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NINE MILE POINT
NUCLEAR STATION

November 1, 2013

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

ATTENTION: Document Control Desk

SUBJECT: Nine Mile Point Nuclear Station, Unit 2
Renewed Facility Operating License No. NPF-69
Docket No. 50-410

License Amendment Request Pursuant to 10 CFR 50.90: Maximum Extended Load Line Limit Analysis Plus

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 to: 1) allow operation in the expanded Maximum Extended Load Line Limit Analysis Plus (MELLLA+) domain; 2) use of the Detect and Suppress Solution - Confirmation Density (DSS-CD) stability solution, 3) use of the TRACG04 analysis code; 4) increase the isotopic enrichment of boron-10 in the sodium pentaborate solution used to prepare the neutron absorber solution in the Standby Liquid Control System (SLS); and 5) increase the Safety Limit Minimum Critical Power Ratio (SLMCPR) for two recirculation loops in operation.

NMP2 is currently licensed to operate in the Maximum Extended Load Line Limit Analysis (MELLLA) domain and with the Option III thermal-hydraulic stability solution. This request to permit operation in the MELLLA+ domain expands the operating boundary without changing the maximum licensed core power, core flow, or the current vessel dome pressure. MELLLA+ requires changing the stability solution from Option III to DSS-CD and the use of the GE Hitachi Nuclear Energy (GEH) analysis code TRACG04. This request includes the supporting Technical Specifications (TSs) changes necessary to implement the expanded operating domain, change the stability solution, use the TRACG04 analysis code, increase the boron-10 isotopic enrichment in the sodium pentaborate solution utilized in the SLS, and increase the SLMCPR for two recirculation loops in operation.

Nine Mile Point Nuclear Station
P.O. Box 63, Lycoming, NY 13093

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 10 and 11.

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NRR

The Enclosure to this application and its associated Attachments provide the evaluation of the proposed changes to the NMP2 TSs. As indicated in the Enclosure, NMPNS concludes that the activities associated with the request involve no significant hazards consideration under the standards set forth in 10 CFR 50.92. The Enclosure contains the following Attachments:

1. Nine Mile Point Unit 2 Proposed Changes to Technical Specifications (Mark-ups)

2. Nine Mile Point Unit 2 Changes to Bases for Technical Specifications (Mark-ups)

3. List of Regulatory Commitments

4. MELLLA+ Risk Evaluation

5. Nine Mile Point Unit 2 Power/Flow Operating Map for Current Cycle

6. General Electric – Hitachi Affidavit Justifying Withholding Proprietary Information in NEDC-33576P

7. Global Nuclear Fuel Affidavit Justifying Withholding Proprietary Information in GNF-0000-0156-7490-R0-P

8. NEDC-33576NP, Safety Analysis Report for Nine Mile Point Unit 2 Maximum Extended Load Line Limit Analysis Plus (Non-proprietary)

9. Global Nuclear Fuel Report GNF-0000-0156-7490-R0-NP, "GNF Additional Information Regarding the Requested Change to the Technical Specification SLMCPR," dated August 26, 2013 (Non-proprietary)

10. NEDC-33576P, Safety Analysis Report for Nine Mile Point Unit 2 Maximum Extended Load Line Limit Analysis Plus (Proprietary)

11. Global Nuclear Fuel Report GNF-0000-0156-7490-R0-P, "GNF Additional Information Regarding the Requested Change to the Technical Specification SLMCPR," dated August 26, 2013 (Proprietary)

NMPNS requests approval of this application by November 1, 2014. Once approved, the amendment shall be implemented within 90 days of receipt.

Attachments 10 and 11 of the Enclosure contain information considered to be proprietary as defined by 10 CFR 2.390. GEH and Global Nuclear Fuel (GNF), as the owners of the proprietary information in Attachments 10 and 11, respectively, have executed the affidavits provided in Attachments 6 and 7 to the Enclosure detailing the reasons for withholding the proprietary information. On behalf of GEH and GNF, NMPNS hereby requests that proprietary information in Attachments 10 and 11 to the Enclosure be withheld from public disclosure in accordance with the provisions of 10 CFR 2.390.

Pursuant to 10 CFR 50.91(b)(1), NMPNS has provided a copy of this license amendment request, including the Enclosure with the exception of Attachments 10 and 11, to the appropriate state representative.

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

I declare under penalty of perjury that the foregoing is true and correct. Executed on November 1, 2013.

Sincerely,

Paul M. Swift
Manager, Engineering Services

CRC/STD

Enclosure: Evaluation of the Proposed Change

cc: Regional Administrator, Region I , NRC
Resident Inspector , NRC
Project Manager, NRC
A. L. Peterson NYSERDA (w/o Attachments 10 and 11 of Enclosure)

ENCLOSURE

EVALUATION OF THE PROPOSED CHANGES

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ATTACHMENTS

1.0 SUMMARY DESCRIPTION

This evaluation supports a request to amend Renewed Operating License (OL) NPF-69 for Nine Mile Point Unit 2 (NMP2). The proposed amendment includes supporting changes to the NMP2 Technical Specifications (TSs) necessary to: 1) implement the Maximum Extended Load Line Limit Analysis Plus (MELLLA+) expanded operating domain; 2) change the stability solution to Detect and Suppress Solution – Confirmation Density (DSS-CD); 3) use the TRACG04 analysis code; 4) increase the isotopic enrichment of boron-10 in the sodium pentaborate solution utilized in the Standby Liquid Control System (SLS); and 5) increase the Safety Limit Minimum Critical Power Ratio (SLMCPR) for two recirculation loops in operation.

The following is a list of the proposed changes to the NMP2 TSs:

- Revise Safety Limit (SL) 2.1.1.2 by increasing the SLMCPR for two recirculation loops in operation from ≥ 1.07 to ≥ 1.09
- Revise the acceptance criterion in TS 3.1.7, "Standby Liquid Control (SLC) System," Surveillance Requirement (SR) 3.1.7.7 by increasing the discharge pressure from ≥ 1,327 pounds per square inch gauge (psig) to ≥ 1,335 psig
- Revise the acceptance criterion in TS SR 3.1.7.10 by increasing the sodium pentaborate boron-10 enrichment requirement from ≥ 25 atom percent to ≥ 92 atom percent, and make a corresponding change in TS Figure 3.1.7-1, "Sodium Pentaborate Solution Volume/Concentration Requirements"
- Revise TS Figure 3.1.7-1 to account for the decrease in the minimum volume of the SLS tank from 4,558.6 gallons and 4,288 gallons at sodium pentaborate concentrations of 13.6% and 14.4%, respectively, to 1,600 gallons and 1,530 gallons at sodium pentaborate concentrations of 13.6% and 14.4%, respectively
- Change the Required Actions for Condition F of TS 3.3.1.1, "Reactor Protection System (RPS) Instrumentation"
- Change Condition G of TS 3.3.1.1
- Add new Conditions J and K to TS 3.3.1.1
- Correct an editorial error in Note 3 to TS SR 3.3.1.1.13 (i.e., "ORRM" is changed to "OPRM")
- Eliminate TS SR 3.3.1.1.16 and references to it in TS Table 3.3.1.1-1, "Reactor Protection System Instrumentation"
- Change the allowable value (AV) for TS Table 3.3.1.1-1, Function 2.b, Average Power Range Monitor (APRM) – Flow Biased Simulated Thermal Power (STP) – Upscale from "≤ 0.55W + 60.5% [Rated Thermal Power] RTP and ≤ 115.5% RTP" to "≤ 0.61W + 63.4% RTP and ≤ 115.5% RTP"
- Add a new note to TS Table 3.3.1.1-1, Function 2.b that requires the Flow Biased Simulated Thermal Power – Upscale scram setpoint to be reset to the values defined by the Core Operating Limits Report (COLR) to implement the Automated Backup Stability Protection (BSP) Scram Region in accordance with Required Action F.2.1 of TS 3.3.1.1

- Add a new note to TS Table 3.3.1.1-1, Function 2.e, Oscillation Power Range Monitor (OPRM) – Upscale to denote that following implementation of DSS-CD, DSS-CD is not required to be armed while in the DSS-CD Armed Region during the first reactor startup and during the first controlled shutdown that passes completely through the DSS-CD Armed Region. However, DSS-CD is considered operable and capable of automatically arming for operation at recirculation drive flow rates above the DSS-CD Armed Region
- Change the mode of applicability for TS Table 3.3.1.1-1, Function 2.e, OPRM-Upscale from Mode 1 to ≥ 18% RTP
- Change the allowable value for TS Table 3.3.1.1-1, Function 2.e from "As specified in the COLR" to "NA"
- Add a prohibition to TS Limiting Condition for Operation (LCO) 3.4.1, "Recirculation Loops Operating," that prohibits operation in the Maximum Extended Load Line Limit Analysis (MELLLA) domain or MELLLA+ expanded operating domain as defined in the COLR when in operation with a single recirculation loop
- Add Required Action B.2 to TS 3.4.1 to identify that intentional operation in the MELLLA domain or MELLLA+ domain as defined in the COLR is prohibited when a recirculation loop is declared "not in operation" due to a recirculation loop flow mismatch not within limits
- Revise TS 5.6.5.a.4 to replace "Reactor Protection System Instrumentation Setpoint for the OPRM – Upscale Function Allowable Value for Specification 3.3.1.1" with "The Manual Backup Stability Protection (BSP) Scram Region (Region I), the Manual BSP Controlled Entry Region (Region II), the modified APRM Simulated Thermal Power – High setpoints used in the OPRM (Function 2.e), Automated BSP Scram Region, and the BSP Boundary for Specification 3.3.1.1"
- Add TS 5.6.8, "OPRM Report," to define the contents of the report required by new Required Action F.2.2 of TS 3.3.1.1

Nuclear Regulatory Commission (NRC) approval of the requested operating domain expansion will allow NMP2 to implement operational changes that will increase operational flexibility for power maneuvering, compensate for fuel depletion, and maintain efficient power distribution in the reactor core without the need for more frequent rod pattern changes. MELLLA+ supports operation of NMP2 at Current Licensed Thermal Power (CLTP) of 3,988 Megawatts – Thermal (MW_{th}) with core flow as low as 85% of rated core flow. By operating in the MELLLA+ domain, a significantly lower number of control rod movements will be required than in the present operating domain. This represents a significant improvement in operating flexibility. It also provides safer operation, because reducing the number of control rod manipulations: (a) minimizes the likelihood of fuel failures and (b) reduces the likelihood of accidents initiated by reactor maneuvers required to achieve an operating condition where control rods can be withdrawn.

Attachments 8 and 10 provide the non-proprietary and proprietary versions of the MELLLA+ Safety Analysis Report (MELLLA+ SAR), respectively. The MELLLA+ SAR follows the guidelines contained in GE-Hitachi Nuclear Energy Americas (GEH) Licensing Topical Report (LTR) NEDC-33006P-A, Revision 3, "Maximum Extended Load Line Limit Analysis Plus" (MELLLA+ LTR) (Reference 1). The MELLLA+ SAR provides the technical bases for this request and contains an integrated summary of the results of the underlying safety analyses and evaluations performed specifically for the NMP2 expanded operating domain.

The MELLLA+ SAR also provides the analyses to change the NMP2 stability solution from Option III to DSS-CD and use the GEH analysis code TRACG04. DSS-CD is required by the MELLLA+ LTR Safety Evaluation Report. DSS-CD is being implemented using the guidelines

contained in GEH LTR NEDC-33075P-A, Revision 7, "General Electric Boiling Water Reactor Detect and Suppress Solution - Confirmation Density," (Reference 2). The use of TRACG04 is being implemented using the guidelines contained in GEH LTR "DSS-CD TRACG Application," NEDE-33147P-A, Revision 4, August 2013 (Reference 3).

The proposed change to the SLMCPR value for two recirculation loops in operation is based on an analysis performed by Global Nuclear Fuel (GNF) for NMP2 during Cycle 15 operations with MELLLA+ conditions. The GNF report, GNF-0000-0156-7490-R0, "GNF Additional Information Regarding the Requested Changes to the Technical Specification SLMCPR," dated August 26, 2013, supports changing the two recirculation loops in operation value of SLMCPR from ≥ 1.07 to ≥ 1.09, and maintaining the single recirculation loop in operation value of SLMCPR at ≥ 1.09. These values are based on NRC approved methods and procedures. Attachments 9 and 11 of this Enclosure provide non-proprietary and proprietary versions of the GNF report, respectively.

Attachments 10 and 11 of this Enclosure contain information considered to be proprietary as defined by 10 CFR 2.390. GEH and GNF, as the owners of the proprietary information in Attachments 10 and 11, respectively, have executed the affidavits provided in Attachments 6 and 7 to this Enclosure detailing the reasons for withholding the proprietary information.

Attachment 3 delineates the regulatory commitments associated with the proposed change.

2.0 DETAILED DESCRIPTION

2.1 Background

2.1.1 MELLLA+

Operation of Boiling Water Reactors (BWRs) requires that reactivity balance be maintained to accommodate fuel burn-up. BWR operators have two options to maintain this reactivity balance: (a) control rod movements or (b) core flow adjustments. Because of the strong void reactivity feedback and its distributed effect through the core, flow adjustments are the preferred reactivity control method. Operation at low-flow conditions at rated power level also increases the fuel capacity factor through spectral shift and the increased flow region compensates for reactivity reduction due to fuel depletion during the operating cycle.

At NMP2, an Extended Power Uprate (EPU) was implemented by extending the MELLLA operating domain up to the EPU power level (3,988 MW_{th}). The extension of the MELLLA line to EPU power levels reduces the available core flow window. In addition, the increased core pressure drop with EPU limits the recirculation flow capability. Consequently, EPU plants generally operate with a reduced core flow window and compensate for reactivity loss with control rod movement. Operation in the MELLLA+ expanded operating domain will provide a larger core flow window for NMP2.

In June 2009, the NRC approved the use of the MELLLA+ LTR (NEDO-33006P-A) (Reference 1) as a basis for MELLLA+ operating domain expansion license amendment requests, subject to limitations specified in the MELLLA+ LTR and in the associated NRC safety evaluation. The NMP2 request complies with the specified limitations and conditions as discussed in Appendix B of Attachments 8 and 10.

In January 2008, the NRC approved the use of the DSS-CD LTR (NEDC-33075P) as a basis for implementing DSS-CD as a stability solution to replace the Option III solution in license amendment requests, subject to limitations specified in the DSS-CD LTR and in the associated NRC safety evaluation. The NMP2 request complies with Revision 7 of NEDC-33075P (Reference 2), including the specified limitations and conditions as discussed in Appendix C of Attachments 8 and 10.

The TRACG code for use in DSS-CD applications (NEDE-33147P-A) was approved by NRC in November 2007. The NMP2 request complies with Revision 4 of NEDE-33147P-A (Reference 3).

In addition, the NRC approved the Applicability of GE Methods to Expanded Operating Domain Licensing Topical Report (NEDC-33173P-A) which imposes limitations and requirements for the use of GEH Methods in expanded operating domains including power uprates and MELLLA+ domains. The NMP2 request complies with Revision 4 of NEDC-33173P-A (Reference 4), including the specified limitations and conditions as discussed in Appendix A of Attachments 8 and 10.

Detailed evaluations of the reactor, engineered safety features, power conversion, emergency power, support systems, and design basis accidents were performed and are provided in Attachments 8 and 10. These evaluations demonstrate that NMP2 can safely operate in the MELLLA+ expanded operating domain with DSS-CD as the thermal hydraulic stability solution.

2.1.2 Standby Liquid Control System – Isotopic Enrichment of Boron-10

NMPNS proposes to increase the isotopic enrichment of boron-10 in the sodium pentaborate solution used to prepare the neutron absorber solution in the Standby Liquid Control System (SLS) to ≥ 92 atom-percent. The proposed boron-10 enrichment value allows the minimum net solution volume stored in the SLS storage tank to be decreased to 1,530 gallons at 14.4% sodium pentaborate concentration and 1,600 gallons at 13.6% sodium pentaborate concentration. In addition, NMPNS proposes to increase the acceptance criterion for the SLS pump discharge pressure from 1,327 psig to ≥ 1,335 psig.

2.1.3 Safety Limit Minimum Critical Power Ratio

NMPNS proposes to revise SL 2.1.1.2 by increasing the SLMCPR for two recirculation loops in operation from ≥ 1.07 to ≥ 1.09. The proposed change to the SLMCPR value for two recirculation loops in operation is based on an analysis performed by GNF for NMP2 during Cycle 15 operations with MELLLA+ conditions. The GNF report, GNF-0000-0156-7490-R0-P, "GNF Additional Information Regarding the Requested Changes to the Technical Specification SLMCPR," dated August 26, 2013, supports changing the two recirculation loops in operation value of SLMCPR from ≥ 1.07 to ≥ 1.09, and maintaining the single recirculation loop in operation value of SLMCPR at ≥ 1.09. These values are based on NRC approved methods and procedures. Attachments 9 and 11 of this Enclosure provide non-proprietary and proprietary versions of the GNF report, respectively.

2.2 Proposed Changes to the Nine Mile Point Unit 2 Technical Specifications

NMP2 TS changes are required to allow operation in the expanded MELLLA+ operating domain, use of DSS-CD, increase the isotopic enrichment of boron-10 in the sodium pentaborate solution

used to prepare the neutron absorber solution in the SLS, and increase the SLMCPR for two recirculation loops in operation. Attachment 1 of this Enclosure provides a mark-up of the NMP2 TS showing the proposed changes. Attachment 2 of this Enclosure provides a mark-up of the NMP2 TS Bases to show the corresponding changes to the TS Bases. Attachment 2 is provided for information only. A description of each TS change is provided below.

Safety Limit 2.1.1.2, Safety Limit Minimum Critical Power Ratio

SL 2.1.1.2 is revised to increase the SLMCPR for two recirculation loops in operation from ≥ 1.07 to ≥ 1.09.

TS 3.1.7, Standby Liquid Control (SLC) System

TS SR 3.1.7.7 is revised to increase the acceptance criterion for the SLS pump discharge pressure from ≥ 1,327 psig to ≥ 1,335 psig.

TS SR 3.1.7.10 is revised to increase the boron-10 enrichment requirement of sodium pentaborate from ≥ 25 atom percent to ≥ 92 atom percent. In addition TS Figure 3.1.7-1 is updated to reflect the increase in the boron-10 enrichment requirement.

TS Figure 3.1.7-1, "Sodium Pentaborate Solution Volume/Concentration Requirements," is revised to account for the change in the net volume in the SLS tank that arises from the enrichment increase. The minimum volume is changed from 4,558.6 gallons and 4,288 gallons at sodium pentaborate concentrations of 13.6% and 14.4%, respectively, to 1,600 gallons and 1,530 gallons at a sodium pentaborate concentration of 13.6% and 14.4%, respectively.

TS 3.3.1.1, Reactor Protection System (RPS) Instrumentation

Required Actions F.1 and F.2 of TS 3.3.1.1 and their associated Completion Times are replaced with the following new Required Actions and Completion Times.

	REQUIRED ACTION	COMPLETION TIME
F.1	Initiate Action to implement the Manual BSP Regions defined in the COLR.	Immediately
AND		
F.2.1	Implement the Automated BSP Scram Region using the modified APRM Simulated Thermal Power – High scram setpoints defined in the COLR.	12 hours
AND		
F.2.2	Initiate action in accordance with Specification 5.6.8.	90 days

Condition G is modified to no longer apply in the event a Required Action and associated Completion Time of Condition F is not met.

