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 Attachments 11, 13.1, 13.2, 13.3 & 13.4.

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Constellation Energy
Nine Mile Point Nuclear Station

May 27, 2009

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 (LAR) Pursuant to 10 CFR 50.90: Extended Power Uprate

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 (OL) NPF-69. The proposed amendment would increase the power level authorized by OL Section 2.C.(1), Maximum Power Level, from 3467 megawatts-thermal (MWt) to 3988 MWt. The new maximum power level represents an increase of 20 percent from the Original Licensed Thermal Power level of 3323 MWt and an increase of 15 percent from the Current Licensed Thermal Power level of 3467 MWt. NMP2 Amendment No. 66, dated April 28, 1995, approved a Stretch Power Uprate authorizing the increase from 3323 MWt to 3467 MWt.

The Enclosure and its associated Attachments to this application provide the evaluation of the proposed changes to NMP2 OL Section 2.C.(1) and other affected license conditions and Technical Specifications. As indicated in the Enclosure, NMPNS has concluded that the activities associated with the proposed changes represent no significant hazards consideration under the standards set forth in 10 CFR 50.92.

NMPNS requests approval of this application in 18 months with implementation upon startup from the spring 2012 refueling outage. This submittal contains no regulatory commitments.

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

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 Attachments 11, 13.1, 13.2, 13.3 & 13.4.

A001

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

Very truly yours,

STATE OF NEW YORK :

: **TO WIT:**

COUNTY OF OSWEGO :

I, Keith J. Polson, being duly sworn, state that I am Vice President-Nine Mile Point, and that I am duly authorized to execute and file this License Amendment 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 27 day of May, 2009.

WITNESS my Hand and Notarial Seal:

Lisa M. Clark
Notary Public

My Commission Expires:

9/12/09
Date

LISA M. CLARK
Notary Public in the State of New York
Oswego County Reg. No. 01CL6029220
My Commission Expires 9/12/09

KJP/KHJ

Enclosure: Evaluation of the Proposed Change

cc: NRC Regional Administrator, Region I
NRC Resident Inspector
NRC Project Manager
NYSERDA (w/o Attachments 11, 13.1, 13.2, 13.3 and 13.4 of Enclosure)

ENCLOSURE

EVALUATION OF THE PROPOSED CHANGE

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ENCLOSURE

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ATTACHMENTS

EVALUATION OF THE PROPOSED CHANGE

1.0 SUMMARY DESCRIPTION

The Nine Mile Point Unit 2 (NMP2) Operating License (OL) specifies the maximum power level at which NMP2 may be operated. The proposed amendment would increase the maximum power level authorized from 3467 megawatts-thermal (MWt) to 3988 MWt. The new maximum power level represents an increase of 20 percent from the Original Licensed Thermal Power (OLTP) level of 3323 MWt and an increase of 15 percent from the Current Licensed Thermal Power (CLTP) level of 3467 MWt. Nine Mile Point Unit 2 Amendment No. 66 dated April 28, 1995, approved a Stretch Power Uprate authorizing the increase from 3323 MWt to 3467 MWt. Approval of the proposed amendment will allow Nine Mile Point Nuclear Station (NMPNS) to implement the operational and plant configuration changes necessary to generate and supply a higher steam flow to the turbine-generator. The higher steam flow will enable NMP2 to increase its gross rated generator output from 1211 megawatts-electric (MWe) to 1369 MWe. The current Extended Power Uprate (EPU) implementation plan consists of a phased approach to power increase and installation of plant modifications. A limited number of EPU related modifications were completed during the NMP2 2008 Refueling Outage with the remainder of the modifications scheduled to be completed during the 2010 and 2012 refueling outages. Power will not be increased until all required modifications are completed. Reactor Recirculation System (RCS) modifications or additional system maintenance may be performed during the 2014 refueling outage to optimize plant performance at EPU conditions, but are not necessary prior to power increase.

Technical Specifications (TS) Amendment No. 123, dated February 27, 2008, expanded the NMP2 operating domain by implementing Average Power Range Monitor / Rod Block Monitor / Technical Specifications / Maximum Extended Load Line Limit Analysis (ARTS/MELLLA). The ARTS/MELLLA Amendment expanded the power-to-flow map's operating domain to allow for plant operation at an increased thermal power level. TS Amendment No.125, dated May 29, 2008, permits full implementation of the Alternative Source Term (AST) as described in Regulatory Guide 1.183, Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors. The AST Evaluation was performed at the proposed EPU power level so that the Design Basis Accident analyses could accommodate this EPU submittal.

Attachment 1 to this Enclosure provides the marked-up OL and TS pages showing the proposed changes. Associated TS Bases changes are shown in Attachment 2. The Bases changes are provided for information only and will be processed in accordance with the NMP2 TS Bases Control Program as described in TS 5.5.10.

NEDO-33351, "Safety Analysis Report for Nine Mile Point Nuclear Station Unit 2 Constant Pressure Power Uprate (PUSAR)" (non-proprietary version), is provided as Attachment 3. This report provides an integrated summary of the results of the safety analyses and evaluations performed that support the proposed increase in the maximum power level at NMP2. The PUSAR safety evaluation follows the format and guidance delineated in RS-001 (Revision 0), Office of Nuclear Reactor Regulation, "Review Standard for Extended Power Uprates," to the extent that the review standard is consistent with the design basis of NMP2. For differences between the plant-specific design bases and RS-001 regulatory evaluation sections, the corresponding PUSAR safety evaluation regulatory evaluation section was revised to reflect the NMP2 design basis. As appropriate, the PUSAR's technical evaluations are based on NRC approved topical report NEDC-33004P-A, "Licensing Topical Report Constant Pressure Power Uprate," Revision 4 (CLTR).

A proprietary version of the PUSAR is provided as Attachment 11. This attachment is considered by GE-Hitachi Nuclear Energy (GEH) to contain proprietary information exempt from disclosure pursuant to 10 CFR 2.390. Therefore, on behalf of GEH, NMPNS hereby makes application to withhold this document from public disclosure in accordance with 10 CFR 2.390(b)(1). An affidavit executed by GEH detailing the reasons for the request to withhold the proprietary information is provided in Attachment 4.

Attachment 5 is provided for the regulatory commitments associated with the proposed change. Because no commitments associated with this submittal were identified, this attachment has been retained as a placeholder only.

Plant modifications associated with EPU and their implementation schedule are listed in Attachment 6, Modifications to Support EPU. The EPU Test Plan, Attachment 7, specifies the planned EPU testing activities including a comparison to the initial NMP2 start-up testing program. The Grid Stability Evaluation, provided as Attachment 8, evaluates the impact of EPU on the transmission system stability. NMPNS's assessment of the environmental impacts of the proposed EPU is contained in Attachment 9, Supplemental Environmental Report. This Report was prepared pursuant to 10 CFR 51.41, "Regulations to Submit Environmental Information."

Attachment 10, Flow Induced Vibration (FIV) Piping / Component Evaluation, provides a review of plant system piping and components potentially affected by FIV under EPU conditions.

Attachment 13, Steam Dryer Evaluation, provides an evaluation and validation of the structural adequacy of the NMP2 steam dryers at EPU conditions. This attachment is considered by Continuum Dynamics Incorporated (CDI) to contain proprietary information exempt from disclosure pursuant to 10 CFR 2.390. Therefore, on behalf of CDI, NMPNS hereby makes application to withhold this document from public disclosure in accordance with 10 CFR 2.390(b)(1). An affidavit executed by CDI detailing the reasons for the request to withhold the proprietary information is provided in Attachment 12.

2.0 DETAILED DESCRIPTION

2.1 Operating License 2.C.(1), 2.C.(7) Changes

Nine Mile Point Unit 2 OL Section 2.C.(1), Maximum Power Level, specifies the maximum power level at which NMP2 may be operated. By letter dated July 22, 1993, NMP2 applied for a Stretch Power Uprate to increase the maximum power level specified in OL Section 2.C.(1) from the OLTP level of 3323 MWt to 3467 MWt (104.3% of OLTP). That request was approved by Amendment No. 66 dated April 28, 1995. The purpose of this license amendment request is to increase the maximum power level specified in OL 2.C.(1) from the CLTP level of 3467 MWt to 3988 MWt. The new maximum power level represents an increase of 20 percent (i.e., an Extended Power Uprate) from the OLTP level of 3323 MWt and an increase of 15 percent from the CLTP level of 3467 MWt.

Operating License Section 2.C.(7), Operation with Reduced Feedwater Temperature (Section 15.1, SSER 4), states NMP2 shall not be operated with a feedwater heating capacity less than that required to produce a feedwater temperature of 405° F at steady-state conditions unless analyses supporting such operations are submitted by NMPNS and approved by the staff. The 405° F value will be revised to 420.5° F.

2.2 Technical Specification (TS) Changes

TS changes are required to support the increase in the authorized maximum power level delineated in OL Section 2.C.(1). A description of each TS change is provided below. As indicated, the majority of the proposed changes involve TS values that are expressed as a percentage of Rated Thermal Power (RTP). However, multiple TSs were also identified that contain values expressed in terms of a percentage of RTP that do not require revision to support EPU. To avoid any misunderstanding, these TSs are discussed in Section 2.3 with an explanation as to why a revision is unnecessary.

TS Section 1.1, Definitions - Rated Thermal Power

Rated thermal power is currently defined as the total reactor core heat transfer rate to the reactor coolant (i.e., 3467 MWt). The stated CLTP value of 3467 MWt will be changed to 3988 MWt.

TS Section 2.1.1, Reactor Core Safety Limits (SLs)

TS Section 2.1.1.1 currently states that with the reactor steam dome pressure < 785 psig or core flow < 10 % rated core flow, Thermal Power shall be ≤ 25 % RTP. The stated RTP percentage will be changed from ≤ 25 % RTP to ≤ 23 % RTP.

TS Section 3.1.7, Standby Liquid Control (SLC) System

TS Section 3.1.7, SLC System, Surveillance Requirement (SR) 3.1.7.7, requires verification that each pump develop a flow rate ≥ 41.2 gpm at a discharge pressure of ≥ 1325 psig. The stated discharge pressure will be changed from ≥ 1325 psig to ≥ 1327 psig.

TS Section 3.2.1, Average Planar Linear Heat Generation Rate (APLHGR)

TS Section 3.2.1, APLHGR Applicability, Actions, and Surveillance Requirements are dependent on a percentage of RTP (i.e., 25 % RTP). The stated RTP percentage will be changed from 25 % RTP to 23 % RTP.

TS Section 3.2.2, Minimum Critical Power Ratio (MCPR)

TS Section 3.2.2, MCPR Applicability, Actions, and Surveillance Requirements are dependent on a percentage of RTP (i.e., 25 % RTP). The stated RTP percentage will be changed from 25 % RTP to 23 % RTP.

TS Section 3.2.3, Linear Heat Generation Rate (LHGR)

TS Section 3.2.3, LHGR Applicability, Actions, and Surveillance Requirements are dependent on a percentage of RTP (i.e., 25 % RTP). The stated RTP percentage will be changed from 25 % RTP to 23 % RTP.

TS Section 3.3.1.1, Reactor Protection System (RPS) Instrumentation

The following RPS Instrumentation Actions and Surveillance Requirements contained in TS Section 3.3.1.1, including Table 3.3.1.1-1, are dependent on a percentage of RTP and will be revised as shown:

Required Action E.1, which requires that Thermal Power be reduced to < 30 % RTP, will be revised to require that Thermal Power be reduced to < 26 % RTP.

The threshold for performing SR 3.3.1.1.3 (and associated Note) will be revised from ≥ 25 % RTP to ≥ 23 % RTP.

The threshold for performing SR 3.3.1.1.15, Turbine Stop Valve-Closure and Turbine Control Valve Fast Closure, Trip Oil Pressure-Low Functions, will be revised from ≥ 30% RTP to ≥ 26 % RTP.

The threshold for performing SR 3.3.1.1.16, Average Power Rage Monitor (APRM) Oscillation Power Range Monitor (OPRM)-Upscale Function, will be revised from ≥ 30% RTP to ≥ 26 % RTP.

Table 3.3.1.1-1, Function 2.b, Flow Biased Simulated Thermal Power-Upscale, contains both a flow-biased Allowable Value (AV) (≤ 0.64W + 63.8 % RTP) and a fixed AV clamped at 115.5 % RTP. The flow-biased AV will be changed to (≤ 0.55W + 60.5 % RTP). Note (b) modifies the Function 2.b AV when reset for single loop operation per Limiting Condition for Operation (LCO) 3.4.1, Recirculation Loops Operating. Note (b) will be revised to a value of 0.50(W – 5%) + 53.5 % RTP. W = Recirculation Drive Flow in percent of Rated Flow.

Table 3.3.1.1-1, Function 8, Turbine Stop Valve-Closure and Function 9, Turbine Control Valve Fast Closure, Trip Oil Pressure-Low, both specify an Applicable Mode or other Specified Conditions of ≥ 30 % RTP. The ≥ 30 % RTP value will be revised to ≥ 26 % RTP.

The following notes will be added to the Table 3.3.1.1-1 calibration surveillance requirements for the Flow Biased Simulated Thermal Power – Upscale function:

1. If the As-Found channel setpoint is outside its predefined as-found tolerances, then the channel shall be evaluated to verify that it is functioning as required before returning the channel to service.
2. The instrument channel setpoint shall be reset to a value that is within the As-Left tolerance around the nominal trip setpoint at the completion of the surveillance; otherwise, the channel shall be declared inoperable. Setpoints more conservative than the nominal trip setpoint are acceptable provided that the As-Found and As-Left tolerances apply to the actual setpoint implemented in the surveillance procedures to confirm channel performance. The nominal trip setpoint and the methodologies used to determine the As-Found and the As-Left tolerances are specified in the Bases associated with the specified function.

TS Section 3.3.2.2, Feedwater System and Main Turbine High Water Level Trip Instrumentation

TS Section 3.3.2.2, Feedwater System and Main Turbine High Water Level Trip Instrumentation Applicability and Required Action C.2 are dependent on a percentage of RTP (i.e., 25 % RTP). The stated RTP percentage will be changed from 25 % RTP to 23 % RTP.

TS Section 3.3.4.1, End of Cycle Recirculation Pump Trip (EOC - RPT) Instrumentation

TS Section 3.3.4.1, End of Cycle Recirculation Pump Trip (EOC-RPT) Instrumentation Applicability, Actions, and Surveillance Requirements are dependent on a percentage of RTP (i.e., 30 % RTP). The stated RTP percentages will be changed from 30 % RTP to 26 % RTP.

TS Table 3.3.6.1-1, Primary Containment Isolation Instrumentation

TS Table 3.3.6.1-1, Primary Containment Isolation Instrumentation, Function 1.c, Main Steam Line (MSL) Flow - High, specifies an AV of $\leq$ 122.8 psid. The stated AV of $\leq$ 122.8 psid will be changed to $\leq$ 184.4 psid.

TS Section 3.4.3, Jet Pumps

TS Section 3.4.3. Jet Pumps, SR 3.4.3.1, Note 2, indicates that the surveillance is not required to be performed until 24 hours after > 25 % RTP. The stated RTP percentage will be changed from > 25 % RTP to > 23 % RTP.

TS Section 3.7.5, Main Turbine Bypass System

TS Section 3.7.5, Main Turbine Bypass System, Applicability and Actions, are both dependent on a percentage of RTP (i.e., 25 % RTP). The stated RTP percentage will be changed from 25 % RTP to 23 % RTP.

2.3 Technical Specifications Not Requiring Change

The following table contains a listing of the NMP2 technical specifications that reference %RTP values that are not being changed as a result of EPU. A brief justification for leaving the %RTP value unchanged is provided.

Technical Specification	Justification for %RTP Remaining Unchanged
TS 3.1.3 Control Rod OPERABILITY (10% RTP) Actions	No changes required. The Rod Worth Minimizer (RWM) Low Power Set Point (LPSP) is unchanged in terms of percent power (%RTP) for EPU. The LPSP defines the power level below which the RWM is required. Maintaining this function in effect until 10% RTP will result in a larger RWM range in terms of absolute power; therefore, not revising the LPSP is conservative for EPU.
TS 3.1.4 Control Rod Scram Times (40% RTP) Surveillance Requirement	No changes required. As stated in the T.S. 3.1.4 Bases, the 40% RTP provides a reasonable time to complete the scram time testing following a shutdown. As such, this is a timing consideration to allow for the testing to be completed and does not affect the operation or operability of the control rods. Thus, it is acceptable to maintain the current 40% RTP.
TS 3.1.6 Rod Pattern Control (10% RTP) Applicability	No changes required. The RWM LPSP is unchanged in terms of percent power for EPU. The LPSP defines the power level below which the RWM is required. Maintaining this function in effect until 10% RTP will result in a larger RWM range in terms of absolute power; therefore, not revising this setpoint is conservative for EPU.

Technical Specification	Justification for %RTP Remaining Unchanged
TS 3.3.1.1 Reactor Protection System Instrumentation (20% RTP) Table 3.3.1.1-1 AV (Item 2a)	No changes required. TS Bases states that the Scram Setdown function indirectly ensures that, before the reactor mode switch is placed in the run position, reactor power does not exceed the Thermal Limit Monitoring value. The rescaled 23% RTP Thermal Limit Monitoring value continues to exceed the Scram Setdown Allowable Value (AV) of 20% RTP. Because the Scram Setdown is based on the AV, no change is required to the Scram Setdown.
TS 3.3.1.1 Reactor Protection System Instrumentation (120% RTP) Table 3.3.1.1.-1 AV (Item 2c)	No changes required. The Analytical Limit (AL) associated with the Allowable Value has not changed and the instrument has not changed. Therefore, the value of ≤ 120% RTP does not change.
TS 3.3.1.1 Reactor Protection System Instrumentation (1000 Effective Full Power Hours) SR 3.3.1.1.7	No changes required. There is sufficient margin between the actual and allowed sensitivity of the Local Power Range Monitor (LPRM) detectors to absorb the sensitivity loss due to the increase in detector flux and burnup at EPU conditions. Therefore, LPRM calibration frequency is not affected by EPU and no changes are required.
TS 3.3.2.1 Control Rod Block Instrumentation (10% RTP) SR Table 3.3.2.1-1 (notes)	No changes required. The RWM LPSP is unchanged in terms of percent power for EPU. The LPSP defines the power level below which the RWM is required. Maintaining this function in effect until 10% RTP will result in a larger RWM range in terms of absolute power; therefore, not revising this setpoint is conservative for EPU.
TS 3.3.2.1 Control Rod Block Instrumentation (28%, 63%, 83%, & 90% RTP) SR Table 3.3.2.1-1 (notes)	No changes required. The AL associated with the AV power levels for the various ranges for RBM operability are unchanged in terms of percent power for EPU, thus no setpoint change is required (NEDC-33004P-A, CLTR Section 5.3.5). The power-dependant MCPR multipliers (Kp) at each AL are verified on a cycle specific bases in order to determine if the Kp multiplier is bounding.
TS 3.3.4.2 ATWS Recirculation Pump Trip (5% RTP) Surveillance Requirement	No changes required. SR 3.3.4.2.4 requires verification that for the reactor vessel steam dome pressure – high function, the low frequency motor generator (LFMG) trip is not bypassed for > 29 seconds when thermal power is > 5% RTP. The combination of the reactor vessel steam dome pressure – high function and the LFMG trip is intended to mitigate the effects of an Anticipated Transient Without Scram (ATWS) event. For this surveillance, 5% RTP is a reasonable low power level at which the effects of an ATWS are not severe. The absolute thermal power level (MWt) in this range is approximately equal at CLTP and EPU. Therefore, the thermal power level of 5% RTP is not changed for EPU.
TS 3.4.11 RCS Pressure and Temperature (P/T) Limits (30% RTP in single loop operation (SLO)) Surveillance Requirement	No changes required. Maintaining 30% RTP for implementation of surveillance requirements 3.4.11.5 and 3.4.11.6 is conservative because the surveillances will commence at a higher absolute power level (i.e., earlier) than would be required by scaling.
TS 3.6.2.1 Suppression Pool Average Temperature (1% RTP) LCO and Actions	No changes required. The TS Bases on page B 3.6.2.1-1 states: "At 1% RTP, heat input is approximately equal to normal system heat losses." Because the TS Bases define the criteria for 1% RTP (i.e., the point of adding heat), the same power level will be utilized following EPU and a change to this value is not required.

Technical Specification	Justification for %RTP Remaining Unchanged
TS 3.6.3.2 Primary Containment Oxygen Concentration (15% RTP) Applicability, Actions	No changes required. Reference is made in this section to "< 15% RTP" and "> 15% RTP." This value is provided as an indication of plant startup and provides for the start of containment inerting. Maintaining this value at 15% of the EPU RTP continues to comply with the Technical Specification Bases statement that for reactor power below this value, the potential for an event that generates significant hydrogen and oxygen is low.

3.0 TECHNICAL EVALUATION

3.1 Operating License and Technical Specifications Changes

NEDC-33004P-A, Licensing Topical Report Constant Pressure Power Uprate, Revision 4, provides an NRC accepted approach for performing EPU. The approach is referred to as Constant Pressure Power Uprate (CPPU) and has been used as the basis of multiple power uprate license amendment requests submitted to and approved by the NRC. As the name suggests, the CPPU approach maintains a plant's current maximum operating reactor pressure. The constant pressure constraint, along with other required limitations and restrictions discussed in the CLTR, allows a simplified approach to power uprate analyses and evaluations.

Office of Nuclear Reactor Regulation, Review Standard for Extended Power Uprates, RS-001, Revision 0, December 2003, provides guidance to the NRC Staff when performing reviews of EPU applications. The review standard was developed to enhance the consistency, quality, and completeness of the Staff's reviews and to inform licensees of the guidance documents the Staff would use when reviewing EPU applications. These documents provide the acceptance criteria for the areas of review allowing licensees to prepare EPU applications that are complete with respect to the areas that are within the Staff's scope of review. Section 3.2 of RS-001, Template Safety Evaluation for Boiling-Water Reactor Extended Power Uprate, Inserts 1-13, provides the Staff an outline to follow when generating plant-specific safety evaluations. For each area of concern, a Regulatory Evaluation and Conclusion statement are provided. As noted in RS-001, the use of this review standard was not intended to undermine the NRC's topical report review and approval process. If a licensee references an NRC-approved topical report for an area covered by RS-001, the Staff will review the application only to ensure that the licensee is applying the topical report under conditions for which the topical report was approved, using appropriate plant-specific inputs.

NEDO-33351, Safety Analysis Report for Nine Mile Point Nuclear Station Unit 2 Constant Pressure Power Uprate, (non-proprietary version) is provided as Attachment 3 to this Enclosure. This report provides an integrated summary of the results of the safety analyses and evaluations performed that support the proposed increase to the maximum power level at NMP2 as delineated in OL Section 2.C.(1), Maximum Power Level. The PUSAR, Section 2, Safety Evaluation, follows the format and guidance delineated in RS-001, Section 3.2, to the extent that the review standard is consistent with the design basis of NMP2. Differences between the plant-specific design basis and RS-001 Regulatory Evaluations are described and evaluations provided. As appropriate, the PUSAR's Technical Evaluations are based on NRC approved topical report NEDC-33004P-A, Licensing Topical Report Constant Pressure Power Uprate, Revision 4. A proprietary version of the Safety Analysis is provided as Attachment 11.

EVALUATION OF THE PROPOSED CHANGE

In developing the PUSAR, NMPNS identified certain evaluations that, due to size, level of detail, and/or subject matter, were more appropriately broken out as separate Attachments. These areas include the EPU Testing Plan, Attachment 7, the Grid Stability Evaluation, Attachment 8, the Flow Induced Vibration-Piping/ Component Evaluation, Attachment 10, and the Steam Dryer Evaluation, Attachment 13. These evaluations support the appropriate PUSAR Technical Evaluations.

The following table provides a list of the OL and TS changes along with a brief description of each and the PUSAR section which provides the technical basis for the change:

Technical Specification	Description / Discussion of the Change	Supporting PUSAR Section
Operating License Maximum Power Level Operating License Operation with Reduced Feedwater Temperature	Revise the Rated Thermal Power in Section 2.C.(1) from 3467 MWt to **3988 MWt.** The TS value of minimum feedwater temperature allowed during rated steady-state conditions has been increased by the same amount as the feedwater temperature used in the heat balance in order to maintain the same margin to the original basis. The feedwater temperature used in the heat balance was changed from 425.1 F to 440.5 F; therefore, the TS value in Section 2.C.(7) shall be changed from 405 F to **420.5 F**.	1.2.1 Table 1-2
TS 1.1 Definitions	Revise Rated Thermal Power definition from 3467 MWt to **3988 MWt** to reflect EPU conditions.	1.2.1
TS 2.1.1.1 Reactor Core SLs	The historical 25% of RTP value for the TS Safety Limit, some thermal limits monitoring LCO thresholds, and some SR thresholds are based on generic analyses (evaluated up to ~50% of original RTP) applicable to the plant design with highest average bundle power for all of the BWR product lines. As originally licensed, the highest average bundle power (at 100% RTP) for any BWR6 is 4.8 MWt/bundle. The 25% RTP value is a conservative basis, as described in the plant Technical Specifications, however, this % value should be reduced when any plant is uprated such that at 100% of uprated power the average bundle power is greater than the original generic basis of 4.8 MWt/bundle. Therefore, to maintain the same basis with respect to absolute thermal power, if the uprated average bundle power is > 4.8 MWt/bundle, then the % RTP value is revised to equal (25% * 4.8 MWt/bundle * # of bundles / total uprated MWt). For the EPU, the average bundle power is > 4.8 MWt/bundle. Therefore, the Safety Limit % RTP basis and the thermal limits monitoring LCO and SR % RTP thresholds are reduced to **23% RTP**.	2.8.2.2
TS 3.1.7 Standby Liquid Control Surveillance Requirement	Change ≥ 1325 psig to **≥ 1327 psig.** The maximum pump discharge pressure for limiting ATWS event at EPU conditions is 1326.4 psig.	2.8.4.5
TS 3.2.1 Average Planar Linear Heat Generation Rate (APLHGR) Applicability, Actions, SR	The 25% RTP value is reduced to **23% RTP.** See discussion provided for TS 2.1.1.1.	2.8.2.2
TS 3.2.2 Minimum Critical Power Ratio (MCPR) Applicability, Actions, SR	The 25% RTP value is reduced to **23% RTP.** See discussion provided for TS 2.1.1.1.	2.8.2.2

Technical Specification	Description / Discussion of the Change	Supporting PUSAR Section
TS 3.2.3 Linear Heat Generation Rate (LHGR) Applicability, Actions, SR	The 25% RTP value is reduced to **23% RTP.** See discussion provided for TS 2.1.1.1.	2.8.2.2
TS 3.3.1.1 Reactor Protection System Instrumentation Actions, SR, Table 3.3.1.1-1 Functions 8 and 9	Based on the guidelines in Section F.4.2.3 of General Electric Licensing Topical Report NEDC-32424P-A, the Turbine Stop Valve (TSV) Closure and Turbine Control Valve (TCV) Fast Closure Scram and RPT Bypass analytical limit in % RTP is reduced by the ratio of the power increase. The new analytical limit does not change with respect to absolute thermal power. Because the trip does not change in terms of absolute power, there is no effect on the transient response. Therefore, the revised value in % RTP is: 30% RTP x 3467 MWt/ 3988 MWt = **26% RTP**	2.4.1.3
TS 3.3.1.1 Reactor Protection System Instrumentation Surveillance Requirements	The 25% RTP value is reduced to **23% RTP.** See discussion provided for TS 2.1.1.1.	2.8.2.2
TS 3.3.1.1 Reactor Protection System Instrumentation AV Table 3.3.1.1-1 Item 2b	The AV of the APRM Flow Biased Simulated Thermal Power (STP) - Upscale Scram is revised From: ≤ 0.64 W + 63.8 % RTP To: **≤ 0.55 W + 60.5 % RTP** Additionally, footnote (b) revised From: "Allowable Value is .58(W-5%) + 62% RTP when reset for single loop operation per LCO 3.4.1, 'Recirculation Loops Operating'" To: **"Allowable Value is .50(W-5%) + 53.5% RTP when reset for single loop operation per LCO 3.4.1, 'Recirculation Loops Operating'"** This value is equivalent to the current set point and therefore is not a change in terms of absolute power.	2.4.1.3
TS 3.3.1.1 Reactor Protection System Instrumentation Table 3.3.1.1-1 Item 2b SR 3.3.1.1.13 Notes	The following notes will be added to SR 3.3.1.1.13 for Item 2b: **(c) If the As-Found channel setpoint is outside its predefined as-found tolerances, then the channel shall be evaluated to verify that it is functioning as required before returning the channel to service.** **(d) The instrument channel setpoint shall be reset to a value that is within the As-Left tolerance around the nominal trip setpoint at the completion of the surveillance; otherwise, the channel shall be declared inoperable. Setpoints more conservative than the nominal trip setpoint are acceptable provided that the As-Found and As-Left tolerances apply to the actual setpoint implemented in the surveillance procedures to confirm channel performance. The nominal trip setpoint and the methodologies used to determine the As-Found and the As-Left tolerances are specified in the Bases associated with the specified function.**	Section 3.2.3 of this Enclosure
TS 3.3.2.2 Feedwater System and Main Turbine High Water Level Trip Instrumentation Applicability, Actions	The 25% RTP value is reduced to **23% RTP.** See discussion provided for TS 2.1.1.1.	2.8.2.2

Technical Specification	Description / Discussion of the Change	Supporting PUSAR Section
TS 3.3.4.1 End of Cycle Recirculation Pump Trip Instrumentation Applicability, Actions, SR	The 30% RTP value is reduced to **26% RTP.** See discussion provided for TS 3.3.1.1.	2.4.1.3
TS 3.3.6.1 Primary Containment Isolation Instrumentation AV Table 3.3.6.1-1, Item 1c	The AV of the Main Steam Line Isolation, Main Steam Line Flow - High is revised. From: 122.8 psid To: **184.4 psid**	2.4.1.3
TS 3.4.3 Jet Pumps Surveillance Requirement	The 25% RTP value is reduced to **23% RTP.** See discussion provided for TS 2.1.1.1.	2.8.2.2
TS 3.7.5 Main Turbine Bypass System Applicability, Actions	The 25% RTP value is reduced to **23% RTP.** See discussion provided for TS 2.1.1.1.	2.8.2.2

3.2 Technical Specification Instrument Setpoint Changes

Technical Specification Allowable Values for the functions addressed below will be revised for operation at EPU conditions.

3.2.1 Setpoint Calculation Methodology

<u>APRM Flow-Biased Simulated Thermal Power – Upscale Scram</u>

The setpoints for this function were changed for the implementation of the ARTS/MELLLA amendment (Reference 4) in accordance with NRC approved GEH methodology in NEDC-31336P-A, General Electric Instrument Setpoint Methodology. For EPU, the changes in the instrument uncertainties were sufficiently small that using a simplified process to change the instrument Allowable Value and nominal trip setpoint by the same difference as the change in the Analytical Limit was justified by NRC approved GEH methodology in NEDC-33004P-A, Constant Pressure Power Uprate. The EPU Allowable Value for this function is provided in Section 2.2 of this Enclosure. The nominal trip setpoint (NTSP) is 0.55 W + 57.5 % RTP for two loop operation and 0.50 (W-5%) + 50.5% RTP for single loop operation, where W = Recirculation Drive Flow in percent of Rated Flow. The Analytical Limit for this function is 0.55 W + 63.5 % RTP for two loop operation and 0.50 (W-5%) + 56.5% RTP for single loop operation.

<u>Main Steam Line High Steam Flow Main Steam Isolation Valve (MSIV) Isolation</u>

The Analytical Limit for EPU conditions is maintained at 140% of the rated steam flow. The Allowable Value and nominal trip setpoint both increase in units of psid due to the higher absolute mass flowrate Analytical Limit for EPU. The Allowable Value and nominal trip setpoint were re-calculated using NRC approved GEH methodology in NEDC-31336P-A, General Electric Instrument Setpoint Methodology and NEDC-32889P, General Electric Methodology for Instrumentation Technical Specification and Setpoint Analysis. A sample calculation demonstrating the application of this methodology is provided in Section 2.4.2 of the PUSAR (Attachment 11). The EPU Allowable Value for this function is provided in Section 2.2 of this

Enclosure. The nominal trip setpoint is 183 psid. The Analytical Limit for this function corresponding to 140% rated steam flow is 194.4 psid. There is a plant specific program which verifies that this instrument channel functions as required by verifying the as-left and as-found settings are consistent with those established by the setpoint methodology.

3.2.2 Safety Limit-Related Limiting Safety System Settings (LSSS) Determination

In accordance with 10 CFR 50.36(c)(1)(ii)(A), the following guidance is provided for identifying a list of functions to be included in the subset of LSSSs specified for variables on which Safety Limits (SLs) have been placed as defined in Standard Technical Specifications Sections 2.1.1, "Reactor Core SLs," and 2.1.2, "Reactor Coolant System Pressure SLs." This subset includes automatic protective devices in Technical Specifications for specified variables on which SLs have been placed that: (1) initiate a reactor trip; or (2) actuate safety systems. In accordance with General Design Criteria 10, SLs ensure that specified acceptable fuel design limits are not exceeded during steady state operation, normal operational transients, and anticipated operational occurrences (AOOs). As explained below, there are no SL-Related LSSS functions being revised by this License Amendment Request.

APRM Flow-Biased Simulated Thermal Power – Upscale Scram
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 RPS as required by the NRC approved licensing basis. This function does not provide an automatic trip sepoint that protects against violating the Reactor Core Safety Limit or Reactor Coolant System Pressure Safety Limit during AOOs. As noted in NMP2 Amendment No. 123 dated February 27, 2008, Safety Evaluation Section 3.13.2, Non-SL-Related LSSS, the Flow Biased Simulated Thermal Power-Upscale Function is not a SL-Related LSSS.

Main Steam Line High Steam Flow MSIV Isolation
The Main Steam Line Flow-High Function is directly assumed in the analysis of the main steam line break accident. NMP2 Updated Safety Analysis Report (USAR) Section 15.6.4.1.2, Frequency Classification, categorizes the main steam line break as a limiting fault. USAR Section 15.0.3.1 defines limiting faults as occurrences that are not expected to occur but are postulated because their consequences may result in the release of significant amount of radioactive material. This event is referred to as a design basis (postulated) accident. The isolation action, along with the scram function of the RPS, ensures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46 and the offsite doses do not exceed the 10 CFR 50.67 limits. Because the MSL Flow-High Function is credited only in a DBA and does not provide an automatic trip sepoint that protects against violating the Reactor Core Safety Limit or Reactor Coolant System Pressure Safety Limit during AOOs, it is therefore not considered an SL-Related LSSS function.

3.2.3 Instrument Setpoint Controls

A Surveillance Test Program is in place to ensure the APRM Flow-Biased Scram STP Upscale Scram and Main Steam Line High Flow MSIV Isolation functions will perform in accordance with applicable design requirements. Nominal trip setpoints are specified in controlled setpoint calculations and incorporated into applicable Surveillance Test Procedures. The nominal trip setpoints and As-Left tolerances are selected to ensure that the actual setpoints do not exceed the Allowable Value between successive channel calibrations. Instrument reference accuracy is used for the As-Found and As-Left tolerances. An As-Left setting is procedurally required to be within the As-Left tolerance prior to returning the channel to service. If the As-Found setting is outside the required As-Found tolerance, the device is reset to within the As-Left tolerance.

Operability determinations are integral to the NMPNS 10CFR50, Appendix B, Criterion XVI, Corrective Action Program. When a problem described in a condition report represents an operability concern, an Operability Determination is completed. Return of a degraded or non-conforming component to service is addressed under the corrective action program.

For the APRM Flow-Biased Scram STP Upscale Scram function, setpoints found outside As-Found tolerances are evaluated for functionality through the corrective action program. A channel is inoperable if its actual trip setpoint is not within its required Allowable Value, or if it cannot be reset to within its As-Left tolerance. For Main Steam Line High Flow MSIV Isolation, a channel is inoperable if its actual trip setpoint is not within its required Allowable Value, or if it cannot be reset to within its As-Left tolerance. Applicable surveillance test procedures will be revised to ensure setpoints found outside As-Found tolerances are evaluated for functionality through the corrective action program.

Notes will be added to the applicable TS Surveillance Requirements that are consistent with the NRC staff's position on complying with 10 CFR 50.36 as provided in RIS 2006-17 and further clarified by Technical Specification Task Force (TSTF)-493, Revision 3. Specifically, the following notes are added to the TS Surveillance Requirements for the APRM Flow-Biased simulated Thermal Power Upscale Scram function:

1. If the As-Found channel setpoint is outside its predefined As-Found tolerances, then the channel shall be evaluated to verify that it is functioning as required before returning the channel to service.

2. The instrument channel setpoint shall be reset to a value that is within the As-Left tolerance around the nominal trip setpoint at the completion of the surveillance; otherwise, the channel shall be declared inoperable. Setpoints more conservative than the nominal trip setpoint are acceptable provided that the As-Found and As-Left tolerances apply to the actual setpoint implemented in the surveillance procedures to confirm channel performance. The nominal trip setpoint and the methodologies used to determine the As-Found and the As-Left tolerances are specified in the Bases associated with the specified function.

Information will be added to the applicable TS Bases that are consistent with the NRC staff's position on complying with 10 CFR 50.36 as provided in RIS 2006-17 and further clarified by TSTF-493, Revision 3, and TSTF-09-07 letter to NRC dated February 23, 2009, for non-SL-Related LSSS functions. Specifically, the following information will be added to the TS Bases for the Main Steam Line High Flow MSIV Isolation function:

> There is a plant specific program which verifies that this instrument channel functions as required by verifying the As-Left and As-Found settings are consistent with those established by the setpoint methodology.

4.0 REGULATORY EVALUATION

4.1 Applicable Regulatory Requirements/Criteria

NEDO-33351, Safety Analysis Report for Nine Mile Point Nuclear Station Unit 2 Constant Pressure Power Uprate is provided as Attachment 3 (non-proprietary version) and Attachment 11 (proprietary version). For each PUSAR Safety Evaluation Section, a Regulatory Evaluation is provided which describes the pertinent regulatory requirements and criteria. Also provided is a Technical Evaluation which explains the EPU changes and how the applicable regulatory requirements are met. The PUSAR safety evaluation follows the format and guidance delineated in RS-001 (Revision 0), Office of Nuclear Reactor Regulation, Review Standard for Extended Power Uprates, to the extent that the review standard is consistent with the design basis of NMP2. For differences between the plant-specific design bases and RS-001 regulatory evaluation sections, the corresponding PUSAR safety evaluation regulatory evaluation section was revised to reflect the NMP2 design basis. As appropriate, the PUSAR's technical evaluations are based on NRC approved topical report NEDC-33004P-A, Licensing Topical Report Constant Pressure Power Uprate, Revision 4.

4.2 Significant Hazards Consideration

In accordance with 10 CFR 50.91(a), "At the time a licensee requests an amendment, it must provide to the Commission.... its analysis about the issue of no significant hazards consideration using the standards in § 50.92." The following provides this analysis for the Nine Mile Point Nuclear Station Unit 2 (NMP2) Extended Power Uprate (EPU). The conclusions are based on the evaluations provided in NEDC-33351P, Safety Analysis Report for Nine Mile Point Nuclear Station Unit 2 Constant Pressure Power Uprate, and are summarized as appropriate to the following safety considerations in accordance with 10 CFR 50.92.

1) Will the change involve a significant increase in the probability or consequences of an accident previously evaluated?

No, the increase in power level discussed herein will not significantly increase the probability or consequences of an accident previously evaluated.

The proposed change will increase NMP2's authorized maximum power level from the current licensed thermal power (CLTP) level of 3467 megawatts thermal (MWt) to 3988 MWt. In support of this Constant Pressure Extended Power Uprate (CPPU), a comprehensive evaluation was performed for nuclear steam supply system (NSSS) and balance of plant (BOP) systems, structures, components, and analyses that could be affected by this change. The effect of increasing the maximum power level from the CLTP of 3467 MWt to 3988 MWt on the NMP2 licensing and design bases was evaluated. The result of this evaluation is that all plant components, as modified, will continue to be capable of performing their design function at an uprated core power of 3988 MWt. In addition, an evaluation of the accident analyses concludes that applicable analysis acceptance criteria continue to be met. Power level is an input assumption to the equipment design and accident analyses, but it is not an initiator for any transient or accident. Therefore, no accident initiators are affected by this uprate and no challenges to any plant safety barriers are created by this change.

Therefore, operation of the facility in accordance with the proposed change does not involve a significant increase in the probability of an accident previously evaluated.

This change does not affect the release paths, the frequency of release, or the source term for release for any accidents previously evaluated in the Updated Safety Analysis Report (USAR). Structures, systems, and components (SSC) required to mitigate transients remain capable of performing their design functions, and thus were found acceptable. The source terms used to assess radiological consequences have been reviewed and determined to bound operation at the uprated condition. The results of EPU accident evaluations do not exceed the U. S. Nuclear Regulatory Commission (NRC) approved acceptance limits.

The spectrum of postulated accidents and transients has been investigated and are shown to meet the regulatory criteria to which NMP2 is currently licensed. In the area of fuel and core design, the Safety Limit Minimum Critical Power ratio (SLMCPR) and other applicable Specified Acceptable Fuel Design Limits (SAFDLS) are still met. Continued compliance with the SLMCPR and other SAFDLs is confirmed on a cycle specific basis consistent with criteria accepted by the NRC.

Challenges to the reactor coolant pressure boundary were evaluated at EPU conditions (pressure, temperature, flow, and radiation) and found to meet the acceptance criteria for allowable stresses. Adequate overpressure margin is maintained.

Challenges to the containment have been evaluated and the containment and its associated cooling system continue to meet applicable regulatory requirements. The increase in the calculated post Loss of Coolant Accident (LOCA) suppression pool temperature above the current peak temperature was evaluated and determined to be acceptable.

Radiological release events (accidents) have been evaluated and shown to meet the requirements of 10 CFR 50.67.

Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2) Will the change create the possibility of a new or different kind of accident from any accident previously evaluated?

No, the increase in power level discussed herein will not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed change will increase NMP2's authorized maximum power level from the CLTP level of 3467 MWt to 3988 MWt. Equipment that could be affected by EPU has been evaluated. No new operating mode, safety-related equipment lineup, accident scenario, or equipment failure mode was identified. The full spectrum of accident considerations has been evaluated and no new or different kind of accident has been identified. This Constant Pressure Extended Power Uprate utilizes a standard evaluation methodology applied to known technology employed within the range of current or modified plant capabilities. As such, the plant safety-related equipment continues to operate in accordance with regulatory criteria. Evaluations were performed using NRC approved codes, standards and methods. No new accidents or event precursors have been identified.

All structures, systems and components previously required for the mitigation of a transient remain capable of fulfilling their intended design functions. The proposed changes do not adversely affect safety-related systems or components and do not challenge the performance or integrity of any safety-related system. This change does not adversely affect any current system interfaces or create any new interfaces that could result in an accident or malfunction of a different kind than was previously evaluated. Operating at a core power level of 3988 MWt does not create any new accident initiators or precursors.

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

3) Will the change involve a significant reduction in a margin of safety?

No, the increase in power level discussed herein will not involve a significant reduction in a margin of safety.

Comprehensive analyses of the proposed changes have concluded that relevant design and safety acceptance criteria will be met without a significant reduction in margins of safety. The analyses supporting EPU have demonstrated that the NMP2 SSCs are capable of safely performing at EPU conditions. The analyses identified and defined the major input parameters to the NSSS, analyzed NSSS design transients, and evaluated the capabilities of the NSSS fluid systems, NSSS/BOP interfaces, NSSS control systems, and NSSS and BOP components, as appropriate. Radiological consequences of design basis events remain within regulatory limits and are not increased significantly. The analyses confirmed that NSSS and BOP SSCs are capable, some with modifications, of achieving EPU conditions without significant reduction in margins of safety.

Analyses have shown that the integrity of primary fission product barriers will not be significantly affected as a result of the power increase. Calculated loads on SSCs important to safety have been shown to remain within design allowables under EPU conditions for all design basis event categories. Plant response to transients and accidents do not result in exceeding acceptance criteria. As appropriate, the evaluations that demonstrate acceptability of EPU have been performed using methods that have either been reviewed and approved by the NRC staff, or that are in compliance with regulatory review guidance and standards established for maintaining adequate margins of safety. These evaluations demonstrate that there are no significant reductions in the margins of safety.

Maximum power level is one of the inherent inputs that determine the safe operating range defined by the accident analyses. The Technical Specifications ensure that NMP2 is operated within the bounds of the inputs and assumptions used in the accident analyses. The acceptance criteria for the accident analyses are conservative with respect to the operating conditions defined by the Technical Specifications. The engineering reviews performed for the constant pressure extended power uprate confirm that the accident analyses criteria are met at the revised maximum allowable thermal power level of 3988 MWt, as well as at the rated thermal power (RTP) levels specified in the Facility Operating License and Technical Specifications. Therefore, the adequacy of the revised Facility Operating Licenses and Technical Specifications to maintain the plant in a safe operating range is also confirmed, and the increase in maximum allowable power level does not involve a significant decrease in a margin of safety.

Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

4.3 Conclusions

A CPPU to 120% of original licensed thermal power has been investigated. The NMP2 licensing requirements have been evaluated and the analyses demonstrate how this uprate can be accommodated without a significant increase in the probability or consequences of an accident previously evaluated, without creating the possibility of a new or different kind of accident from any accident previously evaluated, and without creating a significant reduction in the margin of safety.

In conclusion, based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public. Therefore, Nine Mile Point Nuclear Station, LLC concludes that the proposed amendment presents no significant hazards considerations under the standards set forth in 10 CFR 50.92, and, accordingly, a finding of "no significant hazards consideration" is justified.

5.0 ENVIRONMENTAL CONSIDERATION

The proposed OL and TS changes required for implementation of EPU meet the requirements for performing an environmental review as set forth in 10 CFR 51.20, Criteria for and Identification of Licensing and Regulatory Actions Requiring Environmental Impact Statements. Attachment 9, Supplemental Environmental Report, concludes that that the environmental impacts of operation at 3988 MWt are either bounded by the impacts described in earlier National Environmental Policy Act assessments or constrained by applicable regulatory criteria. As a result, NMPNS believes that the EPU would not significantly affect human health or the environment.

6.0 REFERENCES

1. NEDC-33004P-A, Licensing Topical Report Constant Pressure Power Uprate, Revision 4, July 2003.

2. RS-001, Office of Nuclear Reactor Regulation, Review Standard for Extended Power Uprates, Revision 0, December 2003

3. Nine Mile Point Nuclear Station, Unit 2, Technical Specification Amendment No. 66, TAC No. M87088, dated April 28, 1995

4. Nine Mile Point Nuclear Station, Unit 2, Technical Specification Amendment No. 123, TAC No. MD5233, dated February 27, 2008

5. Nine Mile Point Nuclear Station, Unit 2, Technical Specification Amendment No. 125, TAC No. MD5758, dated May 29, 2008

ENCLOSURE

ATTACHMENT 1

Operating License / Technical Specifications Page Markups

Operating License Pages Included in this Markup

4
5

Technical Specifications Pages Included in this Markup

1.1-5
2.0-1
3.1.7-3
3.2.1-1
3.2.2-1
3.2.3-1
3.3.1.1-2
3.3.1.1-4
3.3.1.1-6
3.3.1.1-8
3.3.1.1-9
3.3.1.1-10
3.3.2.2-1
3.3.2.2-2
3.3.4.1-1
3.3.4.1-2
3.3.4.1-3
3.3.6.1-6
3.4.3-2
3.7.5-1

(1) Maximum Power Level

3988

Nine Mile Point Nuclear Station, LLC is authorized to operate the facility at reactor core power levels not in excess of ~~3467~~ megawatts thermal (100 percent rated power) in accordance with the conditions specified herein.

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, as revised through Amendment No. 130 are hereby incorporated into this license. Nine Mile Point Nuclear Station, LLC shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(3) Fuel Storage and Handling (Section 9.1, SSER 4)*

a. Fuel assemblies, when stored in their shipping containers, shall be stacked no more than three containers high.

b. When not in the reactor vessel, no more than three fuel assemblies shall be allowed outside of their shipping containers or storage racks in the New Fuel Vault or Spent Fuel Storage Facility.

c. The above three fuel assemblies shall maintain a minimum edge-to-edge spacing of twelve (12) inches from the shipping container array and approved storage rack locations.

d. The New Fuel Storage Vault shall have no more than ten fresh fuel assemblies uncovered at any one time.

(4) Turbine System Maintenance Program (Section 3.5.1.3.10, SER)

The operating licensee shall submit for NRC approval by October 31, 1989, a turbine system maintenance program based on the manufacturer's calculations of missile generation probabilities. (Submitted by NMPC letter dated October 30, 1989 from C.D. Terry and approved by NRC letter dated March 15, 1990 from Robert Martin to Mr. Lawrence Burkhardt, III).

* The parenthetical notation following the title of many license conditions denotes the section of the Safety Evaluation Report (SER) and/or its supplements wherein the license condition is discussed.

(5) Inservice Inspection (Sections 5.2.4.3 and 6.6.3, SSER 5)

The operating licensee shall submit an inservice inspection program in accordance with 10 CFR 50.55a(g)(4) for staff review by July 31, 1987.

(6) Initial Startup Test Program (Section 14, SER, SSERs 4 and 5)

Any changes to the Initial Test Program described in Section 14 of the Final Safety Analysis Report made in accordance with the provisions of 10 CFR 50.59 shall be reported in accordance with 50.59(b) within one month of such change.

(7) Operation with Reduced Feedwater Temperature (Section 15.1, SSER 4)

Nine Mile Point Nuclear Station, LLC shall not operate the facility with reduced feedwater temperature for the purpose of extending the normal fuel cycle. The facility shall not be operated with a feedwater heating capacity less than that required to produce a feedwater temperature of 420.5°F ~~405°F~~ at rated steady-state conditions unless analyses supporting such operations are submitted by Nine Mile Point Nuclear Station, LLC and approved by the staff.

(8) Safety Parameter Display System (SPDS) (Section 18.2, SSERs 3 and 5)

Prior to startup following the first refueling outage, the operating licensee shall have operational an SPDS that includes the revisions described in their letter of November 19, 1985. Before declaring the SPDS operational, the operating licensee shall complete testing adequate to ensure that no safety concerns exist regarding the operation of the Nine Mile Point Nuclear Station, Unit No. 2 SPDS.

(9) Detailed Control Room Design Review (Section 18.1, SSERs 5 and 6)

(a) Deleted per Amendment No. 24 (12-18-90)

(b) Prior to startup following the first refueling outage, the operating licensee shall provide the results of the reevaluation of normally lit and nuisance alarms for NRC review in accordance with its August 21, 1986 letter.

(c) Prior to startup following the first refueling outage, the operating licensee shall complete permanent zone banding of meters in accordance with its August 4, 1986 letter.

1.1 Definitions (continued)

MODE	A MODE shall correspond to any one inclusive combination of mode switch position, average reactor coolant temperature, and reactor vessel head closure bolt tensioning specified in Table 1.1-1 with fuel in the reactor vessel.
OPERABLE—OPERABILITY	A system, subsystem, division, component, or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified safety function(s) and when all necessary attendant instrumentation, controls, normal or emergency electrical power, cooling and seal water, lubrication, and other auxiliary equipment that are required for the system, subsystem, division, component, or device to perform its specified safety function(s) are also capable of performing their related support function(s).
PHYSICS TESTS	PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation. These tests are: a. Described in Chapter 14, Initial Test Program of the FSAR; b. Authorized under the provisions of 10 CFR 50.59; or c. Otherwise approved by the Nuclear Regulatory Commission.
RATED THERMAL POWER (RTP)	RTP shall be a total reactor core heat transfer rate to the reactor coolant of ~~3467 MWt~~ 3988 MWt.
REACTOR PROTECTION SYSTEM (RPS) RESPONSE TIME	The RPS RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its RPS trip setpoint at the channel sensor until de-energization of the scram pilot valve solenoids. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured.

(continued)

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

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

THERMAL POWER shall be ≤ ~~25%~~ 23% RTP.

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

MCPR shall be ≥ 1.07 for two recirculation loop operation or ≥ 1.09 for single recirculation loop operation.

2.1.1.3 Reactor vessel water level shall be greater than the top of active irradiated fuel.

2.1.2 Reactor Coolant System Pressure SL

Reactor steam dome pressure shall be ≤ 1325 psig.

2.2 SL Violations

With any SL violation, the following actions shall be completed within 2 hours:

2.2.1 Restore compliance with all SLs; and

2.2.2 Insert all insertable control rods.

SURVEILLANCE REQUIREMENTS *(continued)*

	SURVEILLANCE	FREQUENCY
SR 3.1.7.7	Verify each pump develops a flow rate $\geq$ 41.2 gpm at a discharge pressure $\geq$ ~~1325~~ psig. 1327	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 <u>AND</u> Once within 24 hours after piping temperature is restored to $\geq$ 70°F
SR 3.1.7.10	Verify sodium pentaborate enrichment is $\geq$ 25 atom percent B-10.	Prior to addition to SLC tank

3.2 POWER DISTRIBUTION LIMITS

3.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)

LCO 3.2.1 All APLHGRs shall be less than or equal to the limits specified in the COLR.

APPLICABILITY: THERMAL POWER ≥ ~~25%~~ 23% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Any APLHGR not within limits.	A.1 Restore APLHGR(s) to within limits.	2 hours
B. Required Action and associated Completion Time not met.	B.1 Reduce THERMAL POWER to < ~~25%~~ 23% RTP.	4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.2.1.1 Verify all APLHGRs are less than or equal to the limits specified in the COLR.	Once within 12 hours after ≥ ~~25%~~ 23% RTP AND 24 hours thereafter

3.2 POWER DISTRIBUTION LIMITS

3.2.2 MINIMUM CRITICAL POWER RATIO (MCPR)

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

APPLICABILITY: THERMAL POWER ≥ ~~25%~~ 23% RTP.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. Any MCPR not within limits.	A.1	Restore MCPR(s) to within limits.	2 hours
B. Required Action and associated Completion Time not met.	B.1	Reduce THERMAL POWER to < ~~25%~~ 23% RTP.	4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.2.1	Verify all MCPRs are greater than or equal to the limits specified in the COLR.	Once within 12 hours after ≥ ~~25%~~ 23% RTP AND 24 hours thereafter

(continued)

3.2 POWER DISTRIBUTION LIMITS

3.2.3 LINEAR HEAT GENERATION RATE (LHGR)

LCO 3.2.3 All LHGRs shall be less than or equal to the limits specified in the COLR.

APPLICABILITY: THERMAL POWER ≥ ~~25%~~ 23% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Any LHGR not within limits.	A.1 Restore LHGR(s) to within limits.	2 hours
B. Required Action and associated Completion Time not met.	B.1 Reduce THERMAL POWER to < ~~25%~~ 23% RTP.	4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.2.3.1 Verify all LHGRs are less than or equal to the limits specified in the COLR.	Once within 12 hours after ≥ ~~25%~~ 23% RTP **AND** 24 hours thereafter

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. One or more Functions with RPS trip capability not maintained.	C.1 Restore RPS trip capability.	1 hour
D. Required Action and associated Completion Time of Condition A, B, or C not met.	D.1 Enter the Condition referenced in Table 3.3.1.1-1 for the channel.	Immediately
E. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	E.1 Reduce THERMAL POWER to < ~~30%~~ 26% RTP.	4 hours
F. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	F.1 Initiate alternate method to detect and suppress thermal-hydraulic instability oscillations.	12 hours
	<u>AND</u>	
	F.2 Restore required channel to OPERABLE status.	120 days
G. Required Action and associated Completion Time of Condition F not met. <u>OR</u> As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	G.1 Be in MODE 2.	6 hours

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.1.3	-------------------NOTE------------------- Not required to be performed until 12 hours after THERMAL POWER ≥ ~~25%~~ 23% RTP. -- Verify the absolute difference between the average power range monitor (APRM) channels and the calculated power ≤ 2% RTP while operating at ≥ 25% RTP. 23%	7 days
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.	7days

(continued)

1

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.1.13	------------------NOTES------------------ 1. Neutron detectors are excluded. 2. For Functions 1.a and 2.a, not required to be performed when entering MODE 2 from MODE 1 until 12 hours after entering MODE 2. 3. For Function 2.e, the CHANNEL CALIBRATION only requires a verification of OPRM-Upscale setpoints in the APRM by the review of the "Show Parameters" display. --- Perform CHANNEL CALIBRATION.	24 months
SR 3.3.1.1.14	Perform LOGIC SYSTEM FUNCTIONAL TEST.	24 months
SR 3.3.1.1.15	Verify Turbine Stop Valve—Closure, and Turbine Control Valve Fast Closure, Trip Oil Pressure—Low Functions are not bypassed when THERMAL POWER is ≥ ~~30%~~ 26% RTP.	24 months
SR 3.3.1.1.16	Verify APRM OPRM-Upscale Function is not bypassed when THERMAL POWER is ≥ ~~30%~~ 26% RTP and recirculation drive flow is < 60% of rated recirculation drive flow.	24 months

(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~~ SR 3.3.1.1.13[c],[d]	≤ ~~(.64W + 63.8% RTP)~~ .55W + 60.5% 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~~ .50(W-5%) + 53.5% RTP when reset for single loop operation per LCO 3.4.1, "Recirculation Loops Operating."

(c) If the As-Found channel setpoint is outside its predefined As-Found tolerances, then the channel shall be evaluated to verify that it is functioning as required before returning the channel to service.

Insert A

"Insert A"

(d) The instrument channel setpoint shall be reset to a value that is within the As-Left tolerance around the nominal trip setpoint at the completion of the surveillance; otherwise, the channel shall be declared inoperable. Setpoints more conservative than the nominal trip setpoint are acceptable provided that the As-Found and As-Left tolerances apply to the actual setpoint implemented in the surveillance procedures to confirm channel performance. The nominal trip setpoint and the methodologies used to determine the As-Found and the As-Left tolerances are specified in the Bases associated with the specified function.

Table 3.3.1.1-1 (page 2 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
3. Reactor Vessel Steam Dome Pressure – High	1,2	2	H	SR 3.3.1.1.1 SR 3.3.1.1.8 SR 3.3.1.1.9 SR 3.3.1.1.13 SR 3.3.1.1.14 SR 3.3.1.1.17	≤ 1072 psig
4. Reactor Vessel Water Level – Low, Level 3	1,2	2	H	SR 3.3.1.1.1 SR 3.3.1.1.8 SR 3.3.1.1.9 SR 3.3.1.1.13 SR 3.3.1.1.14 SR 3.3.1.1.17	≥ 157.8 inches
5. Main Steam Isolation Valve – Closure	1	8	G	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14 SR 3.3.1.1.17	≤ 12% closed
6. Drywell Pressure – High	1,2	2	H	SR 3.3.1.1.1 SR 3.3.1.1.8 SR 3.3.1.1.9 SR 3.3.1.1.13 SR 3.3.1.1.14	≤ 1.88 psig
7. Scram Discharge Volume Water Level – High					
a. Transmitter/Trip Unit	1,2	2	H	SR 3.3.1.1.1 SR 3.3.1.1.8 SR 3.3.1.1.9 SR 3.3.1.1.11 SR 3.3.1.1.14	≤ 49.5 inches
	5(a)	2	I	SR 3.3.1.1.1 SR 3.3.1.1.8 SR 3.3.1.1.9 SR 3.3.1.1.11 SR 3.3.1.1.14	≤ 49.5 inches
b. Float Switch	1,2	2	H	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≤ 49.5 inches
	5(a)	2	I	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≤ 49.5 inches
8. Turbine Stop Valve – Closure	≥ ~~30%~~ RTP 26%	4	E	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14 SR 3.3.1.1.15 SR 3.3.1.1.17	≤ 7% closed

(continued)

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

Table 3.3.1.1-1 (page 3 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
9. Turbine Control Valve Fast Closure, Trip Oil Pressure – Low	≥ ~~30%~~ 26% RTP	2	E	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14 SR 3.3.1.1.15 SR 3.3.1.1.17	≥ 465 psig
10. Reactor Mode Switch – Shutdown Position	1,2	2	H	SR 3.3.1.1.12 SR 3.3.1.1.14	NA
	5(a)	2	I	SR 3.3.1.1.12 SR 3.3.1.1.14	NA
11. Manual Scram	1,2	4	H	SR 3.3.1.1.4 SR 3.3.1.1.14	NA
	5(a)	4	I	SR 3.3.1.1.4 SR 3.3.1.1.14	NA

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

3.3 INSTRUMENTATION

3.3.2.2 Feedwater System and Main Turbine High Water Level Trip Instrumentation

LCO 3.3.2.2 Three channels of feedwater system and main turbine high water level trip instrumentation shall be OPERABLE.

APPLICABILITY: THERMAL POWER ≥ ~~25%~~ 23% RTP.

ACTIONS

-----------------------------------NOTE-----------------------------------
Separate Condition entry is allowed for each channel.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One feedwater system and main turbine high water level trip channel inoperable.	A.1 Place channel in trip.	7 days
B. Two or more feedwater system and main turbine high water level trip channels inoperable.	B.1 Restore feedwater system and main turbine high water level trip capability.	2 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. Required Action and associated Completion Time not met.	C.1	--------NOTE-------- Only applicable if inoperable channel is the result of an inoperable feedwater pump breaker. -------------------- Remove affected feedwater pump(s) from service.	4 hours
	<u>OR</u>		
	C.2	Reduce THERMAL POWER to < 25% RTP. 23%	4 hours

SURVEILLANCE REQUIREMENTS

------------------------------------NOTE------------------------------------
When a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided feedwater system and main turbine high water level trip capability is maintained.

SURVEILLANCE		FREQUENCY
SR 3.3.2.2.1	Perform CHANNEL CHECK.	24 hours
SR 3.3.2.2.2	Perform CHANNEL FUNCTIONAL TEST.	92 days

(continued)

3.3 INSTRUMENTATION

3.3.4.1 End of Cycle Recirculation Pump Trip (EOC-RPT) Instrumentation

LCO 3.3.4.1

a. Two channels per trip system for each EOC-RPT instrumentation Function listed below shall be OPERABLE:

1. Turbine Stop Valve (TSV)—Closure; and

2. Turbine Control Valve (TCV) Fast Closure, Trip Oil Pressure—Low.

<u>OR</u>

b. LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)," limits for inoperable EOC-RPT as specified in the COLR are made applicable.

APPLICABILITY: THERMAL POWER ≥ ~~30%~~ 26% RTP with any recirculation pump in fast speed.

ACTIONS

-------------------------------------NOTE-------------------------------------
Separate Condition entry is allowed for each channel.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required channels inoperable.	A.1 Restore channel to OPERABLE status. <u>OR</u>	72 hours (continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. (continued)	A.2	---------NOTE--------- Not applicable if inoperable channel is the result of an inoperable breaker. --------------------- Place channel in trip.	72 hours
B. One or more Functions with EOC-RPT trip capability not maintained. AND MCPR limit for inoperable EOC-RPT not made applicable.	B.1	Restore EOC-RPT trip capability.	2 hours
	OR		
	B.2	Apply the MCPR limit for inoperable EOC-RPT as specified in the COLR.	2 hours
C. Required Action and associated Completion Time not met.	C.1	Remove the associated recirculation pump fast speed breaker from service.	4 hours
	OR		
	C.2	Reduce THERMAL POWER to < ~~30%~~ 26% RTP.	4 hours

SURVEILLANCE REQUIREMENTS

------------------------------------NOTE------------------------------------
When a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided the associated Function maintains EOC-RPT trip capability.

SURVEILLANCE		FREQUENCY
SR 3.3.4.1.1	Perform CHANNEL FUNCTIONAL TEST.	92 days
SR 3.3.4.1.2	Perform CHANNEL CALIBRATION. The Allowable Values shall be: a. TSV—Closure: ≤ 7% closed; and b. TCV Fast Closure, Trip Oil Pressure—Low: ≥ 465 psig.	24 months
SR 3.3.4.1.3	Perform LOGIC SYSTEM FUNCTIONAL TEST, including breaker actuation.	24 months
SR 3.3.4.1.4	Verify TSV—Closure and TCV Fast Closure, Trip Oil Pressure—Low Functions are not bypassed when THERMAL POWER is ≥ ~~30%~~ 26% RTP.	24 months
SR 3.3.4.1.5	------------------NOTE------------------ Breaker arc suppression time may be assumed from the most recent performance of SR 3.3.4.1.6. -- Verify the EOC-RPT SYSTEM RESPONSE TIME is within limits.	24 months on a STAGGERED TEST BASIS

(continued)

Table 3.3.6.1-1 (page 1 of 5)
Primary Containment Isolation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION C.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Main Steam Line Isolation					
a. Reactor Vessel Water Level – Low Low Low, Level 1	1,2,3	2	D	SR 3.3.6.1.1 SR 3.3.6.1.3 SR 3.3.6.1.4 SR 3.3.6.1.5 SR 3.3.6.1.6 SR 3.3.6.1.7	≥ 10.8 inches
b. Main Steam Line Pressure – Low	1	2	E	SR 3.3.6.1.1 SR 3.3.6.1.3 SR 3.3.6.1.4 SR 3.3.6.1.5 SR 3.3.6.1.6 SR 3.3.6.1.7	≥ 746 psig
c. Main Steam Line Flow – High	1,2,3	2 per MSL	D	SR 3.3.6.1.1 SR 3.3.6.1.3 SR 3.3.6.1.4 SR 3.3.6.1.5 SR 3.3.6.1.6 SR 3.3.6.1.7	184.4 ≤ ~~122.8~~ psid
d. Condenser Vacuum – Low	1,2(a), 3(a)	2	D	SR 3.3.6.1.1 SR 3.3.6.1.3 SR 3.3.6.1.4 SR 3.3.6.1.5 SR 3.3.6.1.6	≥ 7.6 inches Hg vacuum
e. Main Steam Line Tunnel Temperature – High	1,2,3	2	D	SR 3.3.6.1.1 SR 3.3.6.1.3 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 170.6°F
f. Main Steam Line Tunnel Differential Temperature – High	1,2,3	2	D	SR 3.3.6.1.1 SR 3.3.6.1.3 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 71.7°F
g. Main Steam Line Tunnel Lead Enclosure Temperature – High	1,2,3	2 per area	D	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 151.6°F(b)
h. Manual Initiation	1,2,3	4	G	SR 3.3.6.1.6	NA

(continued)

(a) With any turbine stop valve not closed.

(b) 151.6°F + (0.6)(T_{amb} - 90°F) and ≤ 175.6°F provided the absence of steam leaks in the main steam line tunnel lead enclosure area is verified by visual inspection prior to establishing the Allowable Value.

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.3.1	-------------------NOTES------------------- 1. Not required to be performed until 4 hours after associated recirculation loop is in operation. 2. Not required to be performed until 24 hours after > 25% RTP. --	23%
	Verify at least two of the following criteria (a, b, and c) are satisfied for each operating recirculation loop: a. Jet pump loop flow versus flow control valve position differs by ≤ 10% from established patterns. b. Jet pump loop flow versus recirculation loop drive flow differs by ≤ 10% from established patterns. c. Each jet pump diffuser to lower plenum differential pressure differs by ≤ 20% from established patterns.	24 hours

3.7 PLANT SYSTEMS

3.7.5 Main Turbine Bypass System

LCO 3.7.5 The Main Turbine Bypass System shall be OPERABLE.

<u>OR</u>

LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)," limits for an inoperable Main Turbine Bypass System, as specified in the COLR, are made applicable.

APPLICABILITY: THERMAL POWER ≥ ~~25%~~ 23% RTP.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. Requirements of the LCO not met.	A.1	Satisfy the requirements of the LCO.	2 hours
B. Required Action and associated Completion Time not met.	B.1	Reduce THERMAL POWER to < ~~25%~~ 23% RTP.	4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.5.1	Perform a system functional test.	24 months
SR 3.7.5.2	Verify the TURBINE BYPASS SYSTEM RESPONSE TIME is within limits.	24 months

ENCLOSURE

ATTACHMENT 2

Technical Specifications Bases Page Markups (Information Only)

Pages Included in this Markup

BASES

APPLICABLE SAFETY ANALYSES

2.1.1.1 Fuel Cladding Integrity (continued)

data taken at pressures from 14.7 psia to 800 psia indicate that the fuel assembly critical power at this flow is approximately 3.35 MWt. With the design peaking factors, this corresponds to a THERMAL POWER > 50% RTP. Thus, a THERMAL POWER limit of 25% RTP for reactor pressure < 785 psig is conservative.

23%

2.1.1.2 MCPR

The fuel cladding integrity SL is set such that no significant fuel damage is calculated to occur if the limit is not violated. Since the parameters that result in fuel damage are not directly observable during reactor operation, the thermal and hydraulic conditions that result in the onset of transition boiling have been used to mark the beginning of the region in which fuel damage could occur. Although it is recognized that the onset of transition boiling would not result in damage to BWR fuel rods, the critical power at which boiling transition is calculated to occur has been adopted as a convenient limit. However, the uncertainties in monitoring the core operating state and in the procedures used to calculate the critical power result in an uncertainty in the value of the critical power. Therefore, the fuel cladding integrity SL is defined as the critical power ratio in the limiting fuel assembly for which more than 99.9% of the fuel rods in the core are expected to avoid boiling transition, considering the power distribution within the core and all uncertainties.

The MCPR SL is determined using a statistical model that combines all the uncertainties in operating parameters and the procedures used to calculate critical power. The probability of the occurrence of boiling transition is determined using the approved General Electric Critical Power correlations. Details of the fuel cladding integrity SL calculation are given in References ~~3 and 4.~~ Reference 3 also includes a tabulation of the uncertainties used in the determination of the MCPR SL and Reference 4 also provides the nominal values of the parameters used in the MCPR SL statistical analysis.

3, 4 and 6.

(continued)

BASES (continued)

REFERENCES

1. 10 CFR 50, Appendix A, GDC 10.
2. GE Service Information Letter No. 516, Supplement 2, "Core Flow Indication in the Low-Flow Region," January 19, 1996.
3. NEDE-24011-P-A, "GE Standard Application for Reactor Fuel," (revision specified in the COLR).
4. Supplemental Reload Licensing Report for Nine Mile Point Nuclear Station Unit 2 (revision specified in the COLR).
5. 10 CFR 50.67, "Accident Source Term."
6. NEDC-33173-P-A, "Applicability of GE Methods to Expanded Operating Domains"

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.7.4 and SR 3.1.7.6 (continued)

manual, power operated, and automatic valves in the SLC System flow path ensures that the proper flow paths will exist for system operation. A valve is also allowed to be in the nonaccident position, provided it can be aligned to the accident position from the control room, or locally by a dedicated operator at the valve control. This is acceptable since the SLC System is a manually initiated system. This Surveillance does not apply to valves that are locked, sealed, or otherwise secured in position, since they were verified to be in the correct position prior to locking, sealing, or securing. This verification of valve alignment does not apply to valves that cannot be inadvertently misaligned, such as check valves. This SR does not require any testing or valve manipulation; rather, it involves verification that those valves capable of being mispositioned are in the correct positions. The 31 day Frequency is based on engineering judgment and is consistent with the procedural controls governing valve operation that ensure correct valve positions.

SR 3.1.7.5

This Surveillance requires an examination of the sodium pentaborate solution by using chemical analysis to ensure the proper concentration of boron (measured in weight % sodium pentaborate decahydrate) exists in the storage tank. SR 3.1.7.5 must be performed anytime boron or water is added to the storage tank solution to establish that the boron solution concentration is within the specified limits. This Surveillance must be performed anytime the temperature is restored to within the limit (i.e., ≥ 70°F), to ensure no significant boron precipitation occurred. The 31 day Frequency of this Surveillance is appropriate because of the relatively slow variation of boron concentration between surveillances.

SR 3.1.7.7 1327

Demonstrating each SLC System pump develops a flow rate ≥ 41.2 gpm at a discharge pressure ≥ ~~1325~~ psig ensures that pump performance has not degraded during the fuel cycle. This minimum pump flow rate requirement ensures that, when

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The APLHGR satisfies Criterion 2 of Reference 4.

LCO

The APLHGR limits specified in the COLR are the result of fuel design and DBA analyses. For two recirculation loops operating, the limit is dependent on bundle exposure. With only one recirculation loop in operation, in conformance with the requirements of LCO 3.4.1, "Recirculation Loops Operating," the limit is determined by multiplying the exposure dependent APLHGR limit by a conservative multiplier determined by a specific single recirculation loop analysis (Ref. 2).

APPLICABILITY

The APLHGR limits are primarily derived from fuel design evaluations and LOCA analyses that are assumed to occur at high power levels. Studies and operating experience have shown that as power is reduced, the margin to the required APLHGR limits increases. This trend continues down to the power range of 5% to 15% RTP when entry into MODE 2 occurs. When in MODE 2, the intermediate range monitor (IRM) scram function provides prompt scram initiation during any significant transient, thereby effectively removing any APLHGR limit compliance concern in MODE 2. Therefore, at THERMAL POWER levels ≤ ~~25%~~ 23% RTP, the reactor operates with substantial margin to the APLHGR limits; thus, this LCO is not required.

ACTIONS

A.1

If any APLHGR exceeds the required limits, an assumption regarding an initial condition of the DBA analyses may not be met. Therefore, prompt action is taken to restore the APLHGR(s) to within the required limits such that the plant will be operating within analyzed conditions and within the design limits of the fuel rods. The 2 hour Completion Time is sufficient to restore the APLHGR(s) to within its limits and is acceptable based on the low probability of a DBA occurring simultaneously with the APLHGR out of specification.

(continued)

BASES

ACTIONS (continued)

B.1

If the APLHGR cannot be restored to within its required limits within the associated Completion Time, the plant must be brought to a MODE or other specified condition in which the LCO does not apply. To achieve this status, THERMAL POWER must be reduced to < ~~25%~~ 23% RTP within 4 hours. The allowed Completion Time is reasonable, based on operating experience, to reduce THERMAL POWER to < ~~25%~~ 23% RTP in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.2.1.1

APLHGRs are required to be initially calculated within 12 hours after THERMAL POWER is ≥ ~~25%~~ 23% RTP and then every 24 hours thereafter. They are compared to the specified limits in the COLR to ensure that the reactor is operating within the assumptions of the safety analysis. The 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%~~ 23% RTP is achieved is acceptable given the large inherent margin to operating limits at low power levels.

REFERENCES

1. NEDE-24011-P-A, "GE Standard Application for Reactor Fuel," (revision specified in the COLR).
2. USAR, Chapter 15B.
3. USAR, Chapter 15G.
4. 10 CFR 50.36(c)(2)(ii).

BASES

APPLICABLE SAFETY ANALYSES (continued)

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

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

The MCPR satisfies Criterion 2 of Reference 4.

LCO

The MCPR operating limits specified in the COLR are the result of the Design Basis Accident (DBA) and transient analysis. The MCPR operating limit is determined by the larger of the $MCPR_f$ and $MCPR_p$ limits.

23%

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. 23%

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

23%

(continued)

23%

BASES

APPLICABILITY (continued)

occurs. When in MODE 2, the intermediate range monitor (IRM) provides rapid scram initiation for any significant power increase transient, which effectively eliminates any MCPR compliance concern. Therefore, at THERMAL POWER levels ≤ ~~25%~~ 23% RTP, the reactor is operating with substantial margin to the MCPR limits and this LCO is not required.

ACTIONS

A.1

If any MCPR is outside the required limits, an assumption regarding an initial condition of the design basis transient analyses may not be met. Therefore, prompt action should be taken to restore the MCPR(s) to within the required limits such that the plant remains operating within analyzed conditions. The 2 hour Completion Time is normally sufficient to restore the MCPR(s) to within its limits and is acceptable based on the low probability of a transient or DBA occurring simultaneously with the MCPR out of specification.

B.1

If the MCPR cannot be restored to within the required limits within the associated Completion Time, the plant must be brought to a MODE or other specified condition in which the LCO does not apply. To achieve this status, THERMAL POWER must be reduced to < ~~25%~~ 23% RTP within 4 hours. The allowed Completion Time is reasonable, based on operating experience, to reduce THERMAL POWER to < ~~25%~~ 23% RTP in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.2.2.1

The MCPR is required to be initially calculated within 12 hours after THERMAL POWER is ≥ ~~25%~~ 23% RTP and then every 24 hours thereafter. It is compared to the specified limits in the COLR to ensure that the reactor is operating within the assumptions of the safety analysis. The 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 reaches ≥ ~~25%~~ 23% RTP is acceptable given the large inherent margin to operating limits at low power levels.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

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

The LHGR satisfies Criterion 2 of Reference 4.

LCO

The LHGR is a basic assumption in the fuel design analysis. The fuel has been designed to operate at rated core power with sufficient design margin to the LHGR calculated to cause a 1% fuel cladding plastic strain. The operating limit to accomplish this objective is specified in the COLR.

23%

APPLICABILITY

The LHGR limits are derived from fuel design analysis that is limiting at high power level conditions. At core thermal power levels < 25% RTP, the reactor is operating with a substantial margin to the LHGR limits and, therefore, the Specification is only required when the reactor is operating at ≥ 25% RTP.

23%

ACTIONS

A.1

If any LHGR exceeds its required limit, an assumption regarding an initial condition of the fuel design analysis is not met. Therefore, prompt action should be taken to restore the LHGR(s) to within its required limits such that the plant is operating within analyzed conditions. The 2 hour Completion Time is normally sufficient to restore the LHGR(s) to within its limits and is acceptable based on the low probability of a transient or Design Basis Accident occurring simultaneously with the LHGR out of specification.

(continued)

BASES

ACTIONS (continued)

B.1

If the LHGR cannot be restored to within its required limits within the associated Completion Time, the plant must be brought to a MODE or other specified condition in which the LCO does not apply. To achieve this status, THERMAL POWER must be reduced to < ~~25%~~ 23% RTP within 4 hours. The allowed Completion Time is reasonable, based on operating experience, to reduce THERMAL POWER to < ~~25%~~ 23% RTP in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.2.3.1

The LHGRs are required to be initially calculated within 12 hours after THERMAL POWER is ≥ ~~25%~~ 23% 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%~~ 23% 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).
5. 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, LCO, and APPLICABILITY

<u>2.a. Average Power Range Monitor Neutron Flux—Upscale, Setdown</u> (continued)

With the IRMs at Range 9 or 10, it is possible that the Average Power Range Monitor Neutron Flux—Upscale, Setdown Function will provide the primary trip signal for a core-wide increase in power.

No specific safety analyses take direct credit for the Average Power Range Monitor Neutron Flux—Upscale, Setdown Function. However, this Function indirectly ensures that, before the reactor mode switch is placed in the run position, reactor power does not exceed ~~25%~~ 23% RTP (SL 2.1.1.1) when operating at low reactor pressure and low core flow. Therefore, it indirectly prevents fuel damage during significant reactivity increases with THERMAL POWER < ~~25%~~ 23% RTP.

The APRM System is divided into four APRMs, each providing an input into both trip systems via the 2-Out-Of-4 Voter channels, Function 2.f. Each APRM inputs to all four 2-Out-Of-4 Voter channels, with each APRM input into a 2-Out-Of-4 Voter channel considered a channel. Thus, there are a total of 16 Average Power Range Monitor Neutron Flux—Upscale, Setdown channels, with eight channels per trip system and four channels per logic channel. The system is designed to allow one APRM to be bypassed (and since the APRM provides an input to all four 2-Out-Of-4 Voter channels, one channel in each logic channel is effectively bypassed). Any two APRM channels in a logic channel can cause the associated trip system to trip. Since each APRM inputs into both trip systems, this effectively means that when two APRMs provide a Neutron Flux—Upscale, Setdown signal, two channels in both logic channels in each trip system will trip, producing a scram. Twelve channels of Average Power Range Monitor Neutron Flux—Upscale, Setdown, with three channels per logic channel in each trip system are required to be OPERABLE to ensure that no single failure will preclude a scram from this Function on a valid signal. In addition, to provide adequate coverage of the entire core, at least 20 LPRM inputs are required for each APRM, with at least three LPRM inputs from each of the four axial levels at which the LPRMs are located.

The Allowable Value is based on preventing significant increases in power when THERMAL POWER is < ~~25%~~ 23% RTP.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

2.b. Average Power Range Monitor Flow Biased Simulated Thermal Power—Upscale (continued)

2-Out-Of-4 Voter channels, with each APRM input into a 2-Out-Of-4 Voter channel considered a channel. Thus, there are a total of 16 Average Power Range Monitor Flow Biased Simulated Thermal Power—Upscale channels, with eight channels per trip system and four channels per logic channel. The system is designed to allow one APRM to be bypassed (and since the APRM provides an input to all four 2-Out-Of-4 Voter channels, one channel in each logic channel is effectively bypassed). Any two APRM channels in a logic channel can cause the associated trip system to trip. Since each APRM inputs into both trip systems, this effectively means that when two APRMs provide a Flow Biased Simulated Thermal Power—Upscale signal, two channels in both logic channels in each trip system will trip, producing a scram. Twelve channels of Average Power Range Monitor Flow Biased Simulated Thermal Power—Upscale, with three channels per logic channel in each trip system are required to be OPERABLE to ensure that no single instrument failure will preclude a scram from this Function on a valid signal. In addition, to provide adequate coverage of the entire core, at least 20 LPRM inputs are required for each APRM, with at least three LPRM inputs from each of the four axial levels at which the LPRMs are located. Each APRM receives two flow signals from two flow transmitters, one from each reactor recirculation loop. The total recirculation drive flow signal is generated by the flow processing logic part of the APRM, by summing the flow calculated from these two flow transmitter signal inputs. Each APRM receives flow signals from different flow transmitters (a total of eight flow transmitters).

The nominal trip setpoint for this function is 0.55W + 57.5 % RTP (or 0.50 (W-5 %) + 50.5 % RTP when reset for single loop operation). The nominal trip setpoint, and the As-Found and As-Left tolerances were determined in accordance with the setpoint methodology of Reference 16.

No specific safety analyses take direct credit for the Average Power Range Monitor Flow Biased Simulated Thermal Power—Upscale Function. Originally, the clamped Allowable Value was based on analyses that took credit for the Average Power Range Monitor 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 a clamp is conservative. The THERMAL POWER time constant of ≤ 6.6 seconds is based on the fuel heat transfer dynamics and provides a signal that is proportional to the THERMAL POWER.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

2.e. Average Power Range Monitor OPRM—Upscale (continued)

An OPRM—Upscale trip can also be generated if either the growth rate or amplitude based algorithms detect growing oscillatory changes in the neutron flux for one or more cells. However, this portion of the trip is not required by this Specification; only the period based algorithm is required for OPERABILITY.

The APRM System is divided into four APRMs, each providing an input into both trip systems via the 2-Out-Of-4 Voter channels, Function 2.f. Each APRM inputs to all four 2-Out-Of-4 Voter channels, with each APRM input into a 2-Out-Of-4 Voter channel considered a channel. Thus, there are a total of 16 Average Power Range Monitor OPRM—Upscale channels, with eight channels per trip system and four channels per logic channel. The system is designed to allow one APRM to be bypassed (and since the APRM provides an input to all four 2-Out-Of-4 Voter channels, one channel in each logic channel is effectively bypassed). Any two APRM channels in a logic channel can cause the associated trip system to trip. Since each APRM inputs into both trip systems, this effectively means that when two APRMs provide an OPRM—Upscale signal, two channels in both logic channels in each trip system will trip, producing a scram. Twelve channels of Average Power Range Monitor OPRM—Upscale with three channels per logic channel in each trip system are required to be OPERABLE to ensure that no single instrument failure will preclude a scram from this Function on a valid signal. In addition, to provide adequate coverage of the entire core, a minimum of 21 cells, each with a minimum of two LPRMs, are required for each OPRM—Upscale Function.

The Allowable Value, which is specified in the COLR, is based on ensuring the MCPR Safety Limit is not exceeded due to anticipated thermal-hydraulic power oscillations.

The Average Power Range Monitor OPRM—Upscale Function automatic trip is only enabled when THERMAL POWER, as determined by APRM Simulated Thermal Power, is ≥ ~~30%~~ 26% RTP and reactor core flow, as indicated by recirculation drive flow, is < 60% of rated recirculation drive flow. This is the operating region where actual thermal-hydraulic oscillations may occur. However, the Average Power Range Monitor OPRM—Upscale Function is required to be OPERABLE at all times while in MODE 1. When the automatic trip is bypassed, the Average Power Range Monitor OPRM—Upscale Function is

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

8. Turbine Stop Valve—Closure (continued)

RPS trip system receives an input from four Turbine Stop Valve—Closure channels, each consisting of one valve stem position switch (which is common to a channel in the other RPS trip system) and a switch contact. The logic for the Turbine Stop Valve—Closure Function is such that three or more TSVs must be closed to produce a scram. In addition, certain combinations of two valves closed will result in a half scram.

This Function must be enabled at THERMAL POWER ≥ ~~30%~~ 26% RTP. This is normally accomplished automatically by pressure transmitters sensing turbine first stage pressure; therefore, opening the turbine bypass valves may affect this Function.

The Turbine Stop Valve—Closure, Allowable Value is selected to detect imminent TSV closure thereby reducing the severity of the subsequent pressure transient.

Eight channels of Turbine Stop Valve—Closure, with four channels in each trip system, are required to be OPERABLE to ensure that no single instrument failure will preclude a scram from this Function if the TSVs should close. This Function is required, consistent with analysis assumptions, whenever THERMAL POWER is ≥ ~~30%~~ 26% RTP. This Function is not required when THERMAL POWER is < ~~30%~~ 26% RTP since the Reactor Vessel Steam Dome Pressure—High and the Average Power Range Monitor Fixed Neutron Flux—Upscale Functions are adequate to maintain the necessary safety margins.

9. Turbine Control Valve Fast Closure, Trip Oil Pressure—Low

Fast closure of the TCVs results in the loss of a heat sink that produces reactor pressure, neutron flux, and heat flux transients that must be limited. Therefore, a reactor scram is initiated on TCV fast closure in anticipation of the transients that would result from the closure of these valves. The Turbine Control Valve Fast Closure, Trip Oil Pressure—Low Function is the primary scram signal for the generator load rejection event analyzed in Reference 4. For this event, the reactor scram reduces the amount of energy required to be absorbed and, along with the actions of the EOC-RPT System, ensures that the MCPR SL is not exceeded.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

9. Turbine Control Valve Fast Closure, Trip Oil Pressure—Low (continued)

Turbine Control Valve Fast Closure, Trip Oil Pressure—Low signals are initiated by the EHC fluid pressure at each control valve. There is one pressure switch associated with each control valve, the signal from each switch being assigned to a separate RPS logic channel. This Function must be enabled at THERMAL POWER ≥ ~~30%~~ 26% RTP. This is normally accomplished automatically by pressure transmitters sensing turbine first stage pressure; therefore, opening the turbine bypass valves may affect this Function.

The Turbine Control Valve Fast Closure, Trip Oil Pressure—Low Allowable Value is selected high enough to detect imminent TCV fast closure.

Four channels of Turbine Control Valve Fast Closure, Trip Oil Pressure—Low Function, with two channels in each trip system arranged in a one-out-of-two logic, are required to be OPERABLE to ensure that no single instrument failure will preclude a scram from this Function on a valid signal. This Function is required, consistent with the analysis assumptions, whenever THERMAL POWER is ≥ ~~30%~~ 26% RTP. This Function is not required when THERMAL POWER is < ~~30%~~ 26% RTP since the Reactor Vessel Steam Dome Pressure—High and the Average Power Range Monitor Fixed Neutron Flux—Upscale Functions are adequate to maintain the necessary safety margins.

10. Reactor Mode Switch—Shutdown Position

The Reactor Mode Switch—Shutdown Position Function provides signals, via the manual scram logic channels, that are redundant to the automatic protective instrumentation channels and provide manual reactor trip capability. This Function was not specifically credited in the accident analysis, but it is retained for the overall redundancy and diversity of the RPS as required by the NRC approved licensing basis.

The reactor mode switch is a single switch with four channels (one from each of the four independent banks of contacts), each of which inputs into one of the RPS logic channels.

(continued)

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. 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. 1

An allowance is provided that requires the SR to be performed only at ≥ ~~25%~~ 23% RTP because it is difficult to accurately maintain APRM indication of core THERMAL POWER

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.1.3 (continued)

consistent with a heat balance when < ~~25%~~ 23% RTP. At low power levels, a high degree of accuracy is unnecessary because of the large inherent margin to thermal limits (MCPR, APLHGR, and LHGR). At ≥ ~~25%~~ 23% RTP, the Surveillance is required to have been satisfactorily performed within the last 7 days in accordance with SR 3.0.2. A Note is provided which allows an increase in THERMAL POWER above ~~25%~~ 23% if the 7 day Frequency is not met per SR 3.0.2. In this event, the SR must be performed within 12 hours after reaching or exceeding ~~25%~~ 23% RTP. Twelve hours is based on operating experience and in consideration of providing a reasonable time in which to complete the SR.

SR 3.3.1.1.4

A CHANNEL FUNCTIONAL TEST is performed on each required channel to ensure that the channel will perform the intended function. Any setpoint adjustment shall be consistent with the assumptions of the current plant specific setpoint methodology.

As noted, for Functions 1.a and 1.b, SR 3.3.1.1.4 is not required to be performed when entering MODE 2 from MODE 1 since testing of the MODE 2 required IRM Functions cannot be performed in MODE 1 without utilizing jumpers, lifted leads, or movable links. This allows entry into MODE 2 if the 7 day Frequency is not met per SR 3.0.2. In this event, the SR must be performed within 12 hours after entering MODE 2 from MODE 1. Twelve hours is based on operating experience and in consideration of providing a reasonable time in which to complete the SR.

A Frequency of 7 days provides an acceptable level of system average unavailability over the Frequency interval and is based on reliability analysis (Ref. 10). (The Manual Scram Function CHANNEL FUNCTIONAL TEST Frequency was credited in the analysis to extend many automatic scram Functions Frequencies.)

SR 3.3.1.1.5 and SR 3.3.1.1.6

These Surveillances are established to ensure that no gaps in neutron flux indication exist from subcritical to power

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.1.11 and SR 3.3.1.1.13 (continued)

"Show Parameters" display. This is acceptable because, other than the flow and LPRM input processing, all OPRM functional processing is performed digitally involving equipment or components that cannot be calibrated. The Frequency of SR 3.3.1.1.11 is based upon the assumption of a 18 month calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis. The Frequency of SR 3.3.1.1.13 is based on the assumption of a 24 month calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis.

SR 3.3.1.1.14

The LOGIC SYSTEM FUNCTIONAL TEST (LSFT) demonstrates the OPERABILITY of the required trip logic for a specific channel. The functional testing of control rods, in LCO 3.1.3, "Control Rod OPERABILITY," and SDV vent and drain valves, in LCO 3.1.8, "Scram Discharge Volume (SDV) Vent and Drain Valves," overlaps this Surveillance to provide complete testing of the assumed safety function. In addition, for Function 2.f, the LSFT includes simulating APRM trip conditions at the APRM channel inputs to the 2-Out-Of-4 Voter channel to check all combinations of two tripped APRM channel inputs to the 2-Out-Of-4 Voter logic in the 2-Out-Of-4 Voter channels.

The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the 24 month Frequency.

SR 3.3.1.1.15

26%

This SR ensures that scrams initiated from the Turbine Stop Valve—Closure and Turbine Control Valve Fast Closure, Trip Oil Pressure—Low Functions will not be inadvertently bypassed when THERMAL POWER is ≥ ~~30%~~ RTP. This involves calibration of the bypass channels. Adequate margins for the instrument setpoint methodology are incorporated into the Allowable Value and the actual setpoint. Because main

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.1.15 (continued)

26%

turbine bypass flow can affect this setpoint nonconservatively (THERMAL POWER is derived from turbine first stage pressure), the main turbine bypass valves must remain closed during an in-service calibration at THERMAL POWER ≥ ~~30%~~ RTP to ensure that the calibration is valid.

26%

If any bypass channel setpoint is nonconservative (i.e., the Functions are bypassed at ≥ ~~30%~~ RTP, either due to open main turbine bypass valve(s) or other reasons), then the affected Turbine Stop Valve – Closure and Turbine Control Valve Fast Closure, Trip Oil Pressure – Low Functions are considered inoperable. Alternatively, the bypass channel can be placed in the conservative condition (nonbypass). If placed in the nonbypass condition, this SR is met and the channel is considered OPERABLE.

The Frequency of 24 months is based on engineering judgment and reliability of the components.

SR 3.3.1.1.16

26%

This SR ensures that scrams initiated from the APRM OPRM – Upscale Function will not be inadvertently bypassed when THERMAL POWER is ≥ ~~30%~~ RTP and recirculation drive flow is < 60% rated recirculation drive flow.

If any bypass channel setpoint is nonconservative (i.e., the Function is bypassed at ≥ ~~30%~~ RTP and < 60% rated recirculation drive flow), then the affected channel is considered inoperable.

26%

The Frequency of 24 months is based on Ref. 15.

SR 3.3.1.1.17

This SR ensures that the individual channel response times are less than or equal to the maximum values assumed in the accident analysis. The RPS RESPONSE TIME acceptance criteria are included in Reference 12.

(continued)

BASES

REFERENCES (continued)

8. USAR, Section 15.4.9.

9. Letter, P. Check (NRC) to G. Lainas (NRC), "BWR Scram Discharge System Safety Evaluation," December 1, 1980.

10. NEDO-30851-P-A, "Technical Specification Improvement Analyses for BWR Reactor Protection System," March 1988.

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

12. Technical Requirements Manual.

13. NEDO-32291-A, "System Analyses for the Elimination of Selected Response Time Testing Requirements," October 1995.

14. USAR, Section 7.6.1.4.3.

15. NEDC-32410-P-A, "NUMAC-PRNM Retrofit Plus Option III Stability Trip Functions, Supplement 1," November 1997.

16. NEDC- 31336P-A, GE Nuclear Energy, "General Electric Instrument Setpoint Methodology", Class III (Proprietary), September 1996.

B 3.3 INSTRUMENTATION

B 3.3.2.2 Feedwater System and Main Turbine High Water Level Trip Instrumentation

BASES

BACKGROUND

The Feedwater System and Main Turbine High Water Level Trip Instrumentation is designed to detect a potential failure of the Feedwater Level Control System that causes excessive feedwater flow.

With excessive feedwater flow, the water level in the reactor vessel rises toward the high water level, Level 8 reference point, causing the trip of the three feedwater pumps and the main turbine.

Reactor Vessel Water Level—High, Level 8 signals are provided by differential pressure transmitters that sense the difference between the pressure due to a constant column of water (reference leg) and the pressure due to the actual water level in the reactor vessel (variable leg). Three channels of Reactor Vessel Water Level—High, Level 8 instrumentation are provided as input to a two-out-of-three initiation logic that trips the three feedwater pumps and the main turbine. The channels include electronic equipment (e.g., trip units) that compares measured input signals with pre-established setpoints. When the setpoint is exceeded, the channel output relay actuates, which then outputs a feedwater pump and main turbine trip signal to the trip logic.

A trip of the feedwater pumps limits further increase in reactor vessel water level by limiting further addition of feedwater to the reactor vessel. A trip of the main turbine and closure of the stop valves protects the turbine from damage due to water entering the turbine.

APPLICABLE SAFETY ANALYSES

The Feedwater System and Main Turbine High Water Level Trip Instrumentation is assumed to be capable of providing a feedwater pump and main turbine trip in the design basis transient analysis for a feedwater controller failure, maximum demand event (Ref. 1) and in other design basis events in Reference 2. The Level 8 trip indirectly initiates a reactor scram from the main turbine trip (above 26% ~~30%~~ RTP) and trips the feedwater pumps, thereby terminating the event. The reactor scram mitigates the reduction in MCPR.

(continued)

BASES

LCO (continued)

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

23%

APPLICABILITY

The Feedwater System and Main Turbine High Water Level Trip Instrumentation is required to be OPERABLE at ≥ 25% RTP to ensure that the fuel cladding integrity Safety Limit and the cladding 1% plastic strain limit are not violated during the feedwater controller failure, maximum demand event. As discussed in the Bases for LCO 3.2.1, "Average Planar Linear Heat Generation Rate (APLHGR)," and LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)," sufficient margin to these limits exists below 25% RTP; therefore, these requirements are only necessary when operating at or above this power level.

23%

ACTIONS

A Note has been provided to modify the ACTIONS related to Feedwater System and Main Turbine High Water Level Trip Instrumentation channels. Section 1.3, Completion Times, specifies that once a Condition has been entered, subsequent divisions subsystems, components, or variables expressed in the Condition, discovered to be inoperable or not within limits, will not result in separate entry into the Condition. Section 1.3 also specifies that Required Actions of the Condition continue to apply for each additional failure, with Completion Times based on initial entry into the Condition. However, the Required Actions for inoperable Feedwater System and Main Turbine High Water Level Trip Instrumentation channels provide appropriate compensatory measures for separate inoperable channels. As such, a Note has been provided that allows separate Condition entry for each inoperable Feedwater System and Main Turbine High Water Level Trip Instrumentation channel.

A.1

With one channel inoperable, the remaining two OPERABLE channels can provide the required trip signal. However, overall instrumentation reliability is reduced because a single failure in one of the remaining channels concurrent with feedwater controller failure, maximum demand event, may result in the instrumentation not being able to perform its intended function. Therefore, continued operation is only allowed for a limited time with one channel inoperable. If the inoperable channel cannot be restored to OPERABLE status

(continued)

BASES

ACTIONS (continued)

C.1 and C.2

With a channel not restored to OPERABLE status or placed in trip, THERMAL POWER must be reduced to < ~~25%~~ 23% RTP within 4 hours. As discussed in the Applicability section of the Bases, operation below ~~25%~~ 23% RTP results in sufficient margin to the required limits, and the Feedwater System and Main Turbine High Water Level Trip Instrumentation is not required to protect fuel integrity during the feedwater controller failure, maximum demand event. Alternately, if a channel is inoperable solely due to an inoperable feedwater pump breaker, the affected feedwater pump breaker may be removed from service since this performs the intended function of the instrumentation. The allowed Completion Time of 4 hours is based on operating experience to reduce THERMAL POWER to < ~~25%~~ 23% RTP from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

The Surveillances are modified by a Note to indicate that when a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided the Function maintains feedwater system and main turbine high water level trip capability. Upon completion of the Surveillance, or expiration of the 6 hour allowance, the channel must be returned to OPERABLE status or the applicable Condition entered and Required Actions taken. This Note is based on the reliability analysis (Ref. 4) assumption that 6 hours is the average time required to perform channel Surveillance. That analysis demonstrated that the 6 hour testing allowance does not significantly reduce the probability that the feedwater pumps and main turbine will trip when necessary.

SR 3.3.2.2.1

Performance of the CHANNEL CHECK once every 24 hours 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 instrument channels could be an indication of excessive instrument drift in one of the channels, or something even more serious. A CHANNEL CHECK will detect

(continued)

BASES

BACKGROUND (continued)

trip system to actuate. If either trip system actuates, both recirculation pumps, if operating in fast speed, will trip. There are two EOC-RPT breakers in series per recirculation pump. One trip system trips one of the two EOC-RPT breakers for each recirculation pump and the second trip system trips the other EOC-RPT breaker for each recirculation pump.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

The TSV—Closure and the TCV Fast Closure, Trip Oil Pressure—Low Functions are designed to trip the recirculation pumps, if operating in fast speed, in the event of a turbine trip or generator load rejection to mitigate the neutron flux, heat flux and pressurization transients, and to increase the margin to the MCPR SL. The analytical methods and assumptions used in evaluating the turbine trip and generator load rejection, as well as other safety analyses that assume EOC-RPT, are summarized in Reference 2.

To mitigate pressurization transient effects, the EOC-RPT must trip the recirculation pumps, if operating in fast speed, after initiation of initial closure movement of either the TSVs or the TCVs. The combined effects of this trip and a scram reduce fuel bundle power more rapidly than does a scram alone, resulting in an increased margin to the MCPR SL. Alternatively, MCPR limits for an inoperable EOC-RPT as specified in the COLR are sufficient to mitigate pressurization transient effects. The EOC-RPT function is automatically disabled when THERMAL POWER, as sensed by turbine first stage pressure, is < ~~30%~~ 26% RTP.

EOC-RPT instrumentation satisfies Criterion 3 of Reference 3.

The OPERABILITY of the EOC-RPT is dependent on the OPERABILITY of the individual instrumentation channel Functions. Each Function must have a required number of OPERABLE channels in each trip system, with their setpoints within the specified Allowable Value of SR 3.3.4.1.2. The actual setpoint is calibrated consistent with applicable setpoint methodology assumptions. Channel OPERABILITY also includes the associated EOC-RPT breakers. Each channel (including the associated EOC-RPT breakers) must also respond within its assumed response time.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

Turbine Stop Valve—Closure

Closure of the TSVs and a main turbine trip result in the loss of a heat sink that produces reactor pressure, neutron flux, and heat flux transients that must be limited. Therefore, an RPT is initiated on TSV—Closure, in anticipation of the transients that would result from closure of these valves. EOC-RPT decreases reactor power and aids the reactor scram in ensuring the MCPR SL is not exceeded during the worst case transient.

Closure of the TSVs is determined by measuring the position of each stop valve. There is one valve stem position switch associated with each stop valve, and the signal from each switch is assigned to a separate trip channel. The logic for the TSV—Closure Function is such that two or more TSVs must be closed to produce an EOC-RPT. This Function must be enabled at THERMAL POWER ≥ ~~30%~~ 26% RTP. This is normally accomplished automatically by pressure transmitters sensing turbine first stage pressure; therefore, opening of the turbine bypass valves may affect this Function. Four channels of TSV—Closure, with two channels in each trip system, are available and required to be OPERABLE to ensure that no single instrument failure will preclude an EOC-RPT from this Function on a valid signal. The TSV—Closure Allowable Value is selected to detect imminent TSV closure.

This protection is required, consistent with the safety analysis assumptions, whenever THERMAL POWER is ≥ ~~30%~~ 26% RTP with any recirculating pump in fast speed. Below ~~30%~~ 26% RTP or with the recirculation in slow speed, the Reactor Vessel Steam Dome Pressure—High and the Average Power Range Monitor (APRM) Fixed Neutron Flux—Upscale Functions of the Reactor Protection System (RPS) are adequate to maintain the necessary safety margins.

TCV Fast Closure, Trip Oil Pressure—Low

Fast closure of the TCVs during a generator load rejection results in the loss of a heat sink that produces reactor pressure, neutron flux, and heat flux transients that must be limited. Therefore, an RPT is initiated on TCV Fast Closure, Trip Oil Pressure—Low in anticipation of the transients that would result from the closure of these

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

<u>TCV Fast Closure, Trip Oil Pressure—Low</u> (continued)

valves. The EOC-RPT decreases reactor power and aids the reactor scram in ensuring that the MCPR SL is not exceeded during the worst case transient.

Fast closure of the TCVs is determined by measuring the EHC fluid pressure at each control valve. There is one pressure switch associated with each control valve, and the signal from each switch is assigned to a separate trip channel. The logic for the TCV Fast Closure, Trip Oil Pressure—Low Function is such that two or more TCVs must be closed (pressure switch trips) to produce an EOC-RPT. This Function must be enabled at THERMAL POWER ≥ ~~30%~~ 26% RTP. This is normally accomplished automatically by pressure transmitters sensing turbine first stage pressure; therefore, opening of the turbine bypass valves may affect this Function. Four channels of TCV Fast Closure, Trip Oil Pressure—Low, with two channels in each trip system, are available and required to be OPERABLE to ensure that no single instrument failure will preclude an EOC-RPT from this Function on a valid signal. The TCV Fast Closure, Trip Oil Pressure—Low Allowable Value is selected high enough to detect imminent TCV fast closure.

This protection is required consistent with the analysis, whenever the THERMAL POWER is ≥ ~~30%~~ 26% RTP with any recirculation pump in fast speed. Below ~~30%~~ 26% RTP or with the recirculation pumps in slow speed, the Reactor Vessel Steam Dome Pressure—High and the APRM Fixed Neutron Flux—Upscale Functions of the RPS are adequate to maintain the necessary safety margins. The turbine first stage pressure/reactor power relationship for the setpoint of the automatic enable is identical to that described for TSV closure.

ACTIONS

A Note has been provided to modify the ACTIONS related to EOC-RPT instrumentation channels. Section 1.3, Completion Times, specifies that once a Condition has been entered, subsequent divisions, subsystems, components, or variables expressed in the Condition, discovered to be inoperable or not within limits, will not result in separate entry into the Condition. Section 1.3 also specifies that Required Actions of the Condition continue to apply for each additional failure, with Completion Times based on initial entry into the Condition. However, the Required Actions for

(continued)

BASES

ACTIONS

B.1 and B.2 (continued)

Function not maintaining EOC-RPT trip capability. A Function is considered to be maintaining EOC-RPT trip capability when sufficient channels are OPERABLE or in trip, such that the EOC-RPT System will generate a trip signal from the given Function on a valid signal and both recirculation pumps, if operating in fast speed, can be tripped. This requires two channels of the Function, in the same trip system, to each be OPERABLE or in trip, and the associated EOC-RPT breakers to be OPERABLE or in trip. Alternatively, Required Action B.2 requires the MCPR limit for inoperable EOC-RPT, as specified in the COLR, to be applied. This also restores the margin to MCPR assumed in the safety analysis.

The 2 hour Completion Time is sufficient for the operator to take corrective action, and takes into account the likelihood of an event requiring actuation of the EOC-RPT instrumentation during this period. It is also consistent with the 2 hour Completion Time provided in LCO 3.2.2, Required Action A.1, since this instrumentation's purpose is to preclude a MCPR violation.

C.1 and C.2

With any Required Action and associated Completion Time not met, THERMAL POWER must be reduced to < ~~30%~~ 26% RTP within 4 hours. Alternately, the associated recirculation pump fast speed breaker may be removed from service since this performs the intended function of the instrumentation. The allowed Completion Time of 4 hours is reasonable, based on operating experience, to reduce THERMAL POWER to < ~~30%~~ 26% RTP from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

The Surveillances are modified by a Note to indicate that when a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours, provided the associated Function maintains EOC-RPT trip capability. Upon completion of the Surveillance, or expiration of the 6 hour allowance, the channel must be returned to OPERABLE status or the applicable Condition

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.4.1.3 (continued)

The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power.

Operating experience has shown these components usually pass the Surveillance test when performed at the 24 month Frequency.

SR 3.3.4.1.4 26%

This SR ensures that an EOC-RPT initiated from the TSV – Closure and TCV Fast Closure, Trip Oil Pressure – Low Functions will not be inadvertently bypassed when THERMAL POWER is ≥ ~~30%~~ RTP. This involves calibration of the bypass channels. Adequate margins for the instrument setpoint methodologies are incorporated into the actual setpoint. Because main turbine bypass flow can affect this setpoint nonconservatively (THERMAL POWER is derived from first stage pressure), the main turbine bypass valves must
26% remain closed during an in-service calibration at THERMAL POWER ≥ ~~30%~~ RTP to ensure that the calibration remains valid. If any bypass channel's setpoint is nonconservative (i.e., the Functions are bypassed at ≥ ~~30%~~ RTP either due to open main turbine bypass valves or other reasons), the affected TSV – Closure and TCV Fast Closure, Trip Oil Pressure – Low Functions are considered inoperable. 26%
Alternatively, the bypass channel can be placed in the conservative condition (nonbypass). If placed in the nonbypass condition, this SR is met and the channel considered OPERABLE.

The Frequency of 24 months is based on engineering judgment and reliability of the components.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

1.b. Main Steam Line Pressure – Low (continued)

Function closes the MSIVs prior to pressure decreasing below 766 psig, which results in a scram due to MSIV closure, thus reducing reactor power to < 25% RTP.) 23%

The MSL low pressure signals are initiated from four pressure transmitters that are connected to the MSL header. The transmitters are arranged such that, even though physically separated from each other, each transmitter is able to detect low MSL pressure. Four channels of Main Steam Line Pressure – Low Function are available and are required to be OPERABLE to ensure that no single instrument failure can preclude the isolation function.

The Allowable Value was selected to be high enough to prevent excessive RPV depressurization.

The Main Steam Line Pressure – Low Function is only required to be OPERABLE in MODE 1 since this is when the assumed transient can occur (Ref. 4).

This Function isolates the Group 1 valves.

1.c. Main Steam Line Flow – High

Main Steam Line Flow – High is provided to detect a break of the MSL and to initiate closure of the MSIVs. If the steam were allowed to continue flowing out of the break, the reactor would depressurize and the core could uncover. If the RPV water level decreases too far, fuel damage could occur. Therefore, the isolation is initiated on high flow to prevent or minimize core damage. The Main Steam Line Flow – High Function is directly assumed in the analysis of the main steam line break (MSLB) accident (Ref. 6). The isolation action, along with the scram function of the RPS, ensures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46 and offsite doses do not exceed the 10 CFR 50.67 limits.

There is a plant specific program that verifies that this instrument channel functions as required by verifying the As-Found and As-Left settings are consistent with those established by the setpoint methodology.

The MSL flow signals are initiated from 16 differential pressure transmitters that are connected to the four MSLs (the differential pressure transmitters sense differential pressure across a flow venturi). The transmitters are arranged such that, even though physically separated from

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.3.1 (continued)

Individual jet pumps in a recirculation loop typically do not have the same flow. The unequal flow is due to the drive flow manifold, which does not distribute flow equally to all risers. The jet pump diffuser to lower plenum differential pressure pattern or relationship of one jet pump to the loop average is repeatable. An appreciable change in this relationship is an indication that increased (or reduced) resistance has occurred in one of the jet pumps.

The deviations from normal are considered indicative of a potential problem in the recirculation drive flow or jet pump system (Ref. 3). Normal flow ranges and established jet pump differential pressure patterns are established by plotting historical data as discussed in Reference 3.

The 24 hour Frequency has been shown by operating experience to be adequate to verify jet pump OPERABILITY and is consistent with the Frequency for recirculation loop OPERABILITY verification.

This SR is modified by two Notes. Note 1 allows this Surveillance not to be performed until 4 hours after the associated recirculation loop is in operation, since these checks can only be performed during jet pump operation. The 4 hours is an acceptable time to establish conditions appropriate for data collection and evaluation.

23%

Note 2 allows this SR not to be performed until 24 hours after THERMAL POWER exceeds ~~25%~~ RTP. During low flow conditions, jet pump noise approaches the threshold response of the associated flow instrumentation and precludes the collection of repeatable and meaningful data. The 24 hours is an acceptable time to establish conditions appropriate to perform this SR.

REFERENCES

1. USAR, Section 6.3.

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

3. GE Service Information Letter No. 330 including Supplement 1, "Jet Pump Beam Cracks," June 9, 1980.

(continued)

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.11 RCS Pressure and Temperature (P/T) Limits

BASES

BACKGROUND

All components of the RCS are designed to withstand effects of cyclic loads due to system pressure and temperature changes. These loads are introduced by startup (heatup) and shutdown (cooldown) operations, power transients, and reactor trips. This LCO limits the pressure and temperature changes during RCS heatup and cooldown, within the design assumptions and the stress limits for cyclic operation.

The Specification contains P/T limit curves for heatup, cooldown, system leakage and hydrostatic testing, and criticality, and also limits the maximum rate of change of reactor coolant temperature. The P/T limit curves are applicable up to ~~22 effective full power years.~~

the maximum fluence used for the limiting adjusted reference temperature for the 22 effective full power years as defined in Table 4.1 of Reference 7.

Each P/T limit curve defines an acceptable region for normal operation. The usual use of the curves is operational guidance during heatup or cooldown maneuvering, when pressure and temperature indications are monitored and compared to the applicable curve to determine that operation is within the allowable region.

The LCO establishes operating limits that provide a margin to brittle failure of the reactor vessel and piping of the reactor coolant pressure boundary (RCPB). The vessel is the component most subject to brittle failure. Therefore, the LCO limits apply mainly to the vessel.

10 CFR 50, Appendix G (Ref. 1), requires the establishment of P/T limits for material fracture toughness requirements of the RCPB materials. Reference 1 requires an adequate margin to brittle failure during normal operation, anticipated operational occurrences, and system hydrostatic tests. It mandates the use of the American Society of Mechanical Engineers (ASME) Code, Section III, Appendix G (Ref. 2).

The actual shift in the RT_{NDT} of the vessel material will be established periodically by evaluating the irradiated reactor vessel material data provided as part of the Boiling Water Reactor Vessel and Internals Project (BWRVIP) Integrated Surveillance Program (Refs. 11 and 12), in accordance with 10 CFR 50, Appendix H (Ref. 4). The operating P/T limit curves will be adjusted,

(continued)

B 3.7 PLANT SYSTEMS

B 3.7.5 Main Turbine Bypass System

18.5%

BASES

BACKGROUND

The Main Turbine Bypass System is designed to control steam pressure when reactor steam generation exceeds turbine requirements during unit startup, sudden load reduction, and cooldown. It allows excess steam flow from the reactor to the condenser without going through the turbine. The bypass capacity of the system is approximately ~~22%~~ of the Nuclear Steam Supply System rated steam flow. Sudden load reductions within the capacity of the steam bypass can be accommodated without reactor scram. The Main Turbine Bypass System consists of a five valve manifold connected to the main steam lines between the main steam isolation valves and the main turbine stop valves. Each of these valves is sequentially operated by hydraulic cylinders. The bypass valves are controlled by the pressure regulation function of the Turbine Electro Hydraulic Control System, as discussed in the USAR, Section 7.7.1.5 (Ref. 1). The bypass valves are normally closed, and the pressure regulator controls the turbine control valves, directing all steam flow to the turbine. If the speed governor or the load limiter restricts steam flow to the turbine, the pressure regulator controls the system pressure by opening the bypass valves. When the bypass valves open, the steam flows from the valve manifold, through connecting piping, to the pressure breakdown assemblies, where a series of orifices are used to further reduce the steam pressure before the steam enters the condenser.

APPLICABLE SAFETY ANALYSES

The Main Turbine Bypass System is assumed to function during the design basis feedwater controller failure, maximum demand event, described in the USAR, Section 15.1.2 (Ref. 2). Opening the bypass valves during the pressurization event mitigates the increase in reactor vessel pressure, which affects the MCPR during the event. An inoperable Main Turbine Bypass System may result in an MCPR penalty.

The Main Turbine Bypass System satisfies Criterion 3 of Reference 3.

(continued)

BASES (continued)

LCO

The Main Turbine Bypass System is required to be OPERABLE to limit peak pressure in the main steam lines and maintain reactor pressure within acceptable limits during events that cause rapid pressurization, such that the Safety Limit MCPR is not exceeded. With the Main Turbine Bypass System inoperable, modifications to the MCPR limits (LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)") may be applied to allow continued operation.

An OPERABLE Main Turbine Bypass System requires the bypass valves to open in response to increasing main steam line pressure. This response is within the assumptions of the applicable analysis (Ref. 2). The MCPR limit for the inoperable Main Turbine Bypass System is specified in the COLR.

APPLICABILITY

The Main Turbine Bypass System is required to be OPERABLE at ≥ ~~25%~~ 23% RTP to ensure that the fuel cladding integrity Safety Limit is not violated during the feedwater controller failure, maximum demand event. As discussed in the Bases for LCO 3.2.2, sufficient margin to this limit exists < ~~25%~~ 23% RTP. Therefore, these requirements are only necessary when operating at or above this power level.

ACTIONS

A.1

If the Main Turbine Bypass System is inoperable (one or more bypass valves inoperable), and the MCPR limits for an inoperable Main Turbine Bypass System, as specified in the COLR, are not applied, the assumptions of the design basis transient analysis may not be met. Under such circumstances, prompt action should be taken to restore the Main Turbine Bypass System to OPERABLE status or adjust the MCPR limits accordingly. The 2 hour Completion Time is reasonable, based on the time to complete the Required Action and the low probability of an event occurring during this period requiring the Main Turbine Bypass System.

(continued)

BASES

23% 23%

ACTIONS (continued)

B.1

If the Main Turbine Bypass System cannot be restored to OPERABLE status and the MCPR limits for an inoperable Main Turbine Bypass System are not applied, THERMAL POWER must be reduced to < 25% RTP. As discussed in the Applicability section, operation at < 25% RTP results in sufficient margin to the required limits, and the Main Turbine Bypass System is not required to protect fuel integrity during the feedwater controller failure, maximum demand event. The 4 hour Completion Time is reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.5.1

The Main Turbine Bypass System is required to actuate automatically to perform its design function. This SR demonstrates that, with the required system initiation signals, the valves will actuate to their required position. While this Surveillance can be performed with the reactor at power, operating experience has shown that these components usually pass the SR 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.

SR 3.7.5.2

This SR ensures that the TURBINE BYPASS SYSTEM RESPONSE TIME is in compliance with the assumptions of the appropriate safety analysis. The response time limits are specified in the Technical Requirements Manual (Ref. 4). While this Surveillance can be performed with the reactor at power, operating experience has shown that these components usually pass the SR 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.

(continued)