below 45% of rated core flow, the margin to scram or rod blocks is reduced because of the nonlinear core flow versus drive flow relationship. The normally selected APRM Allowable Value is supported by the analyses presented in Reference 2 that concentrate on events initiated from rated conditions. Design experience has shown that minimum deviations occur within expected margins to operating limits (APLHGR, MCPR, and LHGR), at rated conditions for normal power distributions. However, at other than rated conditions, control rod patterns can be established that significantly reduce the margin to thermal limits. Therefore, the APRM Flow Biased Simulated Thermal Power—Upscale Function Allowable Value may be reduced during operation when the combination of THERMAL POWER and MFLPD indicates an excessive power peaking distribution.

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The APRM neutron flux signal is also adjusted to more closely follow the fuel cladding heat flux during power transients. The APRM neutron flux signal is a measure of the core thermal power during steady state operation. During power transients, the APRM signal leads the actual core thermal power response because of the fuel thermal time constant. Therefore, on power increase transients, the APRM signal provides a conservatively high measure of core thermal power. By passing the APRM signal through an electronic filter with a time constant less than, but approximately equal to, that of the fuel thermal time constant, an APRM transient response that more closely follows actual fuel cladding heat flux is obtained, while a conservative margin is maintained. The delayed response of the filtered APRM signal allows the APRM Flow Biased Simulated Thermal Power—Upscale Function Allowable Value to be positioned closer to the upper bound of the normal power and flow range, without unnecessarily causing reactor scrams during short duration neutron flux spikes. These spikes can be caused by insignificant transients such as performance of main steam line valve surveillances or momentary flow increases of only several percent.

APPLICABLE SAFETY ANALYSES

The acceptance criteria for the APRM gain or setpoint adjustments are that acceptable margins (to APLHGR, MCPR, and LHGR) be maintained to the fuel cladding integrity SL and the fuel cladding 1% plastic strain limit.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

USAR safety analyses (Ref. 2) concentrate on the rated power condition for which the minimum expected margin to the operating limits (APLHGR, MCPR, and LHGR) occurs. LCO 3.2.1, "AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)," LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)," and LCO 3.2.3, "LINEAR HEAT GENERATION RATE (LHGR)," limit the initial margins to these operating limits at rated conditions so that specified acceptable fuel design limits are met during transients initiated from rated conditions. At initial power levels less than rated levels, the margin degradation of the APLHGR, the MCPR, or the LHGR during a transient can be greater than at the rated condition event. This greater margin degradation during the transient is primarily offset by the larger initial margin to limits at the lower than rated power levels. However, power distributions can be hypothesized that would result in reduced margins to the pretransient operating limit. When combined with the increased severity of certain transients at other than rated conditions, the SLs could be approached. At substantially reduced power levels, highly peaked power distributions could be obtained that could reduce thermal margins to the minimum levels required for transient events. To prevent or mitigate such situations, either the APRM gain is adjusted upward by the ratio of the core limiting MFLPD to the FRTP, or the APRM Flow Biased Simulated Thermal Power—Upscale Function Allowable Value is required to be reduced by the ratio of FRTP to the core limiting MFLPD. Either of these adjustments effectively counters the increased severity of some events at other than rated conditions by proportionally increasing the APRM gain or proportionally lowering the APRM Flow Biased Simulated Thermal Power—Upscale Function Allowable Value dependent on the increased peaking that may be encountered.

Delete Entire Page

The APRM gain and setpoint satisfy Criteria 2 and 3 of Reference 4.

LCO

Meeting any one of the following conditions ensures acceptable operating margins for events described above:

a. Limiting excess power peaking;

b. Reducing the APRM Flow Biased Simulated Thermal Power—Upscale Function Allowable Value by multiplying the APRM Flow Biased Simulated Thermal Power—Upscale

(continued)

BASES

LCO (continued)

Function Allowable Value by the ratio of FRTP to the core limiting value of MFLPD; or

c. Increasing the APRM gains to cause the APRM to read greater than 100(%) times MFLPD. This Condition is to account for the reduction in margin to the fuel cladding integrity SL and the fuel cladding 1% plastic strain limit.

MFLPD is the ratio of the limiting LHGR to the LHGR limit for the specific bundle type. As power is reduced, if the design power distribution is maintained, MFLPD is reduced in proportion to the reduction in power. However, if power peaking increases above the design value, the MFLPD is not reduced in proportion to the reduction in power. Under these conditions, the APRM gain is adjusted upward or the APRM Flow Biased Simulated Thermal Power—Upscale Function Allowable Value is reduced accordingly. When the reactor is operating with peaking less than the design value, it is not necessary to modify the APRM Flow Biased Simulated Thermal Power—Upscale Function Allowable Value. Adjusting the APRM gain or modifying the APRM Flow Biased Simulated Thermal Power—Upscale Function Allowable Value is equivalent to maintaining MFLPD less than or equal to FRTP, as stated in the LCO.

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For compliance with LCO Item b (APRM Flow Biased Simulated Thermal Power—Upscale Function Allowable Value modification) or Item c (APRM gain adjustment), only APRMs required to be OPERABLE per LCO 3.3.1.1, Function 2.b, are required to be modified or adjusted. In addition, each APRM may be allowed to have its gain adjusted or Allowable Value modified independently of other APRMs that are having their gain adjusted or Allowable Value modified.

APPLICABILITY

The MFLPD limit, APRM gain adjustment, or APRM Flow Biased Simulated Thermal Power—Upscale Function Allowable Value modification is provided to ensure that the fuel cladding integrity SL and the fuel cladding 1% plastic strain limit are not violated during design basis transients. As discussed in the Bases for LCO 3.2.1, LCO 3.2.2, and LCO 3.2.3, sufficient margin to these limits exists below 25% RTP and, therefore, these requirements are only necessary when the plant is operating at ≥ 25% RTP.

(continued)

BASES (continued)

ACTIONS

A.1

If the APRM gain or Flow Biased Simulated Thermal Power—Upscale Function Allowable Value is not within limits while the MFLPD has exceeded FRTP, the margin to the fuel cladding integrity SL and the fuel cladding 1% plastic strain limit may be reduced. Therefore, prompt action should be taken to restore the MFLPD to within its required limit or make acceptable APRM adjustments such that the plant is operating within the assumed margin of the safety analyses.

The 6 hour Completion Time is normally sufficient to restore either the MFLPD to within limits or the APRM gain or Flow Biased Simulated Thermal Power—Upscale Function Allowable Value to within limits and is acceptable based on the low probability of a transient or Design Basis Accident occurring simultaneously with the LCO not met.

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B.1

If the APRM gain or Flow Biased Simulated Thermal Power—Upscale Function Allowable Value cannot be restored to within their 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 3.2.4.1 and SR 3.2.4.2

The MFLPD is required to be calculated and compared to FRTP or APRM gain or Flow Biased Simulated Thermal Power—Upscale Function Allowable Value to ensure that the reactor is operating within the assumptions of the safety analysis. These SRs are required only to determine the MFLPD and, assuming MFLPD is greater than FRTP, the appropriate APRM gain or Flow Biased Simulated Thermal Power—Upscale Function Allowable Value, and is not intended to be a CHANNEL FUNCTIONAL TEST for the APRM gain or Flow Biased Simulated Thermal Power—Upscale Function circuitry. SR 3.2.4.1 and SR 3.2.4.2 have been modified by Notes, which

(continued)

ASES

SURVEILLANCE REQUIREMENTS

SR 3.2.4.1 and SR 3.2.4.2 (continued)

clarify that the respective SR does not have to be met if the alternate requirement demonstrated by the other SR is satisfied. The 24 hour Frequency of SR 3.2.4.1 is chosen to coincide with the determination of other thermal limits, specifically those for the APLHGR (LCO 3.2.1) and LHGR (LCO 3.2.3). The 24 hour Frequency is based on both engineering judgment and recognition of the slowness of changes in power distribution during normal operation. 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.

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The 12 hour Frequency of SR 3.2.4.2 is required when MFLPD is greater than FRTP, because more rapid changes in power distribution are typically expected.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 10, GDC 13, GDC 20, and GDC 29.

2. USAR, Chapter 15 and Appendix A.

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

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

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.1.1.1 and SR 3.3.1.1.2

Performance of the CHANNEL CHECK once every 12 hours or every 24 hours, as applicable, ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the instrument channels could be an indication of excessive instrument drift on one of the channels or something even more serious. A CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined by the plant staff based on a combination of the channel instrument uncertainties, including indication and readability. If a channel is outside the criteria, it may be an indication that the instrument has drifted outside its limit.

The Frequency is based upon operating experience that demonstrates channel failure is rare. The CHANNEL CHECK supplements less formal, but more frequent, checks of channels during normal operational use of the displays associated with the channels required by the LCO.

SR 3.3.1.1.3

To ensure that the APRMs are accurately indicating the true core average power, the APRMs are calibrated to the reactor power calculated from a heat balance. LCO 3.2.4, "Average Power Range Monitor (APRM) Gain and Setpoint," allows the APRMs to be reading greater than actual THERMAL POWER to compensate for localized power peaking. When this adjustment is made, the requirement for the APRMs to indicate within 2% RTP of calculated power is modified to require the APRMs to indicate within 2% RTP of calculated MFLPD. The Frequency of once per 7 days is based on minor changes in LPRM sensitivity, which could affect the APRM reading between performances of SR 3.3.1.1.7. Delete

An allowance is provided that requires the SR to be performed only at ≥ 25% RTP because it is difficult to accurately maintain APRM indication of core THERMAL POWER

(continued)

B 3.3 INSTRUMENTATION

B 3.3.2.1 Control Rod Block Instrumentation

BASES

BACKGROUND

Control rods provide the primary means for control of reactivity changes. Control rod block instrumentation includes channel sensors, logic circuitry, switches, and relays that are designed to ensure that specified fuel design limits are not exceeded for postulated transients and accidents. During high power operation, the rod block monitor (RBM) provides protection for control rod withdrawal error events. During low power operations, control rod blocks from the rod worth minimizer (RWM) enforce specific control rod sequences designed to mitigate the consequences of the control rod drop accident (CRDA). During shutdown conditions, control rod blocks from the Reactor Mode Switch—Shutdown Position Function ensure that all control rods remain inserted to prevent inadvertent criticalities.

The purpose of the RBM is to limit control rod withdrawal if localized neutron flux exceeds a predetermined setpoint during control rod manipulations (Ref. 1). It is assumed to function to block further control rod withdrawal to preclude a MCPR Safety Limit (SL) violation. The RBM supplies a trip signal to the Reactor Manual Control System (RMCS) to appropriately inhibit control rod withdrawal during power operation above the low power setpoint when a peripheral control rod is not selected. The RBM has two channels, either of which can initiate a control rod block when the channel output exceeds the control rod block setpoint. One RBM channel inputs into one RMCS rod block circuit and the other RBM channel inputs into the second RMCS rod block circuit. ~~The RBM channel signal is generated by averaging a set of local power range monitor (LPRM) signals. One RBM channel averages the signals from LPRM detectors at the A and C positions in the assigned LPRM assemblies, while the other RBM channel averages the signals from LPRM detectors at the B and D positions. Assignment of LPRM assemblies to be used in RBM averaging is controlled by the selection of control rods. The RBM is automatically bypassed and the output set to zero if a peripheral rod is selected or the APRM used to normalize the RBM reading is < 30% RTP. If any LPRM detector assigned to an RBM is bypassed, the computed average signal is automatically adjusted to compensate for the number of LPRM input signals. The minimum number of LPRM inputs required for each RBM channel to prevent an~~

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(continued)

BASES

BACKGROUND (continued)

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instrument inoperative alarm is four when using four LPRM assemblies, three when using three LPRM assemblies, and two when using two LPRM assemblies. Each RBM also receives a recirculation loop flow signal from the reference APRM.

Insert B8 →

When a control rod is selected, the gain of each RBM channel output is normalized to a reference APRM. The gain setting is held constant during the movement of that particular control rod to provide an indication of the change in the relative local power level. If the indicated power increases above the preset limit, a rod block will occur. In addition, to preclude rod movement with an inoperable RBM, a downscale trip and an inoperable trip are provided.

The purpose of the RWM is to control rod patterns during startup and shutdown, such that only specified control rod sequences and relative positions are allowed over the operating range from all control rods inserted to 10% RTP. The sequences effectively limit the potential amount and rate of reactivity increase during a CRDA. A prescribed control rod sequence is stored in the RWM, which will initiate control rod withdrawal and insert blocks when the actual sequence deviates beyond allowances from the stored sequence. The RWM determines the actual sequence based position indication for each control rod. The RWM also uses steam flow signals to determine when the reactor power is above the preset power level at which the RWM is automatically bypassed (Ref. 2). The RWM is a single channel system that provides input into one RMCS rod block circuit.

With the reactor mode switch in the shutdown position, a control rod withdrawal block is applied to all control rods to ensure that the shutdown condition is maintained. This Function prevents inadvertent criticality as the result of a control rod withdrawal during MODE 3 or 4, or during MODE 5 when the reactor mode switch is required to be in the shutdown position. The reactor mode switch has two channels, each inputting into a separate RMCS rod block circuit. A rod block in either RMCS circuit will provide a control rod block to all control rods.

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The RBM channel signal is generated by averaging a set of local power range monitor (LPRM) signals at various core heights surrounding the control rod being withdrawn. A signal from one average power range monitor (APRM) channel assigned to each Reactor Protection System (RPS) trip system supplies a reference signal for the RBM channel in the same trip system. This reference signal is used to determine which RBM power range setpoint (low, intermediate, or high) is enabled. The RBM is automatically bypassed and the output set to zero if a peripheral rod is selected or the APRM used to normalize the RBM reading is < 28% RTP. If any LPRM detector assigned to an RBM is bypassed, the computed average signal is automatically adjusted to compensate for the number of LPRM input signals. The quantity of LPRM detectors in the RBM average flux may vary from a minimum of two to a maximum of eight depending upon the control rod selected and the number of bypassed LPRM detectors.

The Functional Computer calculates a new value of RBM gain each time a new control rod is selected. The gain setting is held constant during the movement of that particular control rod to provide an indication of the change in the relative local power level. If the indicated power increases above the RBM power range setpoint (low, intermediate, or high), a rod block will occur. In addition, to preclude rod movement with an inoperable RBM, an inoperable trip is provided.

BASES (continued)

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

1. Rod Block Monitor

The RBM is designed to prevent violation of the MCPR SL and the cladding 1% plastic strain fuel design limit that may result from a single control rod withdrawal error (RWE) event. The analytical methods and assumptions used in evaluating the RWE event are summarized in Reference 8. The cycle-specific analysis considers the continuous withdrawal of the maximum worth control rod at its maximum drive speed from the reactor, which is operating at rated power with a control rod pattern that results in the core being placed on thermal design limits. The condition is analyzed to ensure that the results obtained are conservative; the approach also serves to demonstrate the function of the RBM.

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Insert B9

The RBM Function satisfies Criterion 3 of Reference 3.

Insert B10

Two channels of the RBM are required to be OPERABLE, with their setpoints within the appropriate Allowable Values to ensure that no single instrument failure can preclude a rod block from this Function. The actual setpoints are calibrated consistent with applicable setpoint methodology.

Nominal trip setpoints are specified in the setpoint calculations. The nominal setpoints are selected to ensure that the setpoints do not exceed the Allowable Values between successive CHANNEL CALIBRATIONS. Operation with a trip setpoint less conservative than the nominal trip setpoint, but within its Allowable Value, is acceptable. Trip setpoints are those predetermined values of output at which an action should take place. The setpoints are compared to the actual process parameter (e.g., reactor power), and when the measured output value of the process parameter exceeds the setpoint, the associated device (e.g., trip unit) changes state. The analytic limits are derived from the limiting values of the process parameters obtained from the safety analysis. The Allowable Values are derived from the analytic limits by accounting for calibration uncertainty, process measurement uncertainty, primary element uncertainty, instrument uncertainty, and applicable environmental effects. The trip setpoints are derived from the analytical limits by accounting for calibration uncertainty, process measurement uncertainty, primary element uncertainty, instrument uncertainty, applicable environmental effects, and drift. The trip setpoints are also derived from the Allowable Values in the conservative direction by considering calibration uncertainty, instrument uncertainty, environmental effects, and drift. The most

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INSERT B9

A statistical analysis of RWE events was performed to determine the RBM response for both channels for each event. From these responses, the fuel thermal performance as a function of RBM Allowable Value was determined. The Allowable Values are chosen as a function of power level. Based on the specified Allowable Values, operating limits are established.

INSERT B10

for the associated power range,

INSERT B11

The setpoints are compared to the actual process parameter (e.g., APRM Simulated Thermal Power), and when the measured output value of the process parameter exceeds the setpoint, the associated device (e.g., trip unit) changes state. The analytical limits are derived from the limiting values of the process parameters obtained from the safety analysis. The Allowable Values are derived from the analytical limits by accounting for calibration uncertainty, process measurement uncertainty, primary element uncertainty, instrument uncertainty, and applicable environmental effects.

The analytical limits are derived from the limiting values of the process parameters. Using the GE setpoint methodology, based on ISA RP 67.04, Part II, "Methodologies for the Determination of Setpoints for Nuclear Safety-Related Instrumentation," setpoint calculation Method 2, the RBM Allowable Values are determined from the analytical limits using the statistical combination of RBM input signal calibration error, process measurement error, primary element accuracy and instrument accuracy under trip conditions. Accounting for these errors assures that a setpoint found during calibration at the Allowable Value has adequate margin to protect the analytical limit, thereby protecting the Safety Limit.

For the digital RBM, there is a normalization process initiated upon rod selection, so that only RBM input signal drift over the interval from the rod selection to rod movement needs to be considered in determining the nominal trip setpoints. The RBM has no drift characteristic with no as-left or as-found tolerances since it only performs digital calculations on the digitized input signals provided by the APRMs.

The Allowable Value is the Limiting Safety System Setting since the RBM has no drift characteristic. The RBM Allowable Value demonstrates that the analytical limit would not be exceeded, thereby protecting the Safety Limit. The trip setpoints and Allowable Values determined in this manner provide adequate protection because instrumentation uncertainties, process effects, calibration tolerances, instrument drift, and environmental errors are accounted for and appropriately applied for the RBM. There are no margins applied to the RBM nominal trip setpoint calculations which could mask RBM degradation.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

1. Rod Block Monitor (continued)

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~~conservatively derived trip setpoints are used. In addition, both the Allowable Values and trip setpoints may have additional conservatisms.~~

Insert B12 →

~~The RBM is assumed to mitigate the consequences of an RWE event when operating ≥ 30% RTP and a peripheral control rod is not selected. Below this power level, or if a peripheral control rod is selected, the consequences of an RWE event will not exceed the MCPR SL and, therefore, the RBM is not required to be OPERABLE (Ref. 4).~~

2. Rod Worth Minimizer

The RWM enforces the banked position withdrawal sequence (BPWS) to ensure that the initial conditions of the CRDA analysis are not violated. The analytical methods and assumptions used in evaluating the CRDA are summarized in Reference 5. The BPWS requires that control rods be moved in groups, with all control rods assigned to a specific group required to be within specified banked positions. Requirements that the control rod sequence is in compliance with the BPWS are specified in LCO 3.1.6, "Rod Pattern Control."

The RWM Function satisfies Criterion 3 of Reference 3.

Since the RWM is a system designed to act as a backup to operator control of the rod sequences, only one channel of the RWM is available and required to be OPERABLE (Ref. 6). Special circumstances provided for in the Required Action of LCO 3.1.3, "Control Rod OPERABILITY," and LCO 3.1.6 may necessitate bypassing the RWM to allow continued operation with inoperable control rods, or to allow correction of a control rod pattern not in compliance with the BPWS. The RWM may be bypassed as required by these conditions, but then it must be considered inoperable and the Required Actions of this LCO followed.

Compliance with the BPWS, and therefore OPERABILITY of the RWM, is required in MODES 1 and 2 when THERMAL POWER is ≤ 10% RTP. When THERMAL POWER is > 10% RTP, there is no possible control rod configuration that results in a control rod worth that could exceed the 280 cal/gm fuel damage limit during a CRDA (Ref. 5). In MODES 3 and 4, all control rods are required to be inserted into the core; therefore, a CRDA

(continued)

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The RBM is assumed to mitigate the consequences of an RWE event when operating ≥ 28% RTP and a peripheral control rod is not selected. Below this power level, or if a peripheral control rod is selected, the consequences of an RWE event will not exceed the MCPR SL and, therefore, the RBM is not required to be OPERABLE. When operating < 90% RTP, analyses have shown that with an initial MCPR ≥ a cycle dependent value specified in the COLR, no RWE event will result in exceeding the MCPR SL. Also, the analyses demonstrate that when operating at ≥ 90% RTP with MCPR ≥ a second cycle dependent value specified in the COLR, no RWE event will result in exceeding the MCPR SL (Ref. 3). Therefore, under these conditions, the RBM is also not required to be OPERABLE (Refs. 4 and 9).

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.2.1.1 and SR 3.3.2.1.2 (continued)

allowance is based on operating experience and in consideration of providing a reasonable time in which to complete the SRs. Operating experience has shown that these components usually pass the Surveillance when performed at the 92 day Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

SR 3.3.2.1.3

A CHANNEL FUNCTIONAL TEST is performed for each RBM channel to ensure that the entire channel will perform the intended function. It includes the Reactor Manual Control Multiplexing System input.

Any setpoint adjustment shall be consistent with the assumptions of the current plant specific setpoint methodology. The Frequency of 184 days is based on the analysis in Reference 8.

SR 3.3.2.1.4

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~~The RBM is automatically bypassed when power is below a specified value or if a peripheral control rod is selected. The power level is determined from the APRM signals input to each RBM channel. The automatic bypass setpoint must be verified periodically to be < 30% RTP. In addition, it must also be verified that when ≥ 30% RTP, the RBM is not bypassed when a control rod that is not a peripheral control rod is selected (only one non-peripheral control rod is required to be verified). If any bypass setpoint is nonconservative, then the affected RBM channel is considered inoperable. Alternatively, the APRM channel can be placed in the conservative condition to enable the RBM. If placed in this condition, the SR is met and the RBM channel is not considered inoperable. As noted, neutron detectors are excluded from the Surveillance because they are passive devices, with minimal drift, and because of the difficulty of simulating a meaningful signal. Neutron detectors are adequately tested in SR 3.3.1.1.3 and SR 3.3.1.1.7. The 24 month Frequency is based on the analysis in Reference 8.~~

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The RBM setpoints are automatically varied as a function of power. Three Allowable Values are specified in Table 3.3.2.1-1, each within a specific power range. The power at which the control rod block Allowable Values automatically change are based on the APRM signal's input to each RBM channel. Below the minimum power setpoint or if a peripheral control rod is selected, the RBM is automatically bypassed. These power Allowable Values must be verified periodically to be less than or equal to the specified values. If any power range setpoint is nonconservative, then the affected RBM channel is considered inoperable. Alternatively, the power range channel can be placed in the conservative condition (i.e., enabling the proper RBM setpoint). If placed in this condition, the SR is met and the RBM channel is not considered inoperable. As noted, neutron detectors are excluded from the Surveillance because they are passive devices, with minimal drift, and because of the difficulty of simulating a meaningful signal. Neutron detectors are adequately tested in SR 3.3.1.1.3 and SR 3.3.1.1.7. The 24 month Frequency is based on the analysis in Reference 8.

BASES

REFERENCES (continued)

7. GENE-770-06-1-A, "Addendum To Bases For Changes To Surveillance Test Intervals And Allowed Out-of-Service Times For Selected Instrumentation Technical Specifications," December 1992.

8. NEDC-32410-P-A, "Nuclear Measurement Analysis and Control Power Range Neutron Monitor (NUMAC-PRNM) Retrofit Plus Option III Stability Trip Function," October 1995.

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INSERT B14

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

BASES

APPLICABLE SAFETY ANALYSES (continued)

The recirculation system is also assumed to have sufficient flow coastdown characteristics to maintain fuel thermal margins during abnormal operational transients (Ref. 3), which are analyzed in Chapter 15 of the USAR.

A plant specific LOCA analysis has been performed assuming only one operating recirculation loop. This analysis has demonstrated that, in the event of a LOCA caused by a pipe break in the operating recirculation loop, the Emergency Core Cooling System response will provide adequate core cooling, provided the APLHGR requirements are modified accordingly (Ref. 4).

The transient analyses in Chapter 15 of the USAR have also been performed for single recirculation loop operation (Ref. 4) and demonstrate sufficient flow coastdown characteristics to maintain fuel thermal margins during the abnormal operational transients analyzed provided the MCPR requirements are modified. During single recirculation loop operation, modification to the Reactor Protection System average power range monitor (APRM) and the Rod Block Monitor Allowable Values is also required to account for the different relationships between recirculation drive flow and reactor core flow. The APLHGR and MCPR limits for single loop operation are specified in the COLR. The APRM Flow Biased Simulated Thermal Power—Upscale Allowable Value is in LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation." The Rod Block Monitor—Upscale Allowable Value is in LCO 3.3.2.1, "Control Rod Block Instrumentation."

Recirculation loops operating satisfies Criterion 2 of Reference 5.

LCO

Two recirculation loops are normally required to be in operation with their flows matched within the limits specified in SR 3.4.1.1 to ensure that during a LOCA caused by a break of the piping of one recirculation loop the assumptions of the LOCA analysis are satisfied. Alternatively, with only one recirculation loop in operation, modifications to the required APLHGR limits (LCO 3.2.1, "AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)"), MCPR limits (LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)"), APRM Flow Biased Simulated Thermal Power—Upscale Allowable Value (LCO 3.3.1.1), and the Rod

and

(continued)

BASES

LCO (continued)

Block Monitor—Upscale Allowable Value (LCO 3.3.2.1) must be applied to allow continued operation consistent with the assumptions of Reference 4.

APPLICABILITY

In MODES 1 and 2, requirements for operation of the Reactor Coolant Recirculation System are necessary since there is considerable energy in the reactor core and the limiting design basis transients and accidents are assumed to occur.

In MODES 3, 4, and 5, the consequences of an accident are reduced and the coastdown characteristics of the recirculation loops are not important.

ACTIONS

A.1 and A.2

With no recirculation loops in operation, the unit is required to be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to MODE 2 within 6 hours and to MODE 3 within 12 hours. In this condition, the recirculation loops are not required to be operating because of the reduced severity of DBAs and transients and minimal dependence on the recirculation loop coastdown characteristics. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

B.1 and C.1

With both recirculation loops operating but the flows not matched, the flows must be matched within 2 hours. If matched flows are not restored, the recirculation loop with lower flow must be declared "not in operation," as required by Required Action B.1. This Required Action does not require tripping the recirculation pump in the lowest flow loop when the mismatch between total jet pump flows of the two loops is greater than the required limits. However, in cases where large flow mismatches occur, low flow or reverse flow can occur in the low flow loop jet pumps, causing vibration of the jet pumps. If zero or reverse flow is detected, the condition should be alleviated by changing flow control valve position to re-establish forward flow or by tripping the pump.

(continued)

ATTACHMENT (4)

REVISIONS TO PLANT-SPECIFIC EVALUATIONS REQUIRED BY NUMAC PRNM RETROFIT PLUS OPTION III STABILITY TRIP FUNCTION TOPICAL REPORT (NEDC-32410P-A) FOR ARTS IMPLEMENTATION

Nine Mile Point Nuclear Station, LLC
March 30, 2007

A4-1. EVALUATION OF NMP2 NUCLEAR MEASUREMENT ANALYSIS AND CONTROL (NUMAC) POWER RANGE NEUTRON MONITOR COMPARED TO THE NUMAC LICENSING TOPICAL REPORT SECTIONS

The NUMAC Power Range Neutron Monitoring System (PRNMS) was initially installed at Nine Mile Point Unit 2 (NMP2) in 1998 (References 1 and 2). The initial installation included a "non-ARTS" version of the Rod Block monitor (RBM).

This license amendment request to implement Average Power Range Monitor/Rod Block Monitor/Technical Specifications/Maximum Extended Load Line Limit Analysis (ARTS/MELLLA) describes the equipment and Technical Specifications (TS) changes that are different from the configuration described in the prior submittal configuration, i.e., a NUMAC PRNMS including the ARTS logic. To support the NMP2 plan for Extended Power Uprate, the ARTS/MELLLA implementation is scheduled during the NMP2 Spring 2008 outage.

The fundamental logic and setpoint changes to implement ARTS and supporting analyses and justifications are covered in Attachment (7) of this submittal. The NUMAC PRNM equipment and system, as described in the NUMAC PRNM Retrofit Plus Option III Stability Trip Function Topical Report NEDC-32401P-A, including Supplement 1 (References 3 & 4), and previously reviewed and approved by the NRC, is designed to handle, with minor hardware modification, ARTS RBM logic. The Reference 1 submittal specifically discussed applicability of the NUMAC topical reports to the non-ARTS configuration as applied at NMP2. This attachment addresses the NMP2 changes in the NUMAC topical report applicability resulting from changing from non-ARTS to ARTS logic.

The implementation of ARTS logic in the NUMAC PRNMS will be managed as a change from the previously completed non-ARTS NUMAC PRNMS. All software changes necessary will undergo full verification and validation activities equivalent to those performed for the previous installation. The specific equipment changes necessary are:

a. Replacement of the firmware in the two RBM channels, specifically in the two RBM chassis, to remove the non-ARTS flow-biased RBM logic and replace it with the power-based trip logic. This involves changing the basic trip logic plus the user interface (user display) to provide for different types of setpoints (power dependent vs. flow-biased) and minor changes to the readouts. The basic ARTS logic for NMP2 is the same as that previously applied at several boiling water reactors (BWRs) with currently installed NUMAC PRNM systems. The change is accomplished by replacing the currently installed plug-in firmware (memory chips) with new ones on two modules in each of the two RBM chassis.

b. Disconnecting and disabling two RBM "push to set-up" switches, one per RBM channel, and eight associated status lights, four for each RBM channel. These switches and associated status lights, which allow the operator to manually "set-up" the rod block limit in the current non-ARTS RBM logic, are not used in the ARTS logic. This change is accomplished by disconnecting the signal from the RBM chassis and either removing the unused equipment or marking it as not used.

c. Installing two jumpers in the PRNM panel, one in each rod block circuit, to permanently bypass (remove from the logic) the recirculation flow comparison rod block signal. As described in the NUMAC PRNM topical reports, the recirculation flow comparison rod block function is not required for the ARTS RBM.

d. Modify the Multi-Vendor Data Acquisition System (MVD) as necessary to reflect the power-based instead of flow-biased RBM setpoints, the status of which is transmitted from the PRNM. The MVD is the interface between the NUMAC PRNMS and the plant computer.

e. Modify slightly the process computer data base to reflect the power-based instead of flow-biased setpoints.

f. Update the APRM simulated thermal power (STP) flow-biased reactor protection system (RPS) trip and rod block setpoints to reflect the ARTS limits, and install the ARTS RBM setpoints.

Required changes to the TSs are as outlined in Attachment (2) to this submittal.

The previous NMP2 NUMAC PRNM submittal included the NMP2-specific responses to all "Utility Actions Required" items in the NUMAC PRNM Retrofit Plus Option III Stability Trip Function Topical Report NEDC-32410P-A, including Supplement 1. Those responses remain unchanged for the currently-installed PRNMS. The following previous NMP2 utility action responses have been revised to incorporate responses for the proposed change to ARTS. In the following table, the section numbers and Utility Action Required identified are consistent with the initial submittal and the Topical Report. In addition to the NMP2-specific information, the table also includes additional justification information where the Topical Report does not specifically cover the NMP2 configuration. Only responses that are changed from those included in the prior submittal are included.

Section No.	Utility Action Required	Response
2.3.4	Plant Unique or Plant-Specific Aspects Confirm that the actual plant configuration is included in the variations covered in the Power Range Neutron Monitor (PRNM) Licensing Topical Report (LTR) [NEDC-32410P-A, Volumes 1 & 2 and Supplement 1], and the configuration alternative(s) being applied for the replacement PRNM are covered by the PRNM LTR. Document in the *plant-specific licensing submittal* for the PRNM project the actual current plant configuration and the configuration of the replacement PRNM, and document confirmation that those are covered by the PRNM LTR. For any changes to the plant operator's panel, document in the submittal the human factors review actions that were taken to confirm compatibility with existing plant commitments and procedures.	The current plant configuration and the modification to the PRNM to implement the ARTS logic are included in the PRNM LTR as follows: (Applicable LTR sections are listed.) No change for ARTS addition: Current APRM 2.3.3.1.2 RBM 2.3.3.2.2 Flow Unit 2.3.3.3.2 Rod Control 2.3.3.4.2 Panel Interface 2.3.3.6.2 For this modification: Current / Proposed ARTS 2.3.3.5.1 / 2.3.3.5.2 Changes made to the plant operator's panel will be reviewed to ensure compliance with the NMP human factors manual, "Human Factors Manual for Future Control Room Design Changes," and documented on a Design Input Checklist. This manual is based, in general, on NUREG-0700, Guidelines for Control Room Design Reviews.
3.4	System Functions As part of the *plant-specific licensing submittal*, the utility should document the following: 1) The pre-modification flow channel configuration, and any changes planned (normally changes will be either adding two channels to reach four or no change planned) NOTE: If transmitters are added, the requirements on the added transmitters	1) There are no changes to the flow channels for this modification.

Section No.	Utility Action Required	Response
	should be: ▪ Non-safety related, but qualified environmentally and seismically to operate in the application environment. ▪ Mounted with structures equivalent or better than those for the currently installed channels. ▪ Cabling routed to achieve separation to the extent feasible using existing cableways and routes. 2) Document the APRM trips currently applied at the plant. If different from those documented in the PRNM LTR, document plans to change to those in the LTR. 3) Document the current status related to ARTS and the planned post modification status as: ▪ ARTS currently implemented, and retained in the PRNM ▪ ARTS will be implemented concurrently with the PRNM (reference ARTS submittal) ▪ ARTS not implemented and will not be implemented with the PRNM ▪ ARTS not applicable	2) There are no changes to the APRM trips. However, as part of the change to ARTS/MELLLA, the Allowable Value and setpoints for the "Flow-Biased Simulated Thermal Power - Upscale" will be revised. 3) ARTS is not currently implemented. The ARTS logic is implemented by the proposed change. ARTS will be implemented via replacement of NUMAC RBM EPROMs and minor plant wiring changes. NMP2 TS 3.3.2.1 will be modified to be similar to that shown in the PRNM LTR, Volume 2, Section H.1.1, except that RBM Downscale, Function 1.e, will not be included. (See additional discussion and justification in the responses to LTR Section 8.5.1.4 and in Section A4-2 following this table.)
7.6	<u>Impact on FSAR</u> The plant-specific action required for FSAR updates will vary between plants. In all cases, however, existing FSAR documents should be reviewed to identify areas that have descriptions specific to the current PRM using the general guidance of Sections 7.2 through 7.5 of the PRNM LTR to identify potential areas impacted. The utility should include in the *plant-specific licensing submittal* a statement of the plans for updating the plant FSAR for the PRNM project.	Applicable sections of the FSAR (Updated Safety Analysis Report – USAR for NMP2) will be reviewed and appropriate revisions of those sections will be prepared and approved as part of the normal design process. Following implementation of the design modification and closure of the design package, the USAR will be revised as part of the routine USAR update in accordance with 10 CFR 50.71(e).

Section No.	Utility Action Required	Response
8.3.6.1	APRM-Related RPS Setpoints Covered by Tech Specs Add to or delete from the appropriate document any changed RPS setpoint information. If ARTS is being implemented concurrently with the PRNM modification, either include the related Tech Spec submittal information with the PRNM information in the plant-specific submittal, or reference the ARTS submittal in the PRNM submittal. In the *plant-specific submittal*, identify what changes, if any, are being implemented and identify the basis or method used for the calculation of setpoints and where the setpoint information or changes will be recorded.	Only the APRM Flow-Biased Simulated Thermal Power Allowable Value for two-loop operation is affected by the proposed change. The Allowable Value will be included in the TSs and is comparable to what is currently in the NMP2 TSs. See the NMP2 TS markups in Attachment (2) of this submittal for the specific changes.
8.5.1.4	APRM-Related Control Rod Block Functions-Functions Covered by Tech Specs If ARTS will be implemented concurrently with the PRNM modification, include or reference those changes in the *plant-specific PRNM submittal.* Implement the applicable portion of the above described changes via modifications to the Tech Specs and related procedures and documents. In the *plant-specific submittal*, identify functions currently in the plant Tech Specs and which, if any, changes are being implemented. For any functions deleted from Tech Specs, identify where setpoint and surveillance requirements will be documented. NOTE: A utility may choose not to delete some or all of the items identified in the PRNM LTR from the plant Tech Specs.	The proposed change replaces the flow-biased RBM rod blocks with power-based rod blocks. To implement this change, the RBM rod block Limiting Condition for Operation (LCO) 3.3.2.1 Function 1 is modified as follows: Current RBM rod block functions: 1.a Upscale 1.b Inop 1.c Downscale For the proposed change, the following functions will replace the current RBM functions: 1.a Low Power Range - Upscale 1.b Intermediate Power Range - Upscale 1.c High Power Range - Upscale 1.d Inop The selection of setpoints in the ARTS logic in the RBM is based on APRM STP. This change reduces the risk of spurious rod block signals and assures a clean transition between RBM setpoints as power increases or decreases.

Section No.	Utility Action Required	Response
		The proposed TS changes to the RBM Functions are consistent with those shown in the LTR except for deletion of the RBM Downscale Function. With the implementation of the ARTS logic in the RBM, the Allowable Values for the RBM setpoints will be located in the Core Operating limits Report (COLR), rather than in LCO 3.3.2.1. This change is being made because the RBM power dependent setpoints must be reconfirmed or modified on a cycle-specific basis. In addition, the surveillance and operability requirements for each RBM "power range" Function will be modified from that shown in the PRNM LTRs (for ARTS) by revision to the notes to TS Table 3.3.2.1-1 and SR 3.3.2.1.4. The deletion of the RBM Downscale Function is intended to simplify the TS by deleting a Function that has no significant value due to differences between the original analog equipment and the replacement digital system. [Note: See justification in Section A4-2 following this table.] The surveillance (SR) and operability requirements for each RBM power range are being modified from those shown in the LTR to clarify that the SR and operability requirements do not apply when a peripheral control rod is selected. This is a current feature of the RBM that is not being modified by the proposed changes. The SR and operability requirements are also written based on APRM STP input, the digital signal that is actually used in the NUMAC RBM. These additional SR and operability requirements clarifications are consistent with the PRNM LTR and result in no functional changes in the equipment performance or

Section No.	Utility Action Required	Response
		operational limits. See the NMP2 TS and Bases markups (Attachments 2 and 3 to this submittal) for the specific changes.
8.5.4.1.4	<u>APRM-Related Control Rod Block Functions – Required Surveillances and Calibration – Channel Check</u> Delete any requirements for instrument or channel checks related to RBM and, where applicable, recirculation flow rod block functions (non-ARTS plant), and APRM functions. Identify in the plant-specific PRNM submittals if any checks are currently included in Tech Specs, and confirm that they are being deleted.	Consistent with the PRNM LTRs, the proposed change replaces the current SR 3.3.2.1.4 requirement, which addresses only a single operability lower limit, with an SR that addresses the operability of the three power level trips in the ARTS RBM logic. See the NMP2 TS and Bases markups (Attachments 2 and 3 to this submittal) for the specific changes.
8.5.6.1	<u>APRM-Related Control Rod Block Function – Required Surveillances and Calibration – Setpoints</u> Add to or delete from the appropriate document any changed control rod block setpoint information. If ARTS is being implemented concurrently with the PRNM modification, either include the related Tech Spec submittal information with the PRNM information in the *plant-specific submittal,* or reference the ARTS submittal in the PRNM submittal. In the *plant-specific submittal*, identify what changes, if any, are being implemented and identify the basis or method used for calculation of setpoints and where the setpoint information or changes will be recorded.	The proposed change implements ARTS/MELLLA. RBM Allowable Values (AVs) are modified to reflect the ARTS limits. With the implementation of ARTS logic in the RBM, the AVs for the RBM will be located in the COLR, rather than TS Table 3.3.2.1-1 to allow for these values to be modified on a cycle specific basis as needed. Similarly, the RBM related setpoints for the power level Minimum Critical Power Ratio (MCPR) limits will be located in the COLR rather than TS Table 3.3.2.1-1, as shown in the PRNM LTRs, to allow these values to be modified on a cycle specific basis, as needed. See the NMP2 TS and Bases markups (Attachments 2 and 3 to this submittal) for the specific changes.
None	<u>Core Operating Limits Report</u> Reporting requirements Section 5.6.5 do not currently address the MCPR conditions for RBM Upscale Functions.	Requirements for RBM power level Allowable Values and MCPR conditions are added in TS 5.6.5a with reference to LCO 3.3.2.1

Section No.	Utility Action Required	Response
		See the NMP2 TS and Bases markups (Attachments 2 and 3 to this submittal) for the specific changes.
9.1.3	Utility Quality Assurance Program As part of the *plant-specific licensing submittal*, the utility should document the established program that is applicable to the project modification. The submittal should also document for the project what scope is being performed by the utility and what scope is being supplied by others. For scope supplied by others, document the utility actions taken or planned to define or establish requirements for the project, to assure those requirements are compatible with the plant-specific configuration. Actions taken or planned by the utility to assure compatibility of the GE quality program with the utility program should also be documented. Utility planned level of participation in the overall V&V process for the project should be documented, along with utility plans for software configuration management and provision to support any required changes after delivery should be documented.	Quality assurance requirements for work performed at the Nine Mile Point Nuclear Station (NMPNS) are defined and described in the Constellation Generation Group (CGG) Quality Assurance Topical Report (QATR). This document describes the planning, implementation, and organizational process for the NMPNS Quality Verification Inspection Program. This program verifies that services and activities affecting safety meet established requirements and conform to applicable documented instructions, procedures and drawings. This includes Quality Oversight of approved vendor activities at CGG nuclear facilities. For the ARTS modification to the PRNM equipment, NMPNS has contracted with General Electric (GE) to include the following PRNM scope: 1) design, 2) hardware/software, 3) licensing support, 4) training, 5) O&M manuals and design documentation, and 6) NMP2 setpoint calculation inputs. On-site engineering work to incorporate the GE-provided design information into a Design Change Package (DCP) or provide supporting, interface DCPs will be performed per the requirements of applicable NMPNS procedures. Modification work to implement the DCPs will be performed per NMPNS procedures or NMPNS-approved contractor procedures. NMPNS has participated and will continue to participate in appropriate reviews of GE's design and verification and validation (V&V) program for the PRNM modification. For software delivered in the form of

Section No.	Utility Action Required	Response
		hardware (EPROMs), NMPNS intends to have GE maintain post delivery configuration control of the actual source code and handle any changes. NMPNS will then handle any changes in the EPROMs as hardware changed under its applicable hardware modification procedures. All changes required to implement the ARTS modification will undergo the same level of V&V as the initial PRNM design described in the prior submittal (Reference 1).

A4-2. JUSTIFICATION FOR DELETION OF ROD BLOCK MONITOR DOWNSCALE, SPECIFICATION 3.3.2.1 (IMPLEMENTATION OF ARTS)

(Ref. the paragraph 3.4 and 8.5.1.4 responses above)

The effect of the differences between analog equipment and the digital equipment on the RBM Downscale Function was not addressed at the time the NUMAC PRNM LTR was prepared, so this deletion was not addressed in the LTR.

The originally intended RBM Downscale Function would detect substantial reductions in the RBM local flux after a "null" is completed (a "null" occurs after a new rod selection). This function, in combination with the RBM Inop Function, was intended in the original system to detect problems with or abnormal conditions in the RBM equipment and system. However, no credit is taken for the RBM Downscale Function in the establishment of the RBM upscale trip setpoints or Allowable Values.

Unlike other neutron monitoring system downscale functions (e.g., the APRM downscale), there are no normal operating conditions that are intended to be detected by the RBM Downscale Function. In an analog RBM, the inclusion of the Downscale Function in addition to the Inop Function had some merit in that the analog equipment had some failure modes that could result in a reduction of signal, but not a full failure. Therefore, the RBM Downscale Function was in fact part of the overall Inop condition detection function.

The replacement of the original analog RBM equipment with the NUMAC digital RBM, which was accomplished with the installation covered by the Reference 1 submittal, resulted in all of the original analog processing being replaced by digital processing. One effect of this change is to eliminate the types of failures that can reasonably be detected by a Downscale Function. In addition, the Inop Function is enhanced in the NUMAC RBM by the use of automatic self-test and other internal logic to increase the detectability of failures and abnormal conditions that can occur in the digital equipment, and to directly include these in the RBM Inop Function.

Therefore, in the NUMAC ARTS RBM, there is no incremental value or benefit provided by the RBM Downscale Function. Consistent with the overall thrust of the Improved TSs to eliminate "no value" requirements, the RBM Downscale Function is being removed from the Technical Specifications and from the related discussion in the Bases. The RBM Inop Function is being retained in Technical Specifications.

A.4-3 REFERENCES

1. Letter from B. R. Sylvia (Niagara Mohawk Power Corporation) to Document Control Desk (NRC), dated October 31, 1997, License Amendment Request to Use NUMAC Power Range Neutron Monitor System (PRNM)

2. Letter from D. S. Hood, (NRC) to J. H. Mueller (Niagara Mohawk Power Corporation) dated March 31, 1998, Issuance of Amendment for Nine Mile Point Nuclear Station, Unit No. 2 (TAC No. MA0150)

3. Licensing Topical Report NEDC-32410P-A Volumes 1 and 2, "Nuclear Measurement Analysis and Control Power Range Neutron Monitor (NUMAC-PRNM) Retrofit Plus Option III Stability Trip Function," dated October 1995.

4. Licensing Topical Report NEDC-32410P-A Supplement 1, "Nuclear Measurement Analysis and Control Power Range Neutron Monitor (NUMAC-PRNM) Retrofit Plus Option III Stability Trip Function," dated November 1997.