New Condition J (see below) is added to address the action to take in the event a Required Action and associated Completion Time of Condition F is not met.

New Condition K (see below) is added to address the action to take in the event a Required Action and associated Completion Time of Condition J is not met.

CONDITION	REQUIRED ACTION		COMPLETION TIME
J. Required Action and associated Completion Time of Condition F not met.	J.1	Initiate action to implement the Manual BSP regions defined in the COLR.	Immediately
	AND		
	J.2	Reduce operation to below the BSP Boundary defined in the COLR.	12 hours
	AND		
	J.3	----------NOTE---------- LCO 3.0.4 is not applicable ---------- Restore required channel to operable.	120 days
K. Required Action and associated Completion Time of Condition J not met.	K.1	Reduce THERMAL POWER to less than 18% RTP	4 hours

"ORRM" is changed to "OPRM" in Note 3 to TS SR 3.3.1.1.13.

TS SR 3.3.1.1.16 is eliminated, and references to it in TS Table 3.3.1.1-1 are eliminated.

TS Table 3.3.1.1-1, Function 2.b, Flow Biased Simulated Thermal Power - Upscale, contains both a flow-biased AV ($\leq$ 0.55W + 60.5% RTP) and a fixed AV at 115.5% RTP. The flow-biased AV will be changed to ($\leq$ 0.61W + 63.4% RTP).

A new note is added to TS Table 3.3.1.1-1, Function 2.b that requires the Flow Biased Simulated Thermal Power – Upscale scram setpoint to be reset to the values defined by the COLR to implement the Automated Backup Stability Protection (BSP) Scram Region in accordance with Required Action F.2.1 of TS 3.3.1.1.

A new note is added to TS Table 3.3.1.1-1, Function 2.e, OPRM – Upscale, to denote that following DSS-CD implementation, DSS-CD is not required to be armed while in the DSS-CD Armed Region during the first reactor startup and during the first controlled shutdown that passes completely through the DSS-CD Armed Region. However, DSS-CD is considered operable and

capable of automatically arming for operation at recirculation drive flow rates above the DSS-CD Armed Region.

The mode of applicability for TS Table 3.3.1.1-1, Function 2.e, OPRM-Upscale is changed from Mode 1 to ≥ 18% RTP.

In addition, the allowable value for Function 2.e is changed from "As specified in the COLR" to "NA."

TS 3.4.1, Recirculation Loops, Operating

LCO 3.4.1 is modified to include an additional provision that will prohibit intentional operation in the MELLLA domain or the MELLLA+ domain as defined in the COLR when only a single recirculation loop is in operation.

A new Required Action B.2 is added to prohibit intentional operation in the MELLLA domain or the MELLLA+ domain defined in the COLR in the event a recirculation loop is declared to be "not in operation" due to a recirculation loop flow mismatch. The Completion Time for this new Required Action is 2 hours.

TS 5.6.5, Core Operating Limits Report (COLR)

TS 5.6.5.a is modified by replacing the reference to "Reactor Protection System Instrumentation Setpoint for the OPRM – Upscale Function Allowable Value for Specification 3.3.1.1" with a reference to "The Manual Backup Stability Protection (BSP) Scram Region (Region I), the Manual BSP Controlled Entry Region (Region II), the modified APRM Simulated Thermal Power – High setpoints used in the OPRM (Function 2.e), Automated BSP Scram Region, and the BSP Boundary for Specification 3.3.1.1."

TS. 5.6.8, OPRM Report

The following new report requirement is added as TS 5.6.8, "OPRM Report:"

> "When a report is required by Required Action F.2.2 of TS 3.3.1.1, "RPS Instrumentation," a report shall be submitted within 90 days of entering CONDITION F. The report shall outline the preplanned means to provide backup stability protection, the cause of the inoperability, and the plans to schedule for restoring the required instrumentation channels to OPERABLE status."

The new TS section is numbered TS 5.6.8, because on November 21, 2012, NMP2 submitted a License Amendment Request (LAR) to create a new TS section that is numbered TS 5.6.7 for the Reactor Coolant System Pressure and Temperature Limits Report (Reference 5). NMPNS anticipates that LAR will be approved by the NRC and implemented at NMP2 prior to approval of the MELLLA+ LAR. The numbering of TS 5.6.8 is an administrative consideration. The MELLLA+ LAR is independent of the LAR submitted on November 21, 2012. NRC approval or rejection of Reference 5 would have no technical impact on the MELLLA+ LAR.

2.3 Modification Summary

The MELLLA+ core operating domain expansion does not require major plant hardware modifications. The core operating domain expansion involves changes to the core power/flow map and a small number of setpoints and alarms. Because there are no increases in the operating pressure, power, steam flow rate, and feedwater flow rate, there are no major modifications to other plant equipment.

The stability solution is being changed from Option III to the DSS-CD solution. The DSS-CD solution algorithm, licensing basis, and application procedures are generically described in the DSS-CD LTR (Reference 2), and are applicable to NMP2. The DSS-CD solution uses the same hardware as the current Option III solution. DSS-CD requires a revision to the existing stability solution software.

The boron-10 enrichment in the sodium pentaborate solution in the SLS is increased from ≥ 25 atom percent to ≥ 92 atom percent. The increase in the boron-10 enrichment in the sodium pentaborate solution for the SLS is sufficient to decrease the sodium pentaborate solution volume stored in the SLS storage tank. In addition, the SLS pump discharge pressure acceptance criterion is changed to ≥ 1,335 psig. Changes to instrumentation setpoints will be made to account for these changes. The increase in the SLMCPR for two recirculation loops in operation does not require any physical modifications to structures, systems, or components.

3.0 TECHNICAL EVALUATION

3.1 MELLLA+

Attachments 8 and 10 of this Enclosure provide non-proprietary and proprietary versions of the "Safety Analysis Report for Nine Mile Point Unit 2 Maximum Extended Load Line Limit Analysis Plus (MELLLA+ SAR)," NEDO-33576NP and NEDC-33576P, respectively. The MELLLA+ SAR summarizes the results of the significant safety evaluations performed that justify the expansion of the core flow operating domain for NMP2. The changes expand the operating domain in the region of operation with less than rated core flow, but do not increase the licensed power level or the maximum core flow. The expanded operating domain is identified as MELLLA+.

The scope of evaluations required to support the expansion of the core flow operating domain to the MELLLA+ boundary is contained in NEDC-33006P-A, "Maximum Extended Load Line Limit Analysis Plus," referred to as the MELLLA+ LTR (Reference 1). The MELLLA+ SAR provides a systematic disposition of the MELLLA+ LTR subjects applied to NMP2, including performance of plant-specific assessments and confirmation of the applicability of generic assessments to support a MELLLA+ core flow operating domain expansion. The MELLLA+ operating domain expansion is applied as an incremental expansion of the operating boundary without changing the maximum licensed power, maximum core flow, or the current plant vessel dome pressure. The MELLLA+ SAR supports operation of NMP2 at a licensed thermal power of 3,988 MWt with core flow as low as 85% of rated core flow.

The MELLLA+ core operating domain expansion does not require major plant systems modifications. NMP2 will implement the DSS-CD solution in accordance with the applicable LTRs (References 3 and 4), including the applicable limitations and conditions. Implementation of DSS-CD requires a revision to the existing stability solution software.

The core operating domain expansion involves changes to the operating power/core flow map and changes to a small number of instrument setpoints. Because there are no increases in the operating pressure, power, steam flow rate, and feedwater flow rate, there are no significant effects on the plant systems outside of the Nuclear Steam Supply System (NSSS). There is a potential increase in the steam moisture content at certain times while operating in the MELLLA+ operating domain. The effects of the potential increase in moisture content on plant systems have been evaluated and determined to be acceptable. The MELLLA+ operating domain expansion does not cause additional requirements to be imposed on any of the safety, balance-of-plant, electrical, or auxiliary systems. No changes to the power generation and electrical distribution systems are required as a result of the MELLLA+ operating domain expansion.

This report also addresses applicable limitations and conditions as described in the MELLLA+ LTR SER for the GEH LTR NEDC-33173P, "Applicability of GE Methods to Expanded Operating Domains" (Methods LTR SER) (Reference 4). A complete listing of the applicable LTR SER limitations and conditions and the sections of the MELLLA+ SAR which address them are presented in Appendices A, B, and C of the MELLLA+ SAR.

Only previously NRC-approved or industry-accepted methods were used for the analyses of accidents and transients. Therefore, because the safety analysis methods have been previously addressed, the details of the methods are not presented for review and approval in the MELLLA+ SAR. Also, event and analysis descriptions that are already provided in other licensing reports or the NMP2 Updated Safety Analysis Report (USAR) are not repeated within the MELLLA+ SAR.

Evaluations of the reactor core and fuel performance, reactor coolant and connected systems, engineered safety features, instrumentation and control, electrical power and auxiliary systems, power conversion systems, radwaste systems and radiation sources, reactor safety performance evaluations were performed. The MELLLA+ SAR summarizes the results of the evaluations that justify the MELLLA+ operating domain expansion to a minimum core flow rate of 85% of rated core flow at 100% RTP.

Section 11.3.1 of Attachments 8 and 10 provides a summary of the modifications that will be required to implement the MELLLA+ operating domain, DSS-CD, and the changes to the SLS.

Section 11.3.2 of Attachments 8 and 10 provides a summary of the MELLLA+ issues including a discussion of the MELLLA+ analysis basis, fuel thermal limits, makeup water sources, design basis accidents, challenges to fuel, challenges to the containment, design basis accident radiological consequences, anticipated operational occurrence analyses, combined effects, non-Loss of Coolant Accident (LOCA) radiological release accidents, equipment qualification, balance-of-plant, and environmental consequences.

An assessment of the risk increase, including core damage frequency (CDF) and large early release frequency (LERF) associated with operation in the MELLLA+ operating domain is provided in Attachment 4 of this Enclosure and Section 10.5 of Attachments 8 and 10 of this Enclosure. The estimated risk increase for at-power events due to MELLLA+ is a delta CDF of 1E-8 and delta LERF of 3E-9. This represents a very small risk change in RG 1.174 (Reference 6). Based on these results, the proposed MELLLA+ operating domain is acceptable on a risk basis.

3.2 DSS-CD

The long-term stability solution is being changed from the currently approved Option III solution to DSS-CD. The DSS-CD solution algorithm, licensing basis, and application procedures are generically described in NEDC-33075P-A (Reference 2) and NEDE-33147P-A (Reference 3), and are applicable to NMP2 including any limitations and conditions associated with their use and approval. Section 2.4 of the MELLLA+ SAR (Attachments 8 and 10) addresses the change to the DSS-CD stability solution. In addition, a complete listing of the required DSS-CD SER and limitations and conditions and the sections of the MELLLA+ SAR which address them is presented in Appendix C of the MELLLA+ SAR, respectively.

DSS-CD is designed to identify the power oscillation upon inception and initiate control rod insertion (scram) to terminate the oscillations prior to any significant amplitude growth. DSS-CD is based on the same hardware design as Option III. However, it introduces an enhanced detection algorithm that detects the inception of power oscillations and generates an earlier power suppression trip signal exclusively based on successive period confirmation recognition. The existing Option III algorithms are retained (with generic setpoints) to provide defense-in-depth protection for unanticipated reactor instability events.

3.3 Standby Liquid Control System – Boron-10 Enrichment

The SLS is described in Section 9.3.5 of the NMP2 USAR. The system provides a backup capability for shutting down the reactor. The SLS is needed only in the event that sufficient control rods cannot be inserted into the reactor core to accomplish shutdown and cooldown in the normal manner. To accomplish this function, the SLS injects a sodium pentaborate solution into the reactor. The SLS consists of a boron solution storage tank, two positive displacement pumps, two explosive valves (provided in parallel for redundancy), and associated piping and valves used to transfer borated water from the storage tank to the reactor pressure vessel (RPV). The borated water solution is discharged into the RPV through the high pressure core spray sparger.

The specified neutron absorber solution is sodium pentaborate. It is prepared by dissolving granularly-enriched sodium pentaborate in demineralized water (NMP2 USAR 9.3.5.2). The sodium pentaborate solution is discharged radially over the top of the core through the High Pressure Core Spray (HPCS) sparger. The boron absorbs thermal neutrons and thereby terminates the nuclear fission chain reaction in the uranium fuel. The sodium pentaborate also acts as a buffer to maintain the suppression pool pH at or above 7.0 to prevent the re-evolution of iodine, when mixed in the suppression pool following a LOCA accompanied by significant fuel damage (NMP2 USAR Section 9.3.5.1).

3.3.1 Reactor Boron Cold Shutdown Concentration Requirements

The reactor boron concentration requirements for achieving cold shutdown (780 parts per million (ppm) natural boron) is not increased for MELLLA+, because there is no change in fuel type and no change to the operating cycle. The total weight of boron-10 required for cold shutdown (including the 25% margin) does change for MELLLA+, because of a conservative increase in the assumed weight of the reactor coolant in the applicable analysis.

The shutdown margin is calculated for each plant reload and is documented in the Supplemental Reload Licensing Report (SRLR).

3.3.2 Change in Boron-10 Enrichment in Sodium Pentaborate

10 CFR 50.62(c)(4) requires:

> "Each boiling water reactor must have a standby liquid control system (SLCS) with the capability of injecting into the reactor pressure vessel a borated water solution at such a flow rate, level of boron concentration and boron-10 isotope enrichment, and accounting for reactor pressure vessel volume, that the resulting reactivity control is at least equivalent to that resulting from injection of 86 gallons per minute of 13 weight percent sodium pentaborate decahydrate solution at the natural boron-10 isotope abundance into a 251-inch inside diameter reactor pressure vessel for a given core design..."

The NRC-approved licensing topical report NEDE-31096-A (Reference 7) provides a method by which the boron equivalency requirement of 10 CFR 50.62(c)(4) can be demonstrated. Equation 1-1 of that document was used to demonstrate injection capacity equivalency as follows:

$$(Q/86) \times (M251/M) \times (C/13) \times (E/19.8) \geq 1$$

Where:

Q	=	expected SLS flow rate (gpm)
$M251$	=	mass of water in the reactor vessel and recirculation system at hot rated conditions (lbs) for a 251-inch diameter vessel reference plant
M	=	mass of water in the NMP2 reactor vessel and recirculation system at hot rated conditions (lbs)
C	=	sodium pentaborate solution concentration (wt%)
E	=	boron-10 isotope enrichment (atom percent)

NMP2 is equipped with a 251-inch diameter reactor vessel (NMP2 USAR Section 15G.5). Consequently the value of the M_{251} / M term in the above equation is 1. Table 1 provides the key assumptions utilized in the analyses of the changes to the SLS.

Substituting the current values defined in Table 1 in the above equation yields:

82.4/86 X 1 X 13.6/13 X 25/19.8 = 1.27 > 1

Substituting the new values defined in Table 1 into the above equation yields:

80/86 X 1 X 13.6/13 X 92/19.8 = 4.52 > 1 (Calculation for 13.6 wt%)

This demonstrates that the boron equivalent control capacity requirement of 10 CFR 50.62(c)(4) is met, when the changes to the SLS flow rate and the boron-10 isotope enrichment are included. In addition, the control margin increases. This is due to increasing the boron-10 enrichment term in the equation by a factor of 3.68 (i.e., 92/25 = 3.68).

Note: TS SR 3.1.7.7 requires each SLC System pump to have a flow rate of at least 41.2 gpm. Maintaining the TS SR 3.1.7.7 acceptance criteria for SLC pump flow rate at 41.2 gpm provides margin with respect to the required flow rate for ATWS mitigation. This issue has been addressed for current operation via the NMPNS Corrective Action System.

As defined in Table 1, the analyzed SLS injection flow rate is reduced to 80 gpm flow rate for two SLC System pumps in operation to account for dilution effects identified by GEH Safety Communication 10-13, Standby Liquid Control System Dilution Flow, with additional margin.

Table 1 – Assumptions regarding SLS Performance

Parameter	Units	Current Value	New Value
Reactor boron concentration for cold shutdown (natural boron)	ppm	780	780
Maximum allowable solution concentration	wt%	14.4	14.4
Minimum allowable solution concentration	wt%	13.6	13.6
Solution concentration assumed in ATWS analysis	wt%	13.6	13.6
Minimum boron-10 enrichment for ATWS analysis	Atom%	25	92
Design SLS pump flow rate	gpm	45	45
Minimum SLS pump flow rate as defined in TS 3.1.7	gpm	41.2	41.2
SLS pump flow rate (two pumps in operation)	gpm	82.4	80

3.3.3 Change in SLS Pump Discharge Pressure Acceptance Criterion

TS SR 3.1.7.7 is revised to increase the acceptance criterion for the SLS pump discharge pressure from ≥ 1,327 psig to ≥ 1,335 psig. This change is required due to the increase in the peak upper plenum pressure after SLS pump startup to 1,241 pounds per square inch – absolute (psia) as identified in Tables 9-4 and 9-7 of Attachments 8 and 10 of this Enclosure. Currently, the peak upper plenum pressure after SLS pump startup is 1,236 psia. Thus, the ATWS analysis for MELLLA+ establishes a pressure differential of five psi. The SLS pump discharge pressure acceptance criterion in TS SR 3.1.7.7 is increased by eight psig to address the increase in the upper plenum pressure and provide an additional three psi margin.

3.3.4 Anticipated Transient without SCRAM

Section 9.3.1 of the MELLLA+ SAR (Attachments 8 and 10 of this Enclosure) provides a summary of the plant-specific analyses of Anticipated Transients Without Scram (ATWS) to demonstrate that the ATWS acceptance criteria are met for operation in the MELLLA+ operating domain. NMP2 meets the ATWS mitigation requirements in 10 CFR 50.62 for an alternate rod insertion (ARI) system, SLS boron injection equivalent to 86 gpm, and automatic RPT logic. The plant-specific ATWS analyses take credit for the ATWS-RPT and SLS. However, ARI is not credited.

Section 9.3.1.1 of the MELLLA+ SAR (Attachments 8 and 10 of this Enclosure) provides a licensing basis ODYN ATWS analysis that demonstrates that the ATWS acceptance criteria

would be met in the event of a NMP2 response to an ATWS event initiated in the MELLLA+ operating domain.

In addition, a plant-specific ATWS analysis was performed at MELLLA+ conditions that assumed operation of a single SLS pump. The analysis and the results are discussed in Section 9.3.1.1.1 of the MELLLA+ SAR (Attachments 8 and 10 of this Enclosure). It concludes: "All ATWS acceptance criteria are met at MELLLA+ conditions with only a single SLS pump operating."

3.3.5 Suppression Pool Buffering

In support of the Alternate Source Term (AST) methodology, the SLS also provides suppression pool buffering following a LOCA accompanied by significant fuel damage, preventing re-evolution of iodine from the suppression pool by maintaining the pool pH above 7.0. Section 9.3.5.1 of the NMP2 USAR requires a sufficient concentration and quantity of sodium pentaborate to be available for injection into the reactor vessel to control pH in the suppression pool for 30 days following a DBA LOCA.

The reduction in the minimum required solution volume results in a reduction in the excess solution available for injection to maintain suppression pool pH $\geq$ 7.0 for 30 days post-LOCA. The minimum sodium pentaborate solution volume required for injection post-LOCA for adequate pH control is 1,065 gallons at the limiting concentration (i.e., a sodium pentaborate concentration of 13.6%). The minimum required tank volume at a concentration of 13.6 % is reduced from 4,558.6 gallons to 1,600 gallons. While this does reduce the amount of excess available solution, adequate margin is maintained to ensure that the SLS can perform its required AST support function.

The proposed boron-10 enrichment changes do not impact the capability to achieve and maintain a pH above 7.0 in the suppression pool following a LOCA, because the chemical properties and concentration of the sodium pentaborate solution injected into the suppression pool will remain the same. Given the reduced volume of solution that will be available, there will be a two hour reduction in the maximum time available to add boron to the suppression pool to maintain pH above 7.0 (nominal time based on low level alarm is within 22 hours versus the current time of within 24 hours). A review of the Emergency Operating Procedures confirmed that the sodium pentaborate solution would be injected within 30 minutes following the occurrence of LOCA. The maximum 22-hour time period provides a large margin to the minimum requirement for manual operator action to inject the sodium pentaborate solution of 30 minutes. In addition, the suppression pool pH is not expected to drop below 7 for several days.

Section 9.3.5.3 of the NMP2 USAR delineates that only one of the two SLS loops was assumed for suppression pool pH control operation. The proposed changes to the SLS do not affect the design redundancy of the SLS for suppression pool buffering.

3.3.6 Change in SLS Storage Tank Solution Volume

The proposed boron-10 enrichment value allows the minimum solution volume stored in the SLS storage tank to be decreased to 1,530 gallons at a sodium pentaborate concentration of 14.4% and 1,600 gallons at a sodium pentaborate concentration of 13.6%. The mark-up of NMP2 TS Figure 3.1.7-1 provided in Attachment 1 of this Enclosure delineates the proposed change in the net SLS storage tank solution volume.

The required minimum volumes for the 13.6 wt% and 14.4 wt% solution volumes were derived by determining the minimum solution volume and then increasing the volume to account for: 1) the dead volume not pumped in the reactor that remains in the SLS and HPCS piping; and 2) instrument accuracy.

The minimum net solution volume for injection meets all considerations for ATWS boron injection rates, AST suppression pool pH control, and assures that the reactor core boron concentration will be greater than 780 ppm natural boron equivalent.

3.3.7 SLS Pump Relief Valve Setpoint Margin

The SLS pump relief valve setpoint margin is the difference between the relief valve nominal setpoint and the maximum SLS pump discharge pressure. A margin of 78 psi provides sufficient margin against inadvertent relief valve lifting. The 78 psi is based on an allowance for the relief valve setpoint drift (typically 3% (3% of 1,600 psi = 48 psi)) and SLS pump pressure pulsations (30 psi).

For MELLLA+ operation during the limiting ATWS event, the relief valve setpoint margin is 205.7 psi. This margin is based on a SLS pump relief valve setpoint of 1552 psig (1600 psig – 3% tolerance (i.e., 48 psig)) and subtracting a SLS pump discharge pressure of 1346.3 psig (i.e., 1552 psig – 1346.3 psig = 205.7 psi). The margin reduces to 175.7 psi if 30 psi for SLS pump pressure pulsations is taken into consideration (i.e., 205.7 psi – 30 psi = 175.7 psi).

3.3.8 Net Positive Suction Head Available ($NPSH_A$) for SLS Pumps

The proposed changes include a reduction in the minimum volume for the SLS storage tank. This results in a reduction in the static head available to provide Net Positive Suction Head (NPSH) for the SLS pumps. The calculation that determines the SLS pump $NPSH_A$ did not take any credit for the static head above the SLS storage tank zero level. The minimum tank level corresponding to the minimum net volume permitted by the proposed change to Figure 3.1.7-1 is greater than three feet above tank zero.

3.4 Safety Limit Minimum Critical Power Ratio

Cycle specific transient analyses are performed to determine the required SLMCPR and the change in Critical Power Ratio (CPR) [ΔCPR] for specific transients. To ensure that adequate margin is maintained, a design requirement based on a statistical analysis was selected, in that moderate frequency transients caused by a single operator error or equipment malfunction shall be limited such that, considering uncertainties in manufacturing and monitoring the core operating state, at least 99.9% of the fuel rods would be expected to avoid boiling transition. The lowest allowable transient MCPR limit which meets the design requirement is termed the fuel cladding integrity SLMCPR.

NUREG-0800, Standard Review Plan, Section 4.4, "Thermal and Hydraulic Design," Acceptance Criterion No. 1.B, states, in part, that the limiting (minimum) value of CPR is to be established such that at least 99.9% of the fuel rods in the core would not be expected to experience departure from nucleate boiling during normal operation or anticipated operational occurrences.

A cycle specific Operating Limit MCPR (OLMCPR) is established to provide adequate assurance that the fuel cladding integrity SLMCPR is not exceeded for any anticipated operational transients. The OLMCPR is obtained by adding the maximum value of ΔCPR for the most limiting transient postulated to occur at the plant to the fuel cladding integrity SLMCPR.

3.4.1 Analytical Methods, Standards, Data and Results

NMPNS proposes to revise SL 2.1.1.2 by increasing the SLMCPR for two recirculation loops in operation from ≥ 1.07 to ≥ 1.09. The proposed change to the SLMCPR value for two recirculation loops in operation is based on an analysis performed by GNF for NMP2 during Cycle 15 operations with MELLLA+ conditions. The GNF report, GNF-0000-0156-7490-R0-P, "GNF Additional Information Regarding the Requested Changes to the Technical Specification SLMCPR," dated August 26, 2013, supports changing the two recirculation loops in operation value of SLMCPR from ≥ 1.07 to ≥ 1.09, and maintaining the single recirculation loop in operation value of SLMCPR at ≥ 1.09. These values are based on NRC approved methods and procedures. Attachments 9 and 11 of this Enclosure provide non-proprietary and proprietary versions of the GNF report, respectively.

GNF performed the SLMCPR calculation in accordance with Revision 19 of NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel," (Reference 8) using the following NRC-approved methodologies and uncertainties:

- NEDC-32601P-A, "Methodology and Uncertainties for Safety Limit MCPR Evaluations," (August 1999) (Reference 9).
- NEDC-32694P-A, "Power Distribution Uncertainties for Safety Limit MCPR Evaluations" (August 1999) (Reference 10).
- NEDC-32505P-A, "R-Factor Calculation Method for GE11, GE12 and GE13 Fuel," (Revision 1, July 1999) (Reference 11).

Section 2.9 of Attachments 9 and 11 of this Enclosure require NMPNS to "provide the current and previous cycle power/flow map in a separate attachment." Figure 1-1 of Attachments 8 and 10 of this Enclosure provide the power/flow operating map for MELLLA+. This will be the power/flow map for NMP2 operations in Cycle 15 following NRC approval of this License Amendment Request. Attachment 5 of this Enclosure provides the NMP2 power/flow operating map for the current operating cycle.

3.4.2 Major Contributors to SLMCPR Change

In general, the calculated safety limit is dominated by two key parameters: (1) flatness of the core bundle-by-bundle MCPR distribution, and (2) flatness of the bundle pin-by-pin power/R-Factor distribution. Greater flatness in either parameter yields more rods susceptible to boiling transition and thus a higher calculated SLMCPR. The MCPR Importance Parameter (MIP) measures the

core bundle-by-bundle MCPR distribution and the R-Factor Importance Parameter (RIP) measures the bundle pin-by-pin power/R-Factor distribution. The impact of the fuel loading pattern on the calculated two recirculation loops in operation SLMCPR has been correlated to the parameter MIPRIP, which combines the MIP and RIP values.

Another factor besides core MCPR distribution or bundle R-factor distribution that significantly impacts the SLMCPR is the expansion of the analysis domain that comes with the initial application of MELLLA+. The rated power / minimum core flow point is analyzed at a lower core flow (than without MELLLA+) using increased uncertainties that tend to increase the SLMCPR. Also, a new point at off-rated power / off-rated flow was analyzed using the increased uncertainties.

Table 3 of the GNF analysis (Attachments 9 and 11 of this Enclosure) presents the MIP and RIP parameters for the previous cycle and the current cycle along with the two recirculation loops in operation SLMCPR estimates using MIPRIP correlations. In addition, Table 3 of the GNF analysis provided in Attachments 9 and 11 presents estimated impacts on the two recirculation loops in operation SLMCPR due to methodology deviations, penalties, and/or uncertainty deviations from approved values. Based on the MIPRIP correlation and any impacts due to deviations from approved values, a final estimated two loops in operation SLMCPR is determined. Section 2.2 of the GNF analysis (Attachments 9 and 11 of this Enclosure) provides a detailed discussion of the items in Table 3 of the GNF analysis (Attachments 9 and 11 of this Enclosure) that result in the increase in the estimated SLMCPR.

3.4.3 Considerations Addressed in the GNF Analysis Regarding R-Factor, Core Flow Rate and Random Effective Tip Reading, and Fuel Axial Power Shape Penalty

Section 2.2.1 of the GNF analysis provides a discussion that justifies an increase in the R-Factor uncertainty value. GNF states that it generically increased the GEXL R-Factor uncertainty to account for an increase in channel bow due to the emerging unforeseen phenomena called control blade shadow corrosion-induced channel bow, which is not accounted for in the channel bow uncertainty component of the approved R-Factor uncertainty. NMP2 has experienced control blade shadow corrosion-induced channel bow. Accounting for the control blade shadow corrosion-induced channel bow, the NMP2 Cycle 15 analysis shows an expected channel bow uncertainty which is bounded by the increased GEXL R-Factor uncertainty. Thus, the use of the increased GEXL RFactor uncertainty value adequately accounts for the expected control blade shadow corrosion-induced channel bow for NMP2 Cycle 15.

Section 2.2.2 of the GNF analysis provides a discussion that identifies that the uncertainty values for the core flow rate and the random effective tip reading in the two recirculation loops in operation calculation were conservatively adjusted by using the single recirculation loop in operation uncertainty values. The GNF analysis states the treatment of the core flow and random effective TIP reading uncertainties is based on the assumption that the signal to noise ratio deteriorates as core flow is reduced.

Section 2.4 of the GNF analysis provides a discussion regarding higher uncertainties and non-conservative bases in the GEXL correlations for the various types of axial power shapes. GNF determined that no power shape penalties were required to be applied to the calculated NMP2 Cycle 15 SLMCPR values.

3.4.4 Conclusion

The proposed change to revise SL 2.1.1.2 by increasing the SLMCPR for two recirculation loops in operation from ≥ 1.07 to ≥ 1.09 is acceptable, and continues to maintain the same level of safety as the current licensing basis.

3.5 NMP2 TS Changes

Table 2 defines the affected NMP2 TS, describes the change, and defines the supporting Attachment to this Enclosure that supports the TS Change.

Table 2 – Changes to NMP2 Technical Specifications

NMP2 TS	Description of the Change	Supporting Attachment
SL 2.1.1.2	Increase the SLMCPR for two recirculation loops in operation from ≥ 1.07 to ≥ 1.09	Attachments 9 and 11
TS 3.1.7 – SR 3.1.7.7	Increase the SLS pump discharge pressure from ≥ 1,327 psig to ≥ 1,335 psig	Section 6.5.3 of Attachments 8 and 10
TS 3.1.7 – SR 3.1.7.7 and TS Figure 3.1.7-1	Increasing the sodium pentaborate boron-10 enrichment requirement from ≥ 25 atom percent to ≥ 92 atom percent	Section 6.5.1 of Attachments 8 and 10
TS Figure 3.1.7-1	Reducing the minimum net volume to 1,600 gallons and 1,530 gallons at sodium pentaborate concentrations of 13.6% and 14.4%, respectively	Section 6.5.1 of Attachments 8 and 10
TS Figure 3.1.7-1	Increasing the sodium pentaborate boron-10 enrichment requirement from ≥ 25 atom percent to ≥ 92 atom percent	Section 6.5.1 of Attachments 8 and 10
TS 3.3.1.1	The Required Actions for Condition F are modified to: 1) Initiate Action to implement the Manual BSP Regions defined in the COLR; 2) Implement the Automated BSP Scram Region using the modified APRM Simulated Thermal Power – High scram setpoints defined in the COLR; and 3) Initiate action in accordance with Specification 5.6.8	Complies with DSS-CD LTR Section 2.4 of Attachments 8 and 10
TS 3.3.1.1	Condition G is modified to no longer apply in the event a Required Action and associated Completion Time of Condition F is not met.	Complies with DSS-CD LTR Section 2.4 of Attachments 8 and 10
TS 3.3.1.1	New Condition J is added to address the action to take in the event a Required Action and associated Completion Time of Condition F is not met.	Complies with DSS-CD LTR Section 2.4 of Attachments 8 and 10

Table 2 – Changes to NMP2 Technical Specifications

NMP2 TS	Description of the Change	Supporting Attachment
TS 3.3.1.1	New Condition K is added to address the action to take in the event a Required Action and associated Completion Time of Condition J is not met.	Complies with DSS-CD LTR Section 2.4 of Attachments 8 and 10
TS SR 3.3.1.1.13	Correct an editorial error in Note 3 (i.e., ORRM is changed to OPRM)	Editorial correction
TS SR 3.3.1.1.16 and TS Table 3.3.1.1-1	Eliminate TS SR 3.3.1.1.16 and references to it in TS Table 3.3.1.1-1	Complies with DSS-CD LTR Section 2.4 of Attachments 8 and 10
TS Table 3.3.1.1-1, Function 2.b	Change the AV for APRM – Flow Biased STP – Upscale from "≤ 0.55W + 60.5% RTP and ≤ 115.5% RTP" to "≤ 0.61W + 63.4% RTP and ≤ 115.5% RTP"	Section 5.3.1 of Attachments 8 and 10
TS Table 3.3.1.1-1, Function 2.b	Add a new note that requires the Flow Biased Simulated Thermal Power – Upscale scram setpoint to be reset to the values defined by the COLR to implement the Automated BSP Scram Region in accordance with Required Action F.2.1 of TS 3.3.1.1	Complies with DSS-CD LTR Section 2.4 of Attachments 8 and 10
TS Table 3.3.1.1-1, Function 2.e	Add a new note for Function 2.e, OPRM – Upscale, to denote that following implementation of DSS-CD, DSS-CD is not required to be armed while in the DSS-CD Armed Region during the first reactor startup and during the first controlled shutdown that passes completely through the DSS-CD Armed Region. However, DSS-CD is considered operable and capable of automatically arming for operation at recirculation drive flow rates above the DSS-CD Armed Region	Complies with DSS-CD LTR Section 2.4 of Attachments 8 and 10
TS Table 3.3.1.1-1, Function 2.e	Change the mode of applicability for TS Table 3.3.1.1-1, Function 2.e, OPRM-Upscale from Mode 1 to ≥ 18% RTP.	Complies with DSS-CD LTR Section 2.4 of Attachments 8 and 10
TS Table 3.3.1.1-1, Function 2.e	Change the allowable value from "As specified in the COLR" to "NA"	Complies with DSS-CD LTR Section 2.4 of Attachments 8 and 10

Table 2 – Changes to NMP2 Technical Specifications

NMP2 TS	Description of the Change	Supporting Attachment
TS LCO 3.4.1	Add a new requirement that prohibits operation in the MELLLA domain or MELLLA+ expanded operating domain as defined in the COLR when in operation with a single recirculation loop	Complies with DSS-CD LTR Sections 1.2.4 and 3.6.3 2.4 of Attachments 8 and 10 address that MELLLA+ is not analyzed for single loop operation In addition, NMP2 does not currently permit single loop operation while in the MELLLA domain, because it is not analyzed.
TS 3.4.1, Condition B	Add Required Action B.2 to identify that intentional operation in the MELLLA domain or MELLLA+ domain as defined in the COLR is prohibited when a recirculation loop is declared "not in operation" due to a recirculation loop flow mismatch not within limits	Complies with DSS-CD LTR Sections 1.2.4 and 3.6.3 2.4 of Attachments 8 and 10 address that MELLLA+ is not analyzed for single loop operation In addition, NMP2 does not currently permit single loop operation while in the MELLLA domain, because it is not analyzed.

Table 2 – Changes to NMP2 Technical Specifications

NMP2 TS	Description of the Change	Supporting Attachment
TS 5.6.5	Replace the reference to "Reactor Protection System Instrumentation Setpoint for the OPRM – Upscale Function Allowable Value for Specification 3.3.1.1" with a reference to "The Manual Backup Stability Protection (BSP) Scram Region (Region I), the Manual BSP Controlled Entry Region (Region II), the modified APRM Simulated Thermal Power – High setpoints used in the OPRM (Function 2.e), Automated BSP Scram Region, and the BSP Boundary for Specification 3.3.1.1."	Complies with DSS-CD LTR Section 2.4 of Attachments 8 and 10
TS 5.6.8	Add a new TS section (i.e., TS 5.6.8) to define the contents of the report required by new Required Action F.2.2 of TS 3.3.1.1	Complies with DSS-CD LTR Section 2.4 of Attachments 8 and 10

3.6 TSTF-493

There are no effects on the current TS or their licensing bases relative to TSTF-493. Two TS Reactor Protection System (RPS) functions are changing in this amendment: (1) the OPRM - Upscale function; and (2) the APRM – Flow Biased Simulated Thermal Power (STP) – Upscale function. The OPRM setpoints are unique to a particular core design for a particular fuel cycle. The OPRM function setpoints do not have specific TS allowable values (AVs). The APRM STP - High AVs are specified in TS Table 3.3.1.1-1.

MELLLA+ changes the OPRM setpoints in that they are now derived from DSS-CD algorithms versus Option III algorithms; however, their protective function remains the same. The revised Bases for TS 3.3.1.1 provided in Attachment 2 of this Enclosure states: "The OPRM Upscale function settings are not traditional instrumentation setpoints determined under an instrument setpoint methodology. There is no Allowable Value for this Function, and the OPRM Upscale Function is not [Limiting Safety System Setting (LSSS) Safety Limit (SL)]-related and [the DSS-CD Licensing Topical Report, NEDC-33075P-A] confirms that the OPRM Upscale Function settings based on DSS-CD also do not have traditional instrumentation setpoints determined under an instrument setpoint methodology."

MELLLA+ also changes the APRM – Flow Biased Simulated Thermal Power - Upscale AV for two loop operations in the MELLLA+ domain and the APRM – Flow Biased Simulated Thermal Power - Upscale function is used for the Automated Backup Stability Protection (ABSP) if the OPRM becomes inoperable. The APRM STP-High AV and setpoint do have setpoint methodology applied as described in TSTF-493. In addition, the TSTF-493 footnotes were previously added to this function in Amendment 140 to the NMP2 Renewed Operating License NPF-69 issued on December 22, 2011 (Reference 12).

3.7 Topics Discussed During NRC Pre-Meetings

On February 27, 2013, representatives from NMPNS met with the NRC to discuss the MELLLA+ LAR. During this meeting, the NRC sought clarification regarding several topics. Table 3 summarizes those topics, and provides a cross reference to the location in Attachments 8 and 10 of this Enclosure that addresses the topic. The NRC issued a summary of this meeting on March 13, 2013 (Reference 13).

Table 3 – Topics Discussed During NRC Pre-Meeting on February 27, 2013

Topic as Summarized in NRC Meeting Summary Issued on March 13, 2013 (Reference 13)	MELLLA+ SAR Reference (Attachments 8 and 10 of this Enclosure)
Automated Backup Stability Protection The NMP2 submittal is based on Revision 6 of NEDC-33075P-A. The NMP2 is planning to take exception to Rev 6 relative to the Automatic Backup Stability Protection (ABSP) set points by using a simplified method that is consistent with the ABSP set point methodology described in Revision 7 of NEDC-33075P. Since the NRC staff has not approved Revision 7 of the Licensing Topical Report (LTR) NEDE-33075P, Re: Detect and Suppress Solution-Confirmation Density (DSS-CD) for Automatic Backup Stability Protection (ABSP), the License Amendment Request (LAR) should not refer to revision 7 of NEDE-33075P, but provide the justifications, consistent with revision 7, for any exceptions taken in the LAR.	The NMPNS submittal is based on Revision 7 of NEDC-33075P-A. Since the February 27, 2013 meeting, the NRC approved Revision 7 of NEDC-33075P-A Justification provided in Section 2.4.3
Emergency Core Cooling System NPSH The NMP2 does not take credit for Containment Accident Pressure (CAP) to assure adequate net positive suction head (NPSH). In response to NRC staff, the licensee stated that a re-analysis of CAP is not required as a result of MELLLA+. Based on feedback from the NRC staff, the NMP2 MELLLA+ submittal will reference the NMP2 Extended Power Uprate (EPU) submittal Requests for Additional Information (RAI's) related to CAP and describe that the NPSH margins in the NMP2 EPU responses remain bounding for MELLLA+.	Information provided in Section 4.2.6
DSS-CD Implementation Implementation of DSS-CD Stability Solution in Place of Option III. The NMP2 MELLLA+ submittal will address the implementation strategy for DSS-CD, including the need for monitoring the timing for arming the protection associated with DSS-CD and the Oscillation Power Range Monitor (OPRM) data analysis already completed.	Information provided in Section 2.4.1

Table 3 – Topics Discussed During NRC Pre-Meeting on February 27, 2013

Topic as Summarized in NRC Meeting Summary Issued on March 13, 2013 (Reference 13)	MELLLA+ SAR Reference (Attachments 8 and 10 of this Enclosure)
TRACG ATWS with Core Instability (ATWSI) The NMP2 submittal will include anticipated transients without SCRAM with instability (ATWSI) sensitivity analysis results using a modified T-min correlation similar to what General Electric Hitachi Nuclear Energy (GEH) provided in response to another licensee's RAI. Additional information on the model was requested if and when it becomes available. However, GEH noted that there is no additional testing at this time.	Information provided in Section 9.3.3
Operator Training Provide the implementation plan outlining the simulator upgrade and operator training plan to support implementation of the LAR.	Information provided in Section 10.6 NMPNS has requested that the NRC approve this LAR by October 2014. To support this schedule, NMPNS plans to upgrade the simulator by the second quarter of 2014 to support operator training in the second and third quarters of 2014.
Reference Core versus Actual Cycle Specific Core Cycle Specific Core Design and Associated Safety Analyses, and Reload Analysis using PRIME Code. The NMP2 submittal will describe the potential differences in the analytical inputs and results between the reference core and the actual reload analysis that will be submitted as a supplement to the MELLLA+ submittal.	See Notes 1 through 3 Information provided in Sections 2.1, 2.2, and 2.6.3 and Footnote 4 of Appendix A
GESTR-M versus PRIME Subsequent to the meeting the NRC staff noted that the licensee's presentation stated that the licensee's LAR submission is going to include the analyses based in GESTR-M Code and it is planning to supplement its LAR with the Analyses based on PRIME Code, The LAR submission based on GESTR-M Code would not be "acceptable," This staff concern has been communicated to the licensee on March 12, 2013. In an email dated March 12, 2013, the NRC staff noted that a LAR submission based on GESTR-M Code would not be "acceptable". A follow-up meeting with the NRC was conducted on March 29, 2013.	Following the NRC discussions, the MELLLA+ SAR was revised to utilize PRIME Thermal-Mechanical (T-M) methodology. In addition, PRIME fuel parameters have been used in the analyses requiring fuel performance parameters. Information provided in Table 1-1, Sections 2.6.3 and 4.3

Table 3 – Topics Discussed During NRC Pre-Meeting on February 27, 2013

Topic as Summarized in NRC Meeting Summary Issued on March 13, 2013 (Reference 13)	MELLLA+ SAR Reference (Attachments 8 and 10 of this Enclosure)
Notes: 1. The fuel and cycle-dependent analyses, including the plant-specific thermal limits assessment, will be submitted for NRC staff confirmation by supplementing the initial MELLLA+ Safety Analysis Report (SAR) in accordance with Limitation and Condition 12.4 of the MELLLA+ Licensing Topical Report (LTR) Safety Evaluation Report (SER). Specifically, CENG will provide the cycle specific Supplemental Reload Licensing Report (SRLR) and Fuel Bundle Information Report (FBIR), which includes the supplemental information to satisfy MELLLA+ LTR SER Limitation and Condition 12.4. CENG will submit this information by February 28, 2014. 2. Nine Mile Point Nuclear Station, LLC (NMPNS) will provide a cycle-specific core design loading map along with a summary of differences between the reference design described in the M+SAR and the cycle-specific core design. This summary will include differences in the energy requirements, average enrichment, and analytical inputs, a cycle-specific thermal limits assessment, and the actual reload analysis results. Additionally, the Supplemental Reload Licensing Report, which includes the cycle specific core map, will be provided. Submittal of the cycle-specific design will satisfy the NRC request made at the MELLLA+ LAR pre-meeting on March 13, 2013. 3. The NMP2 Cycle 15 specific reload analysis will utilize TRACG rather than ODYN for AOO. Section 9.1.1 of the MELLLA+ SAR (Attachments 8 and 10 of this Enclosure) states: "In the event that the cycle-specific reload analysis is based on TRACG rather ODYN for AOO, no 0.01 added to the OLMCPR is required."	

4.0 REGULATORY EVALUATION

4.1 Evaluation of NMP2 License Amendment Requests to Establish that They Are Not Linked

4.1.1 Guidance from NRR Office Instruction LIC-109, "Acceptance Review Procedures"

Revision 1 of Office of Nuclear Reactor Regulation (NRR) Office Instruction LIC-109, "Acceptance Review Procedures," (Reference 14) provides the NRR staff (and other NRC staff supporting NRR licensing activities) a basic framework for performing an acceptance review upon receipt of a Requested Licensing Action (RLA) from a licensee. It defines that the NRC should not accept for NRC review and approval an RLA that is linked to another RLA.

Section 1.3.2 of LIC-109 states linked RLAs "are RLAs, where approval of one RLA is contingent upon the approval of (an) other RLA(s) currently under review. This definition evaluates the independence of an RLA with respect to all other RLAs currently under review."

Section 3.1.1 of LIC-109 states: "Linked RLAs: Determine whether the approval of the RLA is contingent upon the approval of other RLAs currently under review. It is important to note that multiple RLAs can affect the same systems or Technical Specifications (TSs) without being

linked. As such, it may be possible to issue them in any order and without regard to the results of the review of the others. An RLA should not be accepted for NRC review and approval until all prerequisite RLAs have been reviewed and approved by the NRC."

In addition, Example 3 of LIC-109 provides the following example of linked RLAs.

> "While the NRC staff is reviewing a licensee's request to change the accident analyses for a loss-of-coolant accident (LOCA), the licensee submits an application for an extended power uprate (EPU). The analysis and supporting justification for the EPU are based, in part, on the proposed LOCA analysis currently under review."

LIC-109 states that this example is not acceptable, "because the EPU should not begin until all prerequisite reviews have been completed (Linked RLAs). Additionally, the regulatory basis cited in the EPU application (i.e., the currently unapproved LOCA analysis) is not the current licensing basis for the plant (Regulatory Basis)." It stated that it may be acceptable for review if "review and approval of the EPU was not contingent upon the outcome of the NRC Staff's review of the LOCA analysis."

4.1.2 Evaluation of NMP2 RCS Pressure – Temperature and MELLLA+ License Amendment Requests

On November 21, 2012, NMP2 submitted a License Amendment Request (LAR) to create a new TS section that is numbered TS 5.6.7 for the Reactor Coolant System Pressure and Temperature Limits Report (Reference 5). In the MELLLA+ LAR, a new TS is added that is number TS 5.6.8. The numbering of TS 5.6.8 is an administrative consideration. The MELLLA+ LAR is independent of the LAR submitted on November 21, 2012. NRC approval or rejection of Reference 5 would have no technical impact on the MELLLA+ LAR.

4.1.3 Evaluation of NMP2 SLS and MELLLA+ License Amendment Requests

On July 5, 2013, NMPNS submitted a request to amend the NMP2 Renewed Operating License (OL) NPF-69 to increase the isotopic enrichment of boron-10 in the sodium pentaborate solution used to prepare the neutron absorber solution in the SLS (Reference 15). This request includes the supporting changes to the NMP2 Technical Specification (TS) 3.1.7, "Standby Liquid Control (SLC) System," to increase the boron-10 isotopic enrichment in the sodium pentaborate solution utilized in the SLC System and to decrease the SLC System tank volume.

The SLS LAR and the MELLLA+ LAR both affect NMP2 TS 3.1.7, including the same changes to SR 3.1.7.10 and Figure 3.1.7-1 to increase the isotopic enrichment of boron-10 in the sodium pentaborate solution and the associated change in the SLS Tank Minimum volume. Section 3.1.1 of LIC-109 establishes that multiple RLAs can affect the same systems or TSs without being linked.

As stated in Section 1.3.2 of LIC-109, linked RLAs "are RLAs, where approval of one RLA is contingent upon the approval of (an) other RLA(s) currently under review."

The SLS LAR proposes changes to NMP2 TS 3.1.7 that would increase the isotopic enrichment of boron-10 in the sodium pentaborate solution and reduce the SLS Tank Minimum volume requirements, so that they could be implemented during the spring refueling outage in 2014 for NMP2. These changes will be justified utilizing the current licensing basis, and are not

dependent on the analysis that will be submitted in the MELLLA+ LAR. The NRC can review and approve the SLS LAR without reference to the MELLLA+ LAR.

The MELLLA+ LAR proposes changes to the NMP2 TS, including changes to NMP2 TS 3.1.7 that would increase the SLS pump discharge pressure acceptance criterion, increase the isotopic enrichment of boron-10 in the sodium pentaborate solution, and reduce the SLS Tank Minimum volume requirements. These changes will be justified utilizing analyses that are summarized in the MELLLA+ LAR, including a MELLLA+ specific boron equivalency analysis and ATWS analysis. Given that the justification for these changes will be provided in the MELLLA+ LAR, the NRC can review and approve the MELLLA+ LAR without reference to the SLS LAR.

If the SLS LAR is approved by the NRC and implemented prior to the NRC approval of the MELLLA+ LAR, the only impact to the MELLLA+ LAR would be to remove the proposed changes to SR 3.1.7.10 and Figure 3.1.7-1. The analyses provided in the MELLLA+ LAR justify that those values are appropriate for operation in the MELLLA+ domain. Thus, those analyses remain valid, and NRC review is required to justify operation in the MELLLA+ domain with those SLS parameters.

If the SLS LAR is not approved by the NRC, this action would have no impact on the NRC review and approval of the MELLLA+ LAR.

If the MELLLA+ LAR is approved by the NRC and implemented prior to the NRC approval of the SLS LAR, then the SLS LAR would be retracted by NMPNS because the SLS LAR does not address operation in the MELLLA+ operating domain and the applicable changes would be superfluous.

If the MELLLA+ LAR is not approved by the NRC, this action would have no impact on the NRC review and approval of the SLS LAR.

4.1.4 NMP2 Conclusion

Given the above, the RCS Pressure – Temperature LAR (Reference 5), the SLS LAR (Reference 15) and the MELLLA+ LARs are separate and independent licensing actions that the NRC can review and approve independently. Thus, they are not linked RLAs as defined in LIC-109.

4.2 Applicable Regulatory Requirements/Criteria

4.2.1 MELLLA+ and DSS-CD

10 CFR 50.46

10 CFR 50.46(a)(1)(i) states: "Each boiling or pressurized light-water nuclear power reactor fueled with uranium oxide pellets within cylindrical zircaloy or ZIRLO cladding must be provided with an emergency core cooling system (ECCS) that must be designed so that its calculated cooling performance following postulated loss-of-coolant accidents conforms to the criteria set forth in paragraph (b) of this section..."

The acceptance criteria of 10 CFR 50.46(b) are:

> "(1) Peak cladding temperature. The calculated maximum fuel element cladding temperature shall not exceed 2200° F.
>
> "(2) Maximum cladding oxidation. The calculated total oxidation of the cladding shall nowhere exceed 0.17 times the total cladding thickness before oxidation...
>
> "(3) Maximum hydrogen generation. The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam shall not exceed 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react.
>
> "(4) Coolable geometry. Calculated changes in core geometry shall be such that the core remains amenable to cooling.
>
> "(5) Long-term cooling. After any calculated successful initial operation of the ECCS, the calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed for the extended period of time required by the long-lived radioactivity remaining in the core."

Section 4.3 of the MELLLA+ SAR demonstrates that the requirements established in 10 CFR 50.46(a)((1)(i) and the acceptance criteria of 10 CFR 50.46(b)(1) through (5) will be met during operation of NMP2 in the MELLLA+ operating domain.

Appendix A to 10 CFR 50, General Design Criteria

10 CFR 50.36(c)(2)(ii), Criterion 2 requires that TS limiting conditions for operation include process variables, design features, and operating restrictions that are initial conditions of design basis accident analysis. Compliance with the TS ensures that the NMP2 system performance parameters are maintained within the values assumed in the safety analyses. The TS changes are supported by the safety analyses and continue to provide a level of protection comparable to the current TS. Applicable regulatory requirements and significant safety evaluations performed in support of the proposed changes are described in Attachments 8 and 10 of this Enclosure.

Information Notices 2009-23 and 2011-21

NRC Information Notice 2009-23, "Nuclear Fuel Thermal Conductivity Degradation," notified licensees that analyses performed using pre-1999 methods may be less conservative than previously understood (References 16 and 17). In addition, NRC Information Notice 2011-21, "Realistic Emergency Core Cooling System Evaluation Model Effects Resulting from Nuclear Fuel Thermal Conductivity Degradation," notifies addresses that the impact of irradiation on fuel thermal conductivity has the potential to cause errors in ECCS evaluation models, specifically a higher peak cladding temperature (Reference 18).

This issue does not apply to this submittal, because the MELLLA+ SAR utilized to justify operation in the MELLLA+ operating domain includes PRIME T-M methodology as discussed in Section 2.6.3 of Attachments 8 and 10 of this Enclosure.

4.2.2 Standby Liquid Control System – Isotopic Enrichment of Boron-10

Appendix A to 10 CFR 50, General Design Criteria

General Design Criterion (GDC) 26, "Reactivity control system redundancy and capability," states:

> "Two independent reactivity control systems of different design principles shall be provided. One of the systems shall use control rods, preferably including a positive means for inserting the rods, and shall be capable of reliably controlling reactivity changes to assure that under conditions of normal operation, including anticipated operational occurrences, and with appropriate margin for malfunctions such as stuck rods, specified acceptable fuel design limits are not exceeded. The second reactivity control system shall be capable of reliably controlling the rate of reactivity changes resulting from planned, normal power changes (including xenon burnout) to assure acceptable fuel design limits are not exceeded. One of the systems shall be capable of holding the reactor core subcritical under cold conditions."

For BWRs, the provisions of 10 CFR 50.62 require that the second reactivity control system be the SLS. Its function is, per the requirements, to inject into the reactor pressure vessel a borated water solution at a prescribed flow rate, concentration and boron-10 isotopic enrichment. The boron in the solution absorbs neutrons, thus providing reactivity control to shut down the reactor in the event the control rods fail to insert into the core.

GDC 27, "Combined reactivity control systems capability," states:

> "The reactivity control system shall be designed to have a combined capability, in conjunction with poison addition by the emergency core cooling system, of reliably controlling reactivity changes to assure that under postulated accident conditions and with appropriate margin for stuck rods the capability to cool the core is maintained.

The SLS is the poison addition system described in GDC 27.

10 CFR 50.62, "Requirements for reduction of risk from anticipated transients without scram (ATWS) events for light-water-cooled nuclear power plants"

10 CFR 50.62 (c)(4) states:

> "Each boiling water reactor must have a standby liquid control system (SLCS) with the capability of injecting, into the reactor pressure vessel a borated water solution at such a flow rate, level of boron concentration, and boron-10 isotope enrichment, and accounting for reactor pressure vessel volume, that the resulting reactivity control is at least equivalent to that resulting from injection of 86 gallons per minute of 13 weight percent sodium pentaborate decahydrate solution at the natural boron-l0 isotope abundance into a 251-inch inside diameter reactor pressure vessel for a given core design. The SLCS and its injection location must be designed to perform its function in a reliable manner..."

In the NRC-approved licensing topical report, NEDE-31096P-A, "Anticipated Transients Without Scram: Response to NRC ATWS Rule, 10 CFR 50.62," General Electric provides guidance on modifications to the SLC system to ensure licensee compliance with the ATWS rule. The NRC approved the methods presented in NEDE-31096P-A for use by Boiling Water Reactor licensees

to demonstrate compliance with the ATWS Rule. The application of this guidance demonstrates that the equivalency requirement of 10 CFR 50.62 is met.

10 CFR 50.67, "Accident source term"

10 CFR 50.67.b(1) provides guidance to licensees with respect to revision of the licensee's current accident source term in design basis radiological consequence analyses. Specifically, the regulation states that in order to revise the accident source term, a licensee shall apply for a license amendment under 10 CFR 50.90 and that the application shall contain an evaluation of the consequences of applicable design basis accidents previously analyzed in the safety analysis report.

The radiological consequences of certain design basis accidents (DBA) have been reevaluated using a full implementation of an Alternate Source Term as described in Regulatory Guide (RG) 1.183 (Reference 19) and NRC Standard Review Plan (SRP) 15.0.1 (Reference 20). The evaluation was performed at 120 percent of the original licensed power to bound the effects of future power uprates. The evaluation demonstrates that the calculated offsite exposures and control room doses meet the criteria of 10 CFR 50.67.

The supporting analyses for Alternate Source Term assume the pH of the suppression pool is controlled to prevent the re-evolution of iodine following a DBA LOCA. This is accomplished by injecting the SLS solution (i.e., boron solution) following a DBA LOCA to ensure pH is controlled to a value greater than 7.0. Analysis has confirmed that the SLS will continue to maintain suppression pool pH level above 7.0 following a LOCA which involves significant fission product releases.

Information Notice 2001-13

In response to potential non-conservatisms in pressure calculations related to SLS discharge pressure during ATWS scenarios, the NRC issued Information Notice (IN) 2001-13 (Reference 21). IN 2001-13 requested licensees to evaluate relief valve pressure margins on the SLS and confirm to the NRC that the systems remained in compliance with NRC regulations. NMPNS determined that the concerns identified in IN 2001-13 were applicable to NMP2.

This LAR is proposing to revise NMP2 TS SR 3.1.7.7 by increasing the minimum required NMP2 SLS pump test discharge pressure from 1,327 psig to 1,335 psig, while maintaining adequate margin for relief valve lift.

4.2.3 Safety Limit Minimum Critical Power Ratio

10 CFR 50.36

10 CFR 50.36(c)(1), requires that power reactor facility TS include safety limits for process variables that protect the integrity of certain physical barriers that guard against the uncontrolled release of radioactivity. The fuel cladding is one of the physical barriers that separate the radioactive materials from the environment. The purpose of the SLMCPR is to ensure that specified acceptable fuel design limits (SAFDLs) are not exceeded during steady state operation and analyzed transients. The integrity of this cladding barrier is related to its relative freedom from perforations or cracking. Fuel cladding perforations can result from thermal stresses, which can occur from reactor operation significantly above design conditions. 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 cladding damage could occur.

The GNF analysis presented in Attachments 9 and 11 of this Enclosure established that NMP2 continues to meet the requirement of 10 CFR 50.36(c)(1) with the increased acceptance criteria for the SLMCPR for two recirculation loops in operation.

Appendix A to 10 CFR 50, General Design Criteria

10 CFR 50.36(c)(2)(ii), Criterion 10 requires:

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

The fuel cladding must not sustain damage as a result of normal operation and abnormal operational transients. The reactor core safety limits are established to preclude violation of the fuel design criterion such that at least 99.9% of the fuel rods in the core would not be expected to experience the onset of transition boiling.

The GNF analysis presented in Attachments 9 and 11 of this Enclosure established that NMP2 continues to meet 10 CFR 50, Appendix A, Criterion 10 with the increased acceptance criteria for the SLMCPR for two recirculation loops in operation.

4.3 Precedent

4.3.1 MELLLA+ and DSS-CD

This license amendment request is based on approved GEH license topical reports (References 1 through 4) and their associated Safety Evaluation Reports. The NMP2 application follows the methodologies and limitations of those LTRs and their respective SERs. On January 21, 2010, Monticello Nuclear Generating Plant submitted a license amendment request to adopt the expanded MELLLA+ operating domain; this license amendment request remains under review by the NRC. (ADAMS Accession No. ML100280558) (Reference 22)

4.3.2 Standby Liquid Control System – Isotopic Enrichment of Boron-10

The NRC has approved a number of requests to increase the isotopic enrichment of boron-10 in the sodium pentaborate utilized to prepare the solution that is utilized in the SLS. These include:

- Columbia Generating Station - Issuance of Amendment Re: Increased Boron Concentration In Standby Liquid Control System (TAC No. ME4789), dated May 18, 2011, (ADAMS Accession No. ML111170370) (Reference 23)

  This amendment is similar to the NMP2 proposed change with respect to the increase in the isotopic boron-10 enrichment in the sodium pentaborate solution utilized in the SLS. For Columbia Generating Station, the boron-10 enrichment increase was from 22 atom percent to 44 atom percent.

- Susquehanna Steam Electric Station, Units 1 and 2 – Issuance of Amendment Re: Standby Liquid Control System (TAC Nos. MD1424 and MD1425), dated February 28, 2007, (ADAMS Accession No. ML070390215) (Reference 24)

  This amendment is similar to the NMP2 proposed change with respect to the increase in the isotopic boron-10 enrichment in the sodium pentaborate solution utilized in the SLS and the SLS volume decrease. The amendment also reduced the sodium pentaborate concentration; however, NMPNS is not proposing a change to the minimum sodium pentaborate solution concentration.

4.3.3 Safety Limit Minimum Critical Power Ratio

The NRC has approved a number of requests to increase the SLMCPR for two recirculation loops in operation that utilized GNF analysis to support the change. These include:

- LaSalle County Station, Unit 2 - Issuance of Amendment No. 192 Regarding Technical Specification Change For Safety Limit Minimum Critical Power Ratio (TAC No. ME9769), dated February 27, 2013 (ADAMS Accession No. ML13050A637) (Reference 25)

- Cooper Nuclear Station - Issuance of Amendment Re: Revision of Technical Specifications - Safety Limit Minimum Critical Power Ratio (TAC No. ME8853), dated November 9, 2012 (ADAMS Accession No. ML12299A092) (Reference 26)

4.4 Significant Hazards Consideration

Nine Mile Point Nuclear Station LLC (NMPNS) is requesting an amendment to Renewed Facility Operating License NPF-69 for Nine Mile Point Unit 2 (NMP2). The proposed amendment includes supporting changes to the NMP2 Technical Specifications (TSs) necessary to: 1) implement the Maximum Extended Load Line Limit Analysis Plus (MELLLA+) expanded operating domain; 2) change the stability solution to the Detect and Suppress Solution – Confirmation Density (DSS-CD); 3) use the TRACG04 analysis code; 4) increase the isotopic enrichment of boron-10 in the Standby Liquid Control system (SLS); and 5) increase the Safety Limit Minimum Critical Power Ratio (SLMCPR) for two recirculation loops in operation. The proposed changes to the NMP2 TSs:

The following is a list of the proposed changes to the NMP2 TSs:

- Revise Safety Limit (SL) 2.1.1.2 by increasing the SLMCPR for two recirculation loops in operation from ≥ 1.07 to ≥ 1.09
- Revise the acceptance criterion in TS 3.1.7, "Standby Liquid Control (SLC) System," Surveillance Requirement (SR) 3.1.7.7 by increasing the discharge pressure from ≥ 1,327 pounds per square inch gauge (psig) to ≥ 1,335 psig
- Revise the acceptance criterion in TS SR 3.1.7.10 by increasing the sodium pentaborate boron-10 enrichment requirement from ≥ 25 atom percent to ≥ 92 atom percent, and make a corresponding change in TS Figure 3.1.7-1, "Sodium Pentaborate Solution Volume/Concentration Requirements"

- Revise TS Figure 3.1.7-1 to account for the decrease in the minimum volume of the SLS tank from 4,558.6 gallons and 4,288 gallons at sodium pentaborate concentrations of 13.6% and 14.4%, respectively, to 1,600 gallons and 1,530 gallons at sodium pentaborate concentrations of 13.6% and 14.4%, respectively
- Change the Required Actions for Condition F of TS 3.3.1.1, "Reactor Protection System (RPS) Instrumentation"
- Change Condition G of TS 3.3.1.1
- Add new Conditions J and K to TS 3.3.1.1
- Correct an editorial error in Note 3 to TS SR 3.3.1.1.13 (i.e., "ORRM" is changed to "OPRM")
- Eliminate TS SR 3.3.1.1.16 and references to it in TS Table 3.3.1.1-1, "Reactor Protection System Instrumentation"
- Change the allowable value (AV) for TS Table 3.3.1.1-1, Function 2.b, Average Power Range Monitor (APRM) – Flow Biased Simulated Thermal Power (STP) – Upscale from "≤ 0.55W + 60.5% [Rated Thermal Power] RTP and ≤ 115.5% RTP" to "≤ 0.61W + 63.4% RTP and ≤ 115.5% RTP"
- Add a new note to TS Table 3.3.1.1-1, Function 2.b that requires the Flow Biased Simulated Thermal Power – Upscale scram setpoint to be reset to the values defined by the Core Operating Limits Report (COLR) to implement the Automated Backup Stability Protection (BSP) Scram Region in accordance with Required Action F.2.1 of TS 3.3.1.1
- Add a new note to TS Table 3.3.1.1-1, Function 2.e, Oscillation Power Range Monitor (OPRM) – Upscale to denote that following implementation of DSS-CD, DSS-CD is not required to be armed while in the DSS-CD Armed Region during the first reactor startup and during the first controlled shutdown that passes completely through the DSS-CD Armed Region. However, DSS-CD is considered operable and capable of automatically arming for operation at recirculation drive flow rates above the DSS-CD Armed Region
- Change the mode of applicability for TS Table 3.3.1.1-1, Function 2.e, OPRM-Upscale from Mode 1 to ≥ 18% RTP
- Change the allowable value for TS Table 3.3.1.1-1, Function 2.e from "As specified in the COLR" to "NA"
- Add a prohibition to TS Limiting Condition for Operation (LCO) 3.4.1, "Recirculation Loops Operating," that prohibits operation in the Maximum Extended Load Line Limit Analysis (MELLLA) domain or MELLLA+ expanded operating domain as defined in the COLR when in operation with a single recirculation loop
- Add Required Action B.2 to TS 3.4.1 to identify that intentional operation in the MELLLA domain or MELLLA+ domain as defined in the COLR is prohibited when a recirculation loop is declared "not in operation" due to a recirculation loop flow mismatch not within limits
- Revise TS 5.6.5.a.4 to replace "Reactor Protection System Instrumentation Setpoint for the OPRM – Upscale Function Allowable Value for Specification 3.3.1.1" with "The Manual Backup Stability Protection (BSP) Scram Region (Region I), the Manual BSP Controlled Entry Region (Region II), the modified APRM Simulated Thermal Power – High setpoints used in the OPRM (Function 2.e), Automated BSP Scram Region, and the BSP Boundary for Specification 3.3.1.1"
- Add TS 5.6.8, "OPRM Report," to define the contents of the report required by new Required Action F.2.2 of TS 3.3.1.1

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

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

Response: No.

The probability (frequency of occurrence) of Design Basis Accidents occurring is not affected by implementing the MELLLA+ operating domain and DSS-CD stability solution, because NMP2 continues to comply with the regulatory and design basis criteria established for plant equipment. A SLS failure is not a precursor of any previously evaluated accident in the NMP2 USAR. The increase to the SLMCPR for two recirculation loops in operation does not increase the probability of an evaluated accident. Consequently there is no change in the probability of an accident previously evaluated accident.

The spectrum of postulated transients was investigated and shown to remain within the NRC approved acceptance limits. Fuel integrity is maintained by meeting existing design and regulatory limits. Further, a probabilistic risk assessment demonstrates that the calculated core damage frequency and the large early release frequency do not significantly change due to operation in the MELLLA+ domain.

Challenges to the reactor coolant pressure boundary were evaluated for the MELLLA+ operating domain conditions (pressure, temperature, flow, and radiation) and were found to meet their acceptance criteria for allowable stresses and overpressure margin.

Challenges to the containment were evaluated and the containment and its associated cooling systems continue to meet the current licensing basis. The calculated post LOCA suppression pool temperature remains acceptable.

The SLS is used to mitigate the consequences of an Anticipated Transient Without SCRAM (ATWS) special event and is used to limit the radiological dose during a Loss of Coolant Accident (LOCA). The proposed changes do not affect the capability of the SLS to perform these two functions in accordance with the assumptions of the associated analyses. The ATWS evaluation with the proposed changes incorporated demonstrated that all the ATWS acceptance criteria are met. The ability of the SLS to mitigate radiological dose in the event of a LOCA by maintaining suppression pool pH $\geq$ 7.0 is not affected by these changes.

This proposed change to the SLMCPR for two recirculation loops in operation does not result in any modification to the design or operation of the systems that are used in mitigation of accidents. Limits have been established, consistent with NRC approved methods, to ensure that fuel performance during normal, transient, and accident conditions is acceptable. The proposed change to the SLMCPR for two recirculation loops in operation continues to conservatively establish this safety limit such that the fuel is protected during normal operation and during any plant transients or anticipated operational occurrences.

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?

Response: No.

Equipment that could be affected by implementing the MELLLA+ operating domain and DSS-CD stability solution was evaluated. No new operating mode, safety-related equipment lineup, accident scenario, or equipment failure mode was identified. The full spectrum of accident considerations was evaluated and no new or different kind of accident was identified. The MELLLA+ operating domain and DSS-CD stability solution use developed technology and apply it within the capabilities of existing plant safety-related equipment in accordance with the regulatory criteria (including NRC approved codes, standards and methods). No new accident or event precursor was identified.

The long-term stability solution is being changed from the currently approved Option III solution to DSS-CD. DSS-CD is designed to identify the power oscillation upon inception and initiate control rod insertion (scram) to terminate the oscillations prior to any significant amplitude growth. DSS-CD is based on the same hardware design as Option III. However, it introduces an enhanced detection algorithm that detects the inception of power oscillations and generates an earlier power suppression trip signal exclusively based on successive period confirmation recognition. The existing Option III algorithms are retained (with generic setpoints) to provide defense-in-depth protection for unanticipated reactor instability events.

Structures, systems and components (SSCs) 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. The physical changes to the SLS are limited to the increase in the boron-10 enrichment of the sodium pentaborate solution in the SLS storage tank, the corresponding decrease in the net sodium pentaborate solution volume requirement in the SLS storage tank, the increase in the SLS pump discharge pressure acceptance criterion, and the associated instrumentation changes. The proposed changes do not otherwise affect the design or operation of the SLS.

This proposed change to the SLMCPR for two recirculation loops in operation does not result in any modification to the design or operation of the systems that are used in the mitigation of accidents. The proposed change to the SLMCPR for two recirculation loops in operation assures that safety criteria are maintained.

The proposed changes do 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.

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?

Response: No.

The MELLLA+ operating domain affects only design and operational margins. Challenges to the fuel, reactor coolant pressure boundary, and containment were evaluated for the MELLLA+ operating domain conditions. Fuel integrity is maintained by meeting existing design and regulatory limits. The calculated loads on affected SSCs, including the reactor coolant pressure boundary, will remain within their design specifications for design basis event categories. No NRC acceptance criterion is exceeded.

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 have demonstrated that the NMP2 SSCs are capable of safely performing at MELLLA+ conditions. The analyses identified and defined the major input parameters to the Nuclear Steam Supply System (NSSS), analyzed NSSS design transients, and evaluated the capabilities of the NSSS fluid systems, NSSS/Balance of Plant (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 of achieving MELLLA+ 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 change in the operating domain. Calculated loads on SSCs important to safety have been shown to remain within design allowables with MELLLA+ 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 MELLLA+ 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.

The SLS is used to mitigate the consequences of an ATWS event and is used to limit the radiological dose during a LOCA. The proposed changes do not affect the capability of the SLS to perform these two functions in accordance with the assumptions of the associated analyses. The ATWS evaluation with the proposed changes incorporated demonstrated that all the ATWS acceptance criteria are met. The ability of the SLS to mitigate radiological dose in the event of a LOCA by maintaining suppression pool pH $\geq$ 7.0 is not affected by these changes.

This proposed change to the SLMCPR for two recirculation loops in operation provides a margin of safety by ensuring that no more than 0.1% of fuel rods are expected to be in boiling transition if the MCPR limit is not violated. The proposed change will ensure the appropriate level of fuel protection is maintained. Additionally, operational limits are established based on the proposed SLMCPR to ensure that the SLMCPR is not violated during all modes of operation. This will ensure that the fuel design safety criteria are met (i.e., that at least 99.9% of the fuel rods do not experience transition boiling during normal operation as well as anticipated operational occurrences).

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

4.5 Conclusions

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, NMPNS 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

A review has determined that the proposed amendment would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or would change an inspection or surveillance requirement. However, the proposed amendment does not involve: (i) a significant hazards consideration; (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite; or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

6.0 REFERENCES

1. GE Hitachi Nuclear Energy, "General Electric Boiling Water Reactor Maximum Extended Load Line Limit Analysis Plus Licensing Topical Report," NEDC-33006P-A, Revision 3, June 2009 and NEDO-33006-A, Revision 3, June 2009.

2. GE Hitachi Nuclear Energy, "GE Hitachi Boiling Water Reactor, Detect And Suppress Solution - Confirmation Density," NEDC-33075P, Revision 7, June 2011; and Anthony J. Mendiola (NRC) to Jerald G. Head (GEH), "Revised Draft Safety Evaluation for GE-Hitachi Nuclear Energy Americas, LLC Topical Report NEDC-33075P, Revision 7, 'GE Hitachi Boiling Water Reactor Detect and Suppress Solution – Confirmation Density' (TAC No. ME6577)," dated August 6, 2013.

3. GE Hitachi Nuclear Energy, "DSS-CD TRACG Application," NEDE-33147P-A, Revision 4, August 2013.

4. a. GE Hitachi Nuclear Energy, "Applicability of GE Methods to Expanded Operating Domains," NEDC-33173P-A, Revision 4, November 2012.

   b. Letter from R. Kingston (GEH) to NRC, "Clarification of Stability Evaluations – NEDC-33173P," MFN 08-541, June 25, 2008.

   c. Letter from J. Harrison (GEH) to NRC, "Implementation of Methods Limitations – NEDC-33173," MFN 08-693, September 18, 2008.

   d. Letter from J. Harrison (GEH) to NRC, "NEDC-33173P – Implementation of Limitation 12," MFN 09-143, February 27, 2009.

   e. GE Hitachi Nuclear Energy, "Implementation of PRIME Models and Data in Downstream Methods," NEDO-33173, Supplement 4-A, Revision 1, November 2012.

5. Letter from K. Langdon (NMPNS) to the Document Control Desk (NRC), License Amendment Request Pursuant to 10 CFR 50.90: Relocation of Pressure and Temperature Limit Curves to the Pressure and Temperature Limits Report, dated November 21, 2012 (ADAMS Accession Number ML123380336).

6. Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," U.S. NRC, Revision 2, May 2011.

7. NEDE-31096-A, "Anticipated Transients Without Scram Response to NRC ATWS Rule 10CFR50.62," February 1987.

8. GE Hitachi Nuclear Energy, "General Electric Standard Application for Reactor Fuel," NEDE-24011-P-A, and NEDE-24011-P-A-US, Revision 19 April 2012.

9 NEDC-32601P-A, "Methodology and Uncertainties for Safety Limit MCPR Evaluations," August 1999.

10 NEDC-32694P-A, "Power Distribution Uncertainties for Safety Limit MCPR Evaluations" August 1999.

11 NEDC-32505P-A, "R-Factor Calculation Method for GE11, GE12 and GE13 Fuel," Revision 1, July 1999.

12. Letter from (NRC) to K. Langdon (NMPNS), Nine Mile Point Nuclear Station, Unit No. 2 – Issuance of Amendment Re: Extended Power Uprate (TAC No. ME1476), dated December 22, 2011 (ADAMS Accession Number ML1133000041).

13. B. Vaidya (NRC), Summary of February 27,2013, Meeting with Nine Mile Point Nuclear Station, Unit 2, to discuss Planned Amendment Request on Implementation of MELLLA+ (TAC No. MF0587), dated March 13, 2013 (ADAMS Accession Number ML13059A374)

14. NRR Office Instruction LIC-109, "Acceptance Review Procedures," Revision 1.

15. Letter from P.M. Swift (NMPNS) to Document Control Desk (NRC), License Amendment Request Pursuant to 10 CFR 50.90: Standby Liquid Control System – Increase in Isotopic Enrichment of Boron-10, dated July 5, 2013 (ADAMS Accession Number ML13197A221).

16. Information Notice 2009-23, "Nuclear Fuel Thermal Conductivity Degradation," U.S. Nuclear Regulatory Commission, dated October 8, 2009.

17. Information Notice 2009-23, Supplement 1, "Nuclear Fuel Thermal Conductivity Degradation," U.S. Nuclear Regulatory Commission, dated October 26, 2012.

18. Information Notice 2011-21, "Realistic Emergency Core Cooling System Evaluation Model Effects Resulting from Nuclear Fuel Thermal Conductivity Degradation," U.S. Nuclear Regulatory Commission, dated December 13, 2011.

19. Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," July 2000 (ADAMS Accession No. ML003716792).

20. Standard Review Plan (SRP) 15.0.1, "Radiological Consequence Analyses Using Alternative Source Terms," Revision 0, July 2000 (ADAMS Accession No. ML003721661).

21. Information Notice 2001-13, "Inadequate Standby Liquid Control System Relief Valve Margin," U.S. Nuclear Regulatory Commission, August 10, 2001.

22. Letter from T. J. O'Connor (Monticello Nuclear Generating Plant) to U. S. Nuclear Regulatory Commission, "License Amendment Request: Maximum Extended Load Line Limit Analysis Plus," dated January 21, 2010 (ADAMS Accession No. ML100280558).

23. Letter from M. C. Thadani (NRC) to M. E. Reddemann (Energy Northwest), "Columbia Generating Station – Issuance of Amendment Re: Increased Boron Concentration in Standby Liquid Control System (TAC No. ME4789)," dated May 18, 2011 (ADAMS Accession No. ML111170370).

24. Letter from R. V. Guzman (NRC) to B. T. McKinney (PPL Susquehanna, LLC), "Susquehanna Steam Electric Station, Units 1 and 2 – Issuance of Amendment Re: Standby Liquid Control System (TAC Nos. MD1424 and MD1425)," dated February 28, 2007 (ADAMS Accession No. ML070390215).

25. Letter from N. DiFrancesco (NRC) to M. J. Pacillo (Exelon Nuclear), LaSalle County Station, Unit 2 -Issuance of Amendment No. 192 Regarding Technical Specification Change for Safety Limit Minimum Critical Power Ratio (TAC No. ME9769), dated February 27, 2013 (ADAMS Accession No. ML13050A637).

26. Letter from L. E. Wilkins (NRC) to B. J. O'Grady (Nebraska Public Power District), Cooper Nuclear Station -Issuance of Amendment Re: Revision of Technical Specifications -Safety Limit Minimum Critical Power Ratio (TAC No. ME8853), dated November 9, 2012 (ADAMS Accession No. ML12299A092).

ATTACHMENT 1

NINE MILE POINT UNIT 2
PROPOSED CHANGES TO TECHNICAL SPECIFICATIONS
(MARK-UPS)

The current versions of the following Technical Specification pages have been marked-up to reflect the proposed changes:

2.0-1
3.1.7-3 and 4
3.3.1.1-2 and 3
3.3.1.1-6
3.3.1.1-8
3.4.1-1 and 2
5.6-3 and 4

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 ≤ 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~~ 1.09 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:

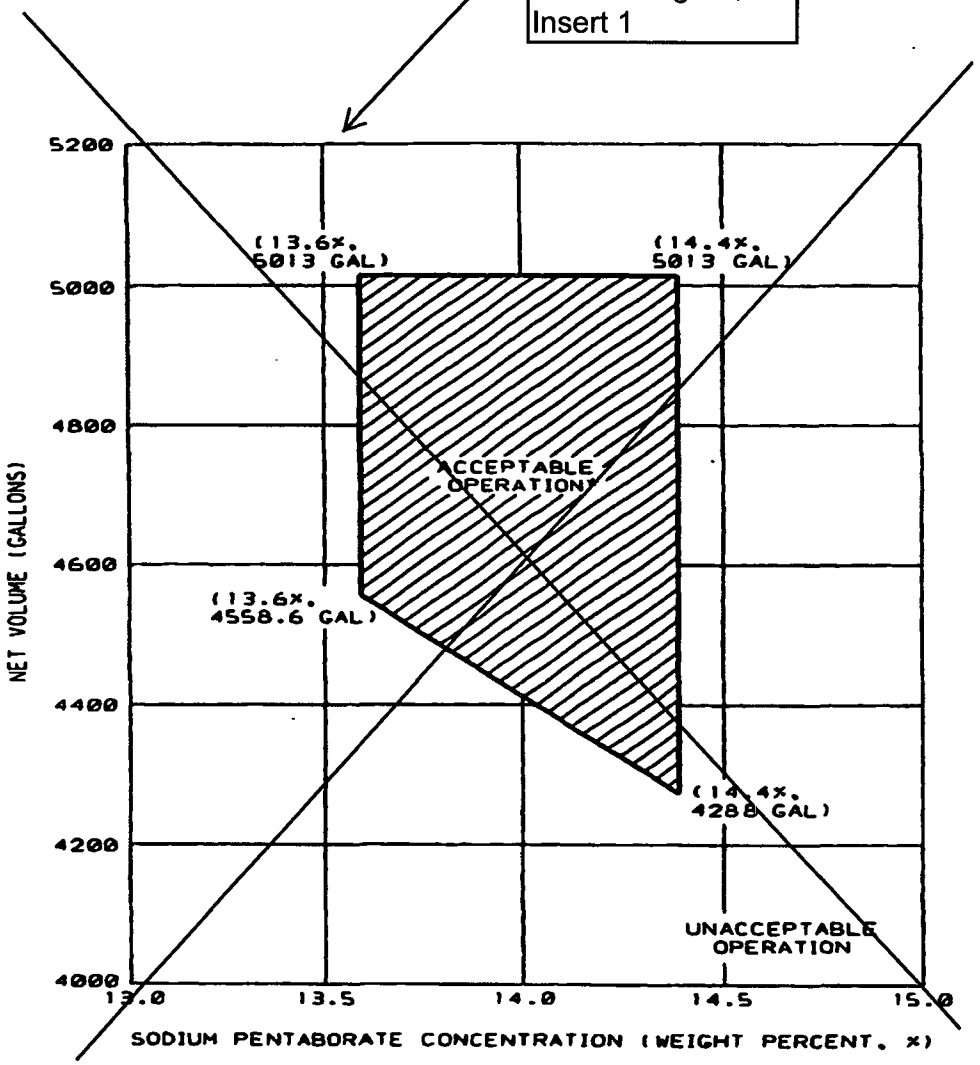
2.2.1 Restore compliance with all SLs; and

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 ≥ 41.2 gpm at a discharge pressure ≥ ~~1327~~ 1335 psig.	In accordance with the Inservice Testing Program
SR 3.1.7.8	Verify flow through one SLC subsystem from pump into reactor pressure vessel.	24 months on a STAGGERED TEST BASIS
SR 3.1.7.9	Verify all heat traced piping between storage tank and pump suction valve is unblocked.	24 months AND Once within 24 hours after piping temperature is restored to ≥ 70°F
SR 3.1.7.10	Verify sodium pentaborate enrichment is ≥ ~~25~~ 92 atom percent B-10.	Prior to addition to SLC tank

Replace with attached figure, Insert 1



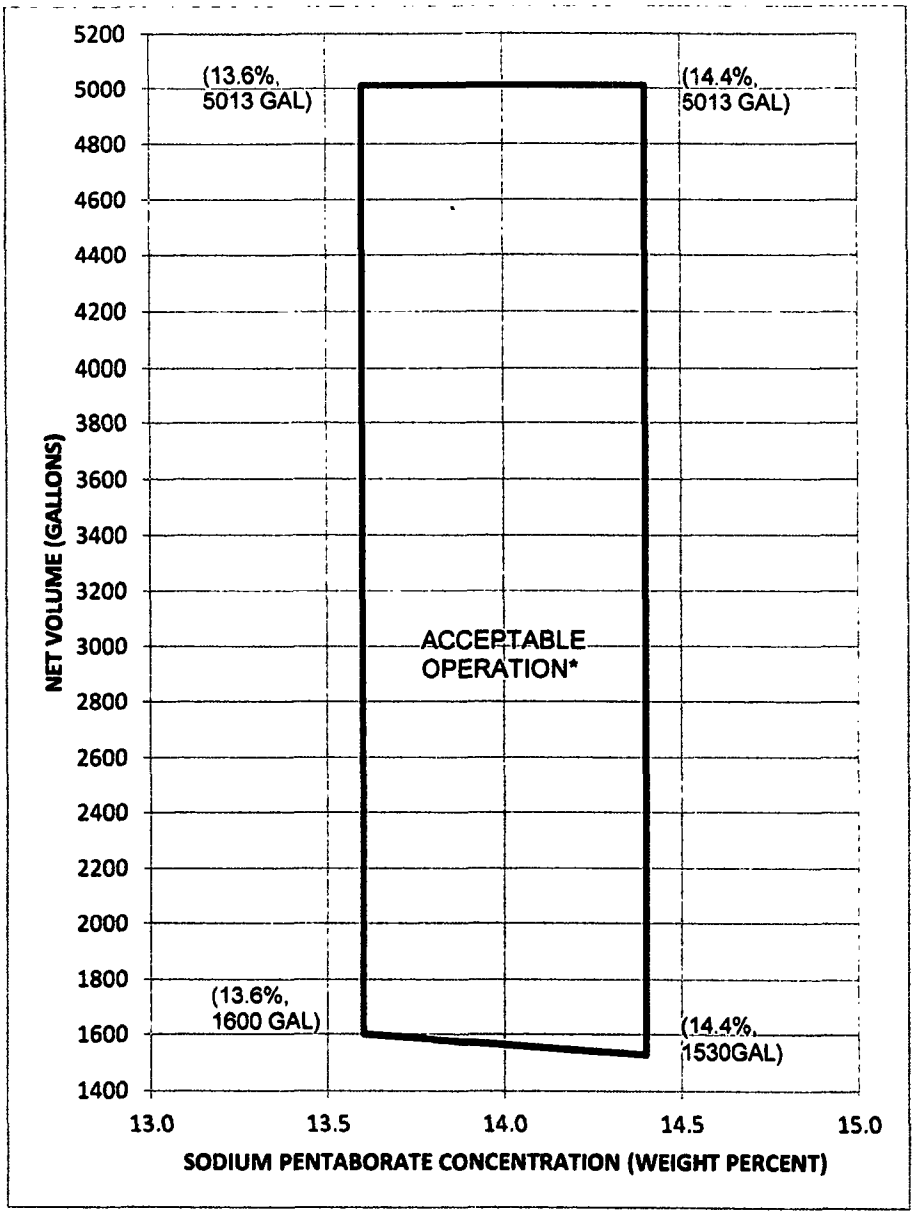
TS Insert 1 - Figure 3.1.7-1

92

*For Boron-10 Isotope Enrichment ≥ ~~25~~ Atom Percent

Figure 3.1.7-1 (Page 1 of 1)
Sodium Pentaborate Solution Volume/Concentration Requirements

TS Insert 1 - Figure 3.1.7-1



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 < 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.~~ See TS Insert 2a - Action F ~~AND~~ ~~F.2 Restore required channel to OPERABLE status.~~	~~12 hours~~ ~~120 days~~
G. ~~Required Action and associated Completion Time of Condition F not met.~~ ~~OR~~ 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)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
H. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	H.1 Be in MODE 3.	12 hours
I. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	I.1 Initiate action to fully insert all insertable control rods in core cells containing one or more fuel assemblies.	Immediately

See TS Insert 2b - Actions J and K

SURVEILLANCE REQUIREMENTS

----------NOTE----------

1. Refer to Table 3.3.1.1-1 to determine which SRs apply for each RPS Function.

2. 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 RPS trip capability.

SURVEILLANCE		FREQUENCY
SR 3.3.1.1.1	Perform CHANNEL CHECK.	12 hours
SR 3.3.1.1.2	Perform CHANNEL CHECK.	24 hours

(continued)

TS Insert 2a – Action F

F. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	F.1	Initiate action to implement the Manual BSP Regions defined in the COLR.	Immediately
	AND		
	F.2.1	Implement the Automated BSP Scram Region using the modified APRM Simulated Thermal Power - High scram setpoints defined in the COLR.	12 hours
	AND		
	F.2.2	Initiate action in accordance with Specification 5.6.8.	90 days

Insert 2b – Actions J and K

J. Required Action and associated Completion Time of Condition F not met.	J.1	Initiate action to implement the Manual BSP Regions defined in the COLR.	Immediately
	AND		
	J.2	Reduce operation to below the BSP Boundary defined in the COLR.	12 hours
	AND		
	J.3	-------NOTE---------- LCO 3.0.4 is not applicable ---------- Restore required channel to OPERABLE.	 120 days
K. Required Action and associated Completion Time of Condition J not met.	K.1	Reduce THERMAL POWER to less than 18% RTP.	4 hours

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 ~~ORRM~~ 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 ≥ 26% RTP.	24 months
SR 3.3.1.1.16	~~Verify APRM OPRM Upscale Function is not bypassed when THERMAL POWER is ≥ 26% RTP and recirculation drive flow is < 60% of rated recirculation drive flow.~~	~~24 months~~

Corrected to 'OPRM'

Deleted.

(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[c],[d]	≤ 0.61W + 63.4% RTP ~~≤ .55W + 66.5% RTP~~ and ≤ 115.5% RTP[b] (e)
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	≥ 18% RTP (f)	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~~ NA
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

See TS Insert 3 for Note (e)

See TS Insert 3 for Note (f)

(continued)

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

(b) Allowable Value is .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.

(d) The instrument channel setpoint shall be reset to a value 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 Insert 3

(e) With OPRM Upscale (function 2.e) inoperable, reset the APRM-STP High scram setpoint to the values defined by the COLR to implement the Automated BSP Scram Region in accordance with Action F.2.1 of this Specification.

(f) Following DSS-CD implementation, DSS-CD is not required to be armed while in the DSS-CD Armed Region during the first reactor startup and during the first controlled shutdown that passes completely through the DSS-CD Armed Region. However, DSS-CD is considered OPERABLE and shall be maintained OPERABLE and capable of automatically arming for operation at recirculation drive flow rates above the DSS-CD Armed Region.

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.1 Recirculation Loops Operating

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

OR

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

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

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

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

APPLICABILITY: MODES 1 and 2.

d. Intentional operation with only one recirculation loop in operation is prohibited while operating in the MELLLA domain or MELLLA+ domain as defined in the COLR.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. No recirculation loops in operation.	A.1	Be in MODE 2.	6 hours
	AND		
	A.2	Be in MODE 3.	12 hours
B. Recirculation loop flow mismatch not within limits.	B.1	Declare the recirculation loop with lower flow to be "not in operation."	2 hours
C. Requirements of the LCO not met for reasons other than Conditions A and B.	C.1	Satisfy the requirements of the LCO.	4 hours
D. Required Action and associated Completion Time of Condition C not met.	D.1	Be in MODE 3.	12 hours

TS Insert 4

TS INSERT 4

AND		
B.2	Prohibit operation in the MELLLA domain or MELLLA+ domain defined in the COLR.	2 hours

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

1. The APLHGR for Specification 3.2.1.

2. The MCPR for Specification 3.2.2.

3. The LHGR for Specification 3.2.3.

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

See TS Insert 5, Item 5.6.5.a.4

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

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

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

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

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

(continued)

TS Insert 5

5.6.5.a

4. The Manual Backup Stability Protection (BSP) Scram Region (Region I), the Manual BSP Controlled Entry Region (Region II), the modified APRM Simulated Thermal Power - High setpoints used in the OPRM (Function 2.e), Automated BSP Scram Region, and the BSP Boundary for Specification 3.3.1.1.

5.6 Reporting Requirements (continued)

5.6.6 Post Accident Monitoring (PAM) Instrumentation Report

When a report is required by Condition B or F of LCO 3.3.3.1, "Post Accident Monitoring (PAM) Instrumentation," a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status.

See TS Insert 6 for added TS 5.6.8

TS Insert 6

5.6.8 OPRM Report

When a report is required by Required Action F.2.2 of TS 3.3.1.1, "RPS Instrumentation," a report shall be submitted within 90 days of entering CONDITION F. The report shall outline the preplanned means to provide backup stability protection, the cause of the inoperability, and the plans to schedule for restoring the required instrumentation channels to OPERABLE status.

ATTACHMENT 2

NINE MILE POINT UNIT 2
CHANGES TO BASES FOR TECHNICAL SPECIFICATIONS
(MARK-UPS)

The current versions of the following Technical Specifications Bases pages have been marked-up to reflect the proposed changes. These Bases pages are provided for information only.

B 3.1.7-5
B 3.3.1.1-9
B 3.3.1.1-12 through 14
B 3.3.1.1-26 and 27
B 3.3.1.1-34
B 3.3.1.1-36
B 3.4.1-3 through 6

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.

1335

SR 3.1.7.7

Demonstrating each SLC System pump develops a flow rate ≥ 41.2 gpm at a discharge pressure ≥ ~~1327~~ psig ensures that pump performance has not degraded during the fuel cycle. This minimum pump flow rate requirement ensures that, when

(continued)

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%~~ 0.61W + 61.4% RTP (or 0.52 (W - 5%) + 49.4% 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)

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

2.d. Average Power Range Monitor – Inop (continued)

provides an input to all four 2-Out-Of-4 Voter channels, one channel per 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 Inop signal, two channels in both logic channels in each trip system will trip, producing a scram. 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.

Twelve channels of Average Power Range Monitor – Inop 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.

There is no Allowable Value for this Function.

This Function is required to be OPERABLE in the MODES where the other APRM Functions are required.

See TS Bases Insert 1

2.e. Average Power Range Monitor OPRM – Upscale

~~The APRM channels provide the primary indication of neutron flux within the core and respond almost instantaneously to neutron flux increases. In addition, the channels can detect oscillatory changes in neutron flux. The Average Power Range Monitor Oscillation Power Range Monitor (OPRM) – Upscale Function is capable of detecting neutron flux oscillations indicative of thermal-hydraulic instabilities, by detecting the related neutron flux oscillations, and generating a trip signal before the MCPR Safety Limit is exceeded (Ref. 14).~~

~~The OPRM – Upscale Function receives input signals from the LPRMs, which are combined into cells (four LPRMs per cell) for evaluation by the OPRM algorithms. An OPRM – Upscale trip signal is issued from an APRM channel when the period based detection algorithm in that channel detects oscillatory changes in neutron flux, indicated by the combined signals of the LPRM detectors in a cell, with periodic confirmations and relative cell amplitude exceeding specific setpoints. One or more OPRM cells in a channel exceeding the trip conditions will result in a channel trip.~~

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

See TS Bases Insert 1

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 ≥ 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)

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

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

~~still considered OPERABLE provided the automatic trip is not bypassed when the proper power and flow conditions exist. Requiring the Average Power Range Monitor OPRM – Upscale Function to be OPERABLE in MODE 1 provides adequate margin to cover the operating region where oscillations may occur as well as the operating regions from which the plant might enter the potential instability region without operator action.~~

2.f. Average Power Range Monitor 2-Out-Of-4 Voter

The 2-Out-Of-4 Voter Function provides the interface between the other APRM Functions and the RPS trip system logic, and as such, supports the safety analyses applicable to the other APRM Functions. Each APRM provides two inputs to all four 2-Out-Of-4 Voter channels (one input is common for Functions 2.a, 2.b, 2.c, and 2.d). The four 2-Out-Of-4 Voter channels are divided into two groups of two channels, with each group providing inputs into one RPS trip system (each channel inputs to one logic channel, similar to most other RPS instrumentation Functions). When two trip signals from any combination of APRM Functions 2.a, 2.b, 2.c, and 2.d are received (from different APRMs) by a 2-Out-Of-4 Voter channel, or two trip signals from any combination of APRM Function 2.e are received by a 2-Out-Of-4 Voter channel, the 2-Out-Of-4 Voter channel provides a trip signal to its associated trip system. Any one 2-Out-Of-4 Voter channel can trip the associated trip system (i.e., a one-out-of-two logic). In addition, while each 2-Out-Of-4 Voter channel provides two inputs to its associated trip system, only one of the inputs is required for OPERABILITY.

Each 2-Out-Of-4 Voter channel also includes self-diagnostic functions. If any channel detects a critical fault in its own processing, a trip signal is provided to its associated trip system. Unlike the other APRM Functions, a bypass capability for the 2-Out-Of-4 Voter Function is not provided.

Four channels of the APRM 2-Out-Of-4 Voter Function with two channels 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.

(continued)

TS Bases Insert 1

2.e. Average Power Range Monitor OPRM-Upscale

The OPRM Upscale Function provides compliance with GDC 10 and GDC 12, thereby providing protection from exceeding the fuel MCPR safety limit (SL) due to anticipated thermal-hydraulic power oscillations.

Reference 17 describes the Detect and Suppress - Confirmation Density (DSS-CD) long-term stability solution and the licensing basis Confirmation Density Algorithm (CDA). Reference 17 also describes the DSS-CD Armed Region and the three additional algorithms for detecting thermal-hydraulic instability related neutron flux oscillations: the period based detection algorithm (PBDA), the amplitude based algorithm (ABA), and the growth rate algorithm (GRA). All four algorithms are implemented in the OPRM Upscale Function, but the safety analysis takes credit only for the CDA. The remaining three algorithms provide defense in depth and additional protection against unanticipated oscillations. OPRM Upscale Function OPERABILITY is based only on the CDA.

The OPRM Upscale Function receives input signals from the local power range monitors (LPRMs) within the reactor core, which are combined into "cells" for evaluation by the OPRM algorithms.

DSS-CD operability requires at least 8 responsive OPRM cells per channel.

The OPRM Upscale Function is required to be OPERABLE when the plant is ≥ 18% RTP, which is established as a power level that is greater than or equal to 5% below the lower boundary of the Armed Region. This requirement is designed to encompass the region of power-flow operation where anticipated events could lead to thermal-hydraulic instability and related neutron flux oscillations. The OPRM Upscale Function is automatically trip-enabled when THERMAL POWER, as indicated by the APRM Simulated Thermal Power, is ≥23% RTP corresponding to the plant-specific MCPR monitoring threshold and reactor recirculation drive flow, is less than 75% of rated flow. This region is the OPRM Armed Region. Note f allows for entry into the DSS-CD Armed Region without automatic arming of DSS-CD prior to completely passing though the DSS-CD Armed Region during both a single startup and a single shutdown following DSS-CD implementation. Note f reflects the need for plant data collection in order to test the DSS-CD equipment. Testing the DSS-CD equipment ensures its proper operation and prevents spurious reactor trips. Entry into the DSS-CD Armed Region without automatic arming of DSS-CD during this initial testing phase also allows for changes in plant operations to address maintenance or other operational needs. However, during this initial testing period, the OPRM upscale function is OPERABLE and DSS-CD operability and capability to automatically arm shall be maintained at recirculation drive flow rates above the DSS-CD Armed Region flow boundary.

An OPRM Upscale trip is issued from an OPRM channel when the confirmation density algorithm in that channel detects oscillatory changes in the neutron flux. The oscillations are indicated by periodic confirmations and amplitude exceeding specified setpoints for a specified number of OPRM cells in

TS Bases Insert 1
(Continued)

the channel. An OPRM Upscale trip is also issued from the channel if any of the defense-in-depth algorithms (PBDA, ABA, GRA) exceed its trip condition for one or more cells in that channel.

Three of the four channels are required to be operable. Each channel is capable of detecting thermal-hydraulic instabilities, by detecting the related neutron flux oscillations, and issuing a trip signal before the MCPR SL is exceeded. There is no Allowable Value for this Function.

The OPRM Upscale function settings are not traditional instrumentation setpoints determined under an instrument setpoint methodology. There is no Allowable Value for this Function and the OPRM Upscale Function is not LSSS SL-related and Reference 18 confirms that the OPRM Upscale Function settings based on DSS-CD also do not have traditional instrumentation setpoints determined under an instrument setpoint methodology.

ACTIONS

C.1 (continued)

The Completion Time is intended to allow the operator time to evaluate and repair any discovered inoperabilities. The 1 hour Completion Time is acceptable because it minimizes risk while allowing time for restoration or tripping of channels.

D.1

Required Action D.1 directs entry into the appropriate Condition referenced in Table 3.3.1.1-1. The applicable Condition specified in the Table is Function and MODE or other specified condition dependent and may change as the Required Action of a previous Condition is completed. Each time an inoperable channel has not met any Required Action of Condition A, B, or C, and the associated Completion Time has expired, Condition D will be entered for that channel and provides for transfer to the appropriate subsequent Condition.

E.1, G.1, and H.1

If the channel(s) is not restored to OPERABLE status or placed in trip (or the associated trip system placed in trip) within the allowed Completion Time, the plant must be placed in a MODE or other specified condition in which the LCO does not apply. The Completion Times are reasonable, based on operating experience, to reach the specified condition from full power conditions in an orderly manner and without challenging plant systems. In addition, the Completion Time of Required Action E.1 is consistent with the Completion Time provided in LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)."

See TS Bases Insert 2a for Action F

F.1

~~If the channel(s) is not restored to OPERABLE status or placed in trip (or the associated trip system placed in trip) within the allowed Completion Time, an alternate method to detect and suppress thermal-hydraulic instability oscillations must be initiated within 12 hours. This is acceptable since suitable methods exist to properly monitor for and suppress thermal-hydraulic instability oscillations.~~

(continued)

BASES

ACTIONS

F.1 (continued)

~~However, since these methods involve operator actions, the allowance to operate in this condition utilizing manual detect and suppress methods exists for only 120 days (i.e., the channel(s) must be restored within 120 days). If the channel(s) is not restored within 120 days, Condition G must be entered and its Required Actions taken.~~

I.1

If the channel(s) is not restored to OPERABLE status or placed in trip (or the associated trip system placed in trip) within the allowed Completion Time, the plant must be placed in a MODE or other specified condition in which the LCO does not apply. This is done by immediately initiating action to fully insert all insertable control rods in core cells containing one or more fuel assemblies. Control rods in core cells containing no fuel assemblies do not affect the reactivity of the core and are, therefore, not required to be inserted. Action must continue until all insertable control rods in core cells containing one or more fuel assemblies are fully inserted.

See TS Bases Insert 2b for Actions J and K

SURVEILLANCE REQUIREMENTS

As noted at the beginning of the SRs, the SRs for each RPS instrumentation Function are located in the SRs column of Table 3.3.1.1-1.

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 RPS 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 RPS reliability analysis (Ref. 10) assumption of 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 RPS will trip when necessary.

(continued)

TS Bases Insert 2a

ACTIONS

F.1

If OPRM Upscale trip capability is not maintained, then actions must be taken to establish Manual Backup Stability Protection (BSP). The Manual BSP Regions are described in Reference 17. The Manual BSP Regions are procedurally established consistent with the guidelines identified in Reference 17 and require specified manual operator actions if certain predefined operational conditions occur.

The Completion Time of immediate is based on the importance of limiting the period of time during which no automatic or alternate detect and suppress trip capability is in place.

F.2.1 and F.2.2

Actions F.2.1 and F.2.2 are both required to be taken in conjunction with Action F.1 if OPRM Upscale trip capability is not maintained. As described in Section 7.4 of Reference 17, the Automated BSP Scram Region is designed to avoid reactor instability by automatically preventing entry into the region of the power and flow-operating map that is susceptible to reactor instability. The reactor trip would be initiated by the modified APRM Simulated Thermal Power – High scram setpoints for flow reduction events that would have terminated in the Manual BSP Region I. The Automated BSP Scram Region ensures an early scram and SLMCPR protection.

The Completion Time of 12 hours to complete the specified actions is reasonable, based on operational experience, and based on the importance of restoring an automatic reactor trip for thermal hydraulic instability events.

Backup Stability Protection is intended as a temporary means to protect against thermal-hydraulic instability events. The reporting requirements of Specification 5.6.8 document the corrective actions and schedule to restore the required channels to an OPERABLE status. The Completion Time of 90 days is adequate to allow time to evaluate the cause of the inoperability and to determine the appropriate corrective actions and schedule to restore the required channels to OPERABLE status.

TS Bases Insert 2b

J.1

If the Required Actions F are not completed within the associated Completion Times, then Action J is required. The Bases for the Manual BSP Regions and associated Completion Time is addressed in the Bases for F.1. The Manual BSP Regions are required in conjunction with the BSP Boundary.

J.2

The BSP Boundary, as described in Section 7.3 of Reference 17, defines an operating domain where potential instability events can be effectively addressed by the specified BSP manual operator actions. The BSP Boundary is constructed such that the immediate final statepoint for a flow reduction event initiated from this boundary and terminated at the core natural circulation line (NCL) would not exceed the Manual BSP Region I stability criterion. Potential instabilities would develop slowly as a result of the feedwater temperature transient (Reference 17).

The Completion Time of 12 hours to complete the specified actions is reasonable, based on operational experience, to reach the specific condition from full power conditions in an orderly manner and without challenging plant system.

J.3

The BSP is a temporary means for protection against thermal-hydraulic instability events. An extended period of inoperability without automatic trip capability is not justified. Consequently, the required channels are required to be restored to OPERABLE status within 120 days.

Based on engineering judgment, the likelihood of an instability event that could not be adequately handled by the use of the BSP Regions (See Action J.1) and the BSP Boundary (See J.2) during a 120-day period is negligibly small. The 120-day period is intended to allow for the case where limited design changes or extensive analysis might be required to understand or correct some unanticipated characteristic of the instability detection algorithms or equipment. This action is not intended and was not evaluated as a routine alternative to returning failed or inoperable equipment to OPERABLE status. Correction of routine equipment failure or inoperability is expected to normally be accomplished within the completion times allowed for Actions for Conditions A and B.

A Note is provided to indicate that LCO 3.0.4 is not applicable. The intent of that Note is to allow plant startup within the 120 day Completion Time for Required Action J.3. The primary purpose of this exclusion is to allow an orderly completion of design and verification activities, in the event of a required design change, without undue impact on plant operation.

TS Bases Insert 2b (Continued)

K.1

If the required channels are not restored to OPERABLE status and the Required Actions of J are not met within the associated Completion Times, then the plant must be placed in an operating condition in which the LCO does not apply. To achieve this status, the plant must be brought to less than 18% RTP within 4 hours. The allowed Completion Time is reasonable, based on operating experience, to reach the specified operating power level from full power conditions in an orderly manner and without challenging plant systems.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.1.15 (continued)

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 ≥ 26% RTP to ensure that the calibration is valid.

If any bypass channel setpoint is nonconservative (i.e., the Functions are bypassed at ≥ 26% 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.

Deleted.

SR 3.3.1.1.16

~~This SR ensures that scrams initiated from the APRM OPRM – Upscale Function will not be inadvertently bypassed when THERMAL POWER is ≥ 26% 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 ≥ 26% RTP and < 60% rated recirculation drive flow), then the affected channel is considered inoperable.~~

~~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.

Add the following references:

17. GE Hitachi Nuclear Energy, "GE Hitachi Boiling Water Reactor, Detect And Suppress Solution - Confirmation Density," NEDC-33075P, Revision 7, June 2011.
18. Anthony J. Mendiola (NRC) to Jerald G. Head (GEH), "Revised Draft Safety Evaluation for GE-Hitachi Nuclear Energy Americas, LLC Topical Report NEDC-33075P, Revision 7, 'GE Hitachi Boiling Water Reactor Detect and Suppress Solution - Confirmation Density' (TAC No. ME6577)," dated August 6, 2013.

BASES

APPLICABLE SAFETY ANALYSES (continued)

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

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

TS Bases Insert 7

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

Recirculation loops operating satisfies Criterion 2 of Reference 5.

LCO

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

(continued)

TS Bases Insert 7

The Maximum Extended Load Line Limit Analysis (MELLLA) operating domain and the MELLLA Plus (MELLLA+) operating domain are not analyzed for single recirculation loop operation. Therefore, intentional single loop operation is prohibited in the MELLLA operating domain and the MELLLA+ operating domain as defined in the COLR (Ref. 6).

BASES

The LCO prohibits intentional operation with a single recirculation pump in operation while in the MELLLA operating domain or MELLLA+ operating domain as defined in the COLR.

LCO (continued)

must be applied to allow continued operation consistent with the assumptions of Reference 4.

APPLICABILITY

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

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

ACTIONS

A.1 and A.2

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

, B.2,

B.1 and C.1

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

(continued)

BASES

ACTIONS

B.1 and C.1 (continued)

, B.2,

or a transient results in operation in the MELLLA or MELLLA+ operating domain with only a single loop in operation

With the requirements of the LCO not met for reasons other than Condition A and B (e.g. one loop is "not in operation"), the recirculation loops ~~must~~ be restored to operation with matched flows within 4 hours. A recirculation loop is considered not in operation when the pump in that loop is idle or when the mismatch between total jet pump flows of the two loops is greater than required limits for greater than 2 hours (i.e. Required Action B.1 has been taken). Should a LOCA occur with one recirculation loop not in operation, the core flow coastdown and resultant core response may not be bounded by the LOCA analyses. Therefore, only a limited time is allowed to restore the inoperable loop to operating status.

to

Required Action C.1 requires

Required Action B.2 identifies that operation in the MELLLA domain or the MELLLA+ domain as defined in the COLR is prohibited, when a recirculation pump is declared to be not in operation due to a recirculation loop flow mismatch.

Alternatively, if the single loop requirements of the LCO are applied to the APLHGR and MCPR operating limits and RPS and RBM Allowable Values, operation with only one recirculation loop would satisfy the requirements of the LCO and the initial conditions of the accident sequence.

The 2 hour and 4 hour Completion Times are based on the low probability of an accident occurring during this time period, on a reasonable time to complete the Required Action, and on frequent core monitoring by operators allowing abrupt changes in core flow conditions to be quickly detected.

D.1

If the Required Action and associated Completion Time of Condition C is not met, the unit is required to be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to MODE 3 within 12 hours. In this condition, the recirculation loops are not required to be operating because of the reduced severity of DBAs and minimal dependence on the recirculation loop coastdown characteristics. The allowed Completion Time of 12 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.4.1.1

This SR ensures the recirculation loop flows are within the allowable limits for mismatch. At low core flow (i.e., effective core flow < 70% of rated core flow), the APLHGR and MCPR requirements provide larger margins to the fuel cladding integrity Safety Limit such that the potential adverse effect of early boiling transition during a LOCA is reduced. A larger flow mismatch can therefore be allowed when effective core flow is < 70% of rated core flow. The jet pump loop flow, as used in this Surveillance, is the summation of the flows from all of the jet pumps associated with a single recirculation loop. The effective core flow shall be calculated by assuming both loops are at the smaller value of the two jet pump loop flows.

The mismatch is measured in terms of percent of rated core flow. If the flow mismatch exceeds the specified limits, the loop with the lower flow is considered not in operation. This SR is not required when both loops are not in operation since the mismatch limits are meaningless during single loop or natural circulation operation. The Surveillance must be performed within 24 hours after both loops are in operation. The 24 hour Frequency is consistent with the Frequency for jet pump OPERABILITY verification and has been shown by operating experience to be adequate to detect off normal jet pump loop flows in a timely manner.

REFERENCES

1. USAR, Section 6.3 and Appendix A Section 6.

2. USAR, Section 6.3.3.7.

3. USAR, Section 5.4.1.3.

4. USAR Chapter 15B.

5. 10 CFR 50.36(c)(2)(ii).

6. NEDC-33006P-A, "Maximum Extended Load Line Limit Analysis Plus Licensing Topical Report," Revision 3, June 2009.

ATTACHMENT 3

LIST OF REGULATORY COMMITMENTS

The following table is a list of regulatory commitments made in this letter. Any other statements in this submittal represent intended or planned actions. They are provided for information purposes and are not considered to be regulatory commitments.

Regulatory Commitments	Scheduled Completion Date
1) The fuel and cycle-dependent analyses, including the plant-specific thermal limits assessment, will be submitted for NRC staff confirmation by supplementing the initial MELLLA+ Safety Analysis Report (SAR) in accordance with Limitation and Condition 12.4 of the MELLLA+ Licensing Topical Report (LTR) Safety Evaluation Report (SER). Specifically, CENG will provide the cycle specific Supplemental Reload Licensing Report (SRLR) and Fuel Bundle Information Report (FBIR), which includes the supplemental information to satisfy MELLLA+ LTR SER Limitation and Condition 12.4. 2) Nine Mile Point Nuclear Station, LLC (NMPNS) will provide a cycle-specific core design loading map along with a summary of differences between the reference design described in the M+SAR and the cycle-specific core design. This summary will include differences in the energy requirements, average enrichment, and analytical inputs, a cycle-specific thermal limits assessment, and the actual reload analysis results. Additionally, the Supplemental Reload Licensing Report, which includes the cycle specific core map, will be provided. Submittal of the cycle-specific design will satisfy the NRC request made at the MELLLA+ LAR pre-meeting on March 13, 2013.	February 28, 2014
If NMP2 is the first plant-specific implementation of MELLLA+, then the cycle-specific eigenvalue tracking data will be evaluated and submitted to NRC to establish the performance of nuclear methods under the operation of the new operating domain. The following data will be analyzed: • Hot critical eigenvalue, • Cold critical eigenvalue, • Nodal power distribution (measured and calculated traversing in-core probe (TIP) comparison), • Bundle power distribution (measured and calculated TIP comparison), • Thermal margin, • Core flow and pressure drop uncertainties, and • The MCPR Importance Parameter (MIP) Criterion (i.e., determine if core and fuel design selected is expected to produce a plant response outside the prior experience base).	June 30, 2016

ATTACHMENT 4

MELLLA+ RISK EVALUATION

Nine Mile Point Nuclear Station, LLC
November 1, 2013

1 Introduction

The ability to operate in the Maximum Extended Load Line Limit Analysis Plus (MELLLA+) region of the power/flow map provides significant benefits to NMP2 operators by increasing the fuel capacity factor during the operating cycle and allowing greater flexibility in using flow adjustments to control reactivity. The purpose of this evaluation is to identify changes to the NMP2 Probabilistic Risk Assessment (PRA) model as a result of the MELLLA+ enhanced operating range and estimate the resulting risk impact from these changes. The baseline PRA used in this evaluation already includes consideration of Extended Power Uprate (EPU) modifications and impacts. In addition, the MELLLA+ change does not require major plant hardware modifications and no increases in operating pressure, power, steam flow rate and feedwater flow rate. Changes are limited to the revised power/core flow map, a small number of instrument setpoints, implementation of DSS-CD (detect and suppress solution-confirmatory density) solution software and potential increase in steam moisture content at certain times.

The NMP2 PRA (Reference 1) underwent an Internal Events and Internal Flooding industry peer review in August 2009 utilizing ASME/ANS RA-Sa-2009 (Reference 3) and Regulatory Guide 1.200 Rev 2 (Reference 2). Subsequent post peer review updates to the PRA have resolved most observations as well as incorporated the impact of EPU. The remaining open observations were reviewed with respect to this application and found to have negligible risk impact.

Besides including EPU, the NMP2 PRA scope also includes the results of the IPEEE for fires and seismic initiating events at power. The quality of the External Events modeling has not been peer reviewed to Regulatory Guide 1.200 Rev 2 and ASME/ANS RA-Sa-2009; however the quality was found acceptable for the NMP2 EDG AOT application approved by NRC in 2011 (Reference 4). Also, the risk impact associated with the MELLLA+ operating range on external events risk is minimal as described later in this evaluation.

2 Methodology

This evaluation is conducted consistent with the approved guidance documents NEDC-33006P-A, NEDC-32424P-A, NEDC-32523P-A, and NEDC-33004P-A (References 5 through 8) and Regulatory Guide 1.174 (Reference 9). In addition, Matrix 13 of the NRC Standard Review Plan for EPU (RS-001) is used as a guide for the MELLLA+ risk assessment (Reference 10).

The methodology can be summarized as follows:

- Changes to the plant configuration, operating margins, and safety and design margins as a result of MELLLA+ are evaluated in several MELLLA+ task reports. These reports are first reviewed with respect to any potential impact on the PRA (see Section 3.1). This also provides the technical baseline for further evaluation of PRA Elements in the next step.

- Each PRA Element in ASME/ANS RA-Sa-2009 (PRA technical elements) is assessed based on the plant changes identified in the first step considering any potential impact to the PRA due to MELLLA+ (see Section 3.2).

- PRA changes identified from the above are summarized and the risk impact estimated with respect to NMP2 compliance with Regulatory Guide 1.174 (see Section 3.3).

3 Evaluation

This section summarizes plant changes due to MELLLA+ and potential impacts on the PRA. First, the evaluation considers the task report evaluations (Section 3.1.1) that are required to support the safety evaluation and then the results of the review are summarized by major plant change category (Section 3.1.2). Then, the evaluation is conducted by questioning whether MELLLA+ impacts the PRA Model (Section 3.2) by considering the PRA technical elements as well as external events and shutdown operation. Section 3.3 summarizes the PRA model changes and the quantitative estimates of risk impact.

3.1 Plant Change Evaluation

3.1.1 Summary of MELLLA+ Engineering Evaluations

The reference 11 and supporting engineering evaluations (i.e., Task reports) necessary to support the extension of the NMP2 operating domain to permit operating up to 120% of original licensed thermal power at core flows as low as 85% of rated (known as maximum extended load line limit analysis plus [MELLLA+]) have been reviewed. In summary, MELLLA+ operation affects the core and some aspects of the NSSS, but it does not change thermal power, normal operating pressure, steam flow, feedwater flow, or feedwater temperature. The power conversion systems, electrical systems, and other auxiliary systems are not changed as a result of MELLLA+ operation. The following types of changes were identified with respect to potential impact on the PRA model:

(1) increased ATWS power impact on time available for operator actions

(2) increased ATWS power impact on the probability of a stuck open SRV

(3) plant trip frequency impact from changes to software and setpoints

(4) plant trip frequency impact due to the potential for loss of a single feedwater pump

(5) plant trip frequency impact due to moisture carryover impact on main condenser

3.1.2 Plant Change Summary

This section summarizes the review of plant changes by major plant change category and potential impacts on the PRA based on the detailed Tasks Reports evaluated above.

<u>Hardware Modification</u>
There are no important hardware modifications for MELLLA+ and none of the systems in the NMP2 PRA require hardware modification.

The MELLLA+ reactor operating domain requires an update to the plant software configuration, including the process computer and applicable operating procedures. The DSS-CD (detect and suppress solution – confirmation density) stability solution implementation, including setpoints,

technical specification changes and operating procedures impact the plant configuration, however, there is no direct impact on design operating margin. These changes have no impact on the PRA models.

Although an increase in plant trips is not anticipated because a primary objective of plant change is to prevent such an event, past industry experience indicates that plant changes (in this case hardware, software and trip set points) can lead to an unplanned trip regardless of intentions to preclude an event. Thus, a sensitivity analysis will be performed to investigate this potential increase in risk.

Procedure Changes
No changes to the NMP2 EOPs, Severe Accident Management Guidelines (SAMGs) or Special Operating Procedures are required for MELLLA+. Therefore there is no direct impact on the PRA model or human reliability analysis as a result of procedure changes due to MELLLA+.

Changes will be needed for procedures, training documents, the process computer, control room displays, and simulator related APRM setpoints discussed below and hardware changes described above. However, there is no direct impact on the PRA models or human reliability analysis as result of these changes.

Setpoint Changes
New allowable values and nominal trip setpoints (NTSP) have been addressed for APRM flow biased simulated thermal power upscale scram and rod block during MELLLA+ operation. The values for rod block monitor setpoints in the technical specifications do not change. Instrumentation for the changed setpoint functions need to be recalibrated for revised NTSPs and changes will be needed for associated procedures, training, process computer, control room simulator, and control room displays. None of these changes impact the PRA except for a potential increased likelihood of a plant trip due to new equipment and set point changes. Thus, as described under hardware changes, a sensitivity analysis will be performed to investigate this potential increase in risk.

Operating Conditions
No configuration or operational changes are required of MELLLA+ that have a direct impact on the PRA. Reactor thermal power, operating pressure, steam flow, and feedwater flow are not changed by MELLLA+. Also, MELLLA+ does not change the operating conditions of systems modeled in the PRA.

Core instability issues were evaluated; the DSS-CD modification ensures that operating and safety margins are maintained for both normal operation and ATWS conditions. No impact on the PRA was identified.

Single reactor recirculation pump operation during MELLLA+ is not allowed.

Conditions during an ATWS are potentially more severe post reactor recirculation pump trip (RPT) during MELLLA+ operation. At reduced flow rates with power at 120%, the post RPT power level may be slightly higher during MELLLA+ operation. This would potentially reduce

the time for operator response and this requires evaluation for PRA impact. RPV water level would potentially drop quicker and containment heat up would initially be quicker if power level is a little higher. However, the increased SLCS boron 10 enrichment ensures that the integrated heat-up of the suppression pool is unchanged. In addition, the potential increased probability of a stuck open SRV during an ATWS event is evaluated as there could be additional challenges and such challenges could last a little longer.

3.2 PRA Element Evaluation

The previous section evaluated and identified MELLLA+ changes to the plant (hardware, procedures, setpoints and operation) and potential impact to the PRA. This section performs a separate evaluation by PRA Technical Element to investigate NMP2 PRA Model changes. This provides another assessment for potential PRA impacts due to MELLLA+ as well as a more detailed assessment of the PRA change. The evaluation is conducted by PRA Technical Element in the following subsections.

3.2.1 Initiating Events (IE)

The NMP2 PRA includes internal events such as transients, LOCAs and internal floods and other hazards such as internal fire and seismic initiating events. Each of these as well as shutdown events are considered below:

Transient

The changes described in the previous section for MELLLA+ do not result in any new initiating events. The changes are not extensive and of the type that could result in a new initiating event. The power conversion systems, electrical systems, and other auxiliary systems are not changed as a result of MELLLA+ operation except for some potential effects of increased moisture content in the steam. MELLLA+ operation affects the core and some aspects of the NSSS, but it does not change thermal power, normal operating pressure, steam flow, feedwater flow, or feedwater temperature. Although a recirculation pump trip or feedwater pump trip could result in a reactor trip, this frequency is low and subsumed by the total reactor trip partial loss of feedwater initiating event frequency. It could be postulated that a recirculation pump trip while in certain MELLLA+ regions could result in instabilities that cause an unwanted scram from the new DSS-CD scram software. However, this frequency is small in comparison to the total frequency of plant trip in the PRA. The limited plant changes (computer, software, procedures, setpoints etc) are not expected to result in an impact on initiating event frequencies, as the intent of plant change is to prevent such events. As with all plant changes, if the future indicates an increase plant trip frequency it will be included in PRA updates. However, a sensitivity analyses will be performed to consider the potential risk increase associated with MELLLA+ implementation.

LOCA

No changes are made to the reactor coolant system piping or piping in other systems connected to the reactor coolant system, including the piping inspection programs. In addition, MELLLA+ does not involve any changes to reactor operating temperature and pressure or feedwater flow.

Increased moisture in the steam is not judged to represent a significant impact to secondary system steam piping. Therefore, there are no impacts on LOCA frequency.

Internal Flood
No changes are being made to piping systems in the plant, their flow characteristics or their inspection programs. Therefore, there is no impact on internal flood initiating event frequency due to MELLLA+.

Fire, Seismic, and Other External Hazards
The frequency of these hazards is not affected by MELLLA+. Therefore, there is no impact on these initiating event frequencies.

Shutdown Events
MELLLA+ has no impact on shutdown risk. There is no impact on operator actions, procedures, outage activities, initiating events, success criteria and systems credited in the accident response to shutdown events. Post plant trip the plant looks the same as before MELLLA+.

3.2.2 Accident Sequence Analysis (AS)

MELLLA+ condition does not change the plant configuration and operation such that new accident sequences or changes to existing accident sequences occur. There could be a slight change in allowed operator response time (reduction) for ATWS sequences, which is addressed under human reliability below, but the accident sequence model itself is not changed. The increased boron enrichment for SLCS is a potential positive change that could improve the present ATWS model with regard to taking more credit for operators preventing HCTL and keeping the MSIVs open. This can be considered in future updates to the PRA.

3.2.3 Success Criteria (SC)

Reactor thermal power, operating pressure, steam flow, and feedwater flow are not changed by MELLLA+. The power conversion systems, electrical systems, and other auxiliary systems are not changed as a result of MELLLA+ operation. Also, MELLLA+ does not change the operating conditions of systems modeled in the PRA. There is no impact on the success criteria provided for the critical safety functions in the PRA; reactivity control, pressure control, inventory control at high pressure, emergency depressurization, inventory control at low pressure and containment heat removal; the following summarizes:

- **Reactivity Control** – the number of control rods and RPS success criteria is unchanged. One of two SLCS pumps as a successful alternate shutdown system is unchanged and in fact this success criterion is further supported by the MELLLA+ evaluations and the increase in boron10 enrichment. Although MELLLA+ has no impact on the probability of scram failure, the plant may be at a slightly higher power during ATWS until SLCS is injected. This can affect the timing of operator response as described in Section 3.2.5.

- **Pressure Control (RPV overpressure protection)** – there is no impact on the number of SRVs required for success. RPV dome operating pressure is not increased and there is no impact for non-ATWS events. The higher power condition during ATWS was evaluated and the assumed success criteria (16 of 18 SRVs required) in the PRA is still met with MELLLA+ conditions. GEH analysis indicates margin in over pressure protection with two SRVs out of service; therefore the probability of over pressure due to failure of several SRVs is still dominated by common cause failure of the SRVs, which is unchanged in the PRA.

- **Pressure Control (SRVs reclose)** – the success criteria is that all SRV reclose, which is unchanged. There is no impact on number of SRV challenges for non-ATWS events as operating pressure and power is not changed. However, the SRVs are likely open for a longer time during ATWS due to higher initial power level. The NMP2 turbine bypass is rated at approximately 18.5% of rated steam flow. Thus, until power level is reduced to the equivalent bypass flow rate, SRVs will be open. In the case of a more severe transient such as closure of all MSIVs, SRVs will be open until SLCS is injected; however, the increased boron 10 enrichment ensures that the time to reactor shutdown is not increased due to MELLLA+. The potential increase in probability of a stuck open SRV in the ATWS model is considered with regard to PRA model change (i.e., see Section 3.3, PRA Model Changes & Resulting Risk Impact).

- **High Pressure Injection** – No change in the number of pumps required for success. The MELLLA+ plant changes do not result in changes to injection systems and reactor power & pressure are unchanged. Thus, there is no impact on injection system success criteria for non-ATWS events. The potential for higher power level during ATWS until SLCS injection does not impact the systems credited for initial level control. The timing associated with operator response is evaluated (see Section 3.2.5, Human Reliability, and Section 3.3, PRA Model Changes & Resulting Risk Impact).

- **Emergency Depressurization** – No change in the number of SRVs required supporting low pressure injection success. MELLLA+ does not involve changes to the ADS and does not change reactor power or pressure. Although ATWS power is potentially higher until SLCS is injected, there is no impact on success criteria. However, timing associated with operator response during ATWS is evaluated (see Section 3.2.5, Human Reliability, and Section 3.3, PRA Model Changes & Resulting Risk Impact).

- **Low Pressure Injection** – No change in the systems and number of pumps required for success. The MELLLA+ plant changes do not result in changes to injection systems and reactor power & pressure are unchanged. Thus, there is no impact on injection system success criteria for non-ATWS events. The potential for slightly higher power level during initial stages of ATWS does not impact the systems credited for level control after emergency depressurization and during SLCS injection.

- **Containment Heat Removal** – No change to the systems and success criteria for this function. Plant changes for MELLLA+ do not result in changes to containment heat removal systems and reactor power & pressure are unchanged. Thus, there is no impact

on heat removal success criteria for non-ATWS events. Also, for mitigated ATWS events (SLCS injection), the RHR success criteria are unchanged. The potential reduction in time to align RHR is considered in Section 3.2.5. Although the suppression pool heat-up could be initially faster during ATWS due to potentially higher power level, the SLCS increased boron 10 enrichment maintains the integrated containment heat up unchanged.

- **Containment Response** – containment analysis for LOCA and ATWS under MELLLA+ conditions indicate the dynamic loads and containment conditions remain acceptable. No impact on the PRA was identified (see also Section 3.2.7 "Level 2 Model").

3.2.4 Systems Analysis (SY)

MELLLA+ does not add new equipment that would be modeled in the system fault trees. There is also no need to change existing fault trees or any system alignments. No changes were required to the SRV or SLCS system logic. The number of SRVs required for success was unchanged. The increase in SLCS boron 10 enrichment and related analyses provide additional bases and margin for the present SLCS model (1 of 2 pumps as a success path for alternate shutdown).

3.2.5 Human Reliability (HR)

There are no changes required to the NMP2 EOPs, SAMGs or Special Operating Procedures for MELLLA+. Also, there are no new operator actions as a result of MELLLA+. There is no change in sensible and decay heat in the MELLLA+ operating domain. There also is no change to the instruments and cues in the control room that are used in the human reliability analysis. Therefore there is no direct impact on the PRA except for ATWS where the power level could be slightly higher after automatic reactor recirculation pump trip until SLCS is injected. At NMP2, feedwater runback is automatic, which helps to control power immediately after an ATWS. NMP2 also has an automatic initiation of SLCS ensuring that the reactor will be shutdown without the need for operator action to be modeled. The following summarizes the operator actions in ATWS accident sequence model that require evaluation for MELLLA+:

Table 1: ATWS Model Human Actions

Action	Description	HEP	Accident Sequence Impact
ZFW01	Restore feedwater before Level 1	0.5	Failure of either action results in MSIV isolation at Level 1 and loss of the main condenser, which impacts suppression pool heat-up. Although conservative, this modeling guarantees loss of the main condenser in the NMP2 ATWS model.
ZMO01	Disable Level 1 MSIV Isolation	0.5	
ZMS01	Put Mode Switch in SHUTDOWN	1.2E-3	Failure results in loss of main condenser when pressure <850 psig
ZAI01	ADS Inhibit	6.4E-3	Failure is assumed to result in core damage due to vessel flooding, power excursion and flushing boron out of the vessel
ZEPC1	Secure HPCS	2.4E-2	Failure of any one of these actions results in HCTL being exceeded with RPV blowdown. ZOH02 only applies to the
ZEPC2	Control FW and RPV level	6.8E-2	

Action	Description	HEP	Accident Sequence Impact
ZOH02	Align both trains of suppression pool cooling early to prevent HCTL	0.98	case where the main condenser is unavailable; it is set to 1.0 because there was inadequate analysis to justify 2 trains of RHR and SLCS preventing HCTL without the main condenser available. The more recent MELLLA+ evaluation appears to provide a basis for the future.
ZOD03	RPV Blowdown for LPI (loss of FW)	1.2E-2	Failure is assumed to result in core damage due to core uncover at high pressure without adequate high pressure make-up
ZCH01	Terminate & Prevent Overfilling	3.9E-2	Failure is assumed to result in core damage due to vessel flooding, power excursion and flushing boron out of the vessel. ZCH01 applies to all sequences and ZCH02 only to loss of FW.
ZCH02	Slowly Raise Level with LPI (loss of FW)	3.4E-2	

Other operator actions in the ATWS model that apply given ATWS mitigation success such as long term containment pressure control are not included because these actions are similar to a transient and only modeled for the case of ATWS mitigation success (e.g., SLCS success).

3.2.6 Data Analysis (DA)

MELLLA+ does not result in any changes to plant equipment that would result in component failure rates or unavailability to change as modeled in the PRA.

3.2.7 Level 2 Model (LE)

The Level 2 accident sequence modeling is not impacted by MELLLA+. The following summarizes:

- **Containment**: MELLLA+ does not result in any change to the containment structure or containment isolation system. In addition, containment challenges and containment capability is not affected.

- **Accident Sequence Model**: there are no changes to plant configuration and operation that could result in new accident sequences and there is no accident sequence modeling or success criteria changes due to MELLLA+.

- **Human Reliability**: although MELLLA+ is postulated to potentially result in higher ATWS power and a potential reduction in time available for operator response, there is no impact on the Level 2 model since the operator action times are defined relative to core damage and core melt progression timing. This timing is later and not directly related to ATWS timing.

- **Releases**: MELLLA+ has no impact on the PRA radionuclide release categorization, magnitudes or timing. Changes in design basis source terms are minor. Although the timing of ATWS scenarios may have a minor impact, this timing reduction has no impact on the release timing categorization of ATWS severe accidents because all ATWS releases are already assigned the earliest release categorization in the PRA.

The only change is the Level 2 quantification due to changes in the core damage accident sequence model.

3.2.8 Quantification (QU)

No changes to the NMP2 PRA quantification process will result from MELLLA+. Small changes in the quantification results are described in Section 3.3 as a result of ATWS HEP changes and sensitivity analysis for potential changes.

3.2.9 Shutdown Operation

MELLLA+ has no impact on shutdown risk for similar reasons described for power operation (modes 1 and 2) above. There is no impact on initiating events during shutdown and no impact on success criteria or systems or data or human reliability. Post plant trip there is no change to plant response due to MELLLA+. There is also no impact on outage scheduling or procedures.

3.3 PRA Model Changes & Resulting Risk Impact

3.3.1 PRA Model Changes

The evaluations in the previous sections, which are based on detailed evaluations (task reports) in accordance with References 5 through 8, concluded there are two types of change to the NMP2 PRA:

1. Potential change – sensitivity analysis considered since PRA model is not changed

    a. Potential transient initiating event frequency increase due to software and setpoint changes. This type of speculative change is not included in the PRA as a result of MELLLA+ because it is not planned and the intent is to prevent such impacts. Future PRA updates will account for any change in the transient initiating event frequency, which could be lower or higher. However, since past experience has shown that an unforeseen transient could occur due to these types of changes, a sensitivity analysis is provided to show that this potential risk is low.

    b. Potential for scram initiating event frequency increase due to loss of a single feedwater pump. This is a partial loss of feedwater initiating event in the PRA. This type of speculative change is not included in the PRA as a result of MELLLA+ because it is already included as potential initiator and the frequency is not judged to be changed. Future PRA updates will account for any change in the transient initiating event frequency, which could be lower or higher. However, a sensitivity analysis is provided to show that this potential risk is low.

c. Potential for loss of main condenser initiating event frequency increase due to increased moisture content in steam (e.g., SJAE performance). This event is modeled in the PRA. This type of speculative change is not included in the PRA as a result of MELLLA+ because it is already included as potential initiator and the frequency is not judged to be changed. Future PRA updates will account for any change in the transient initiating event frequency, which could be lower or higher. However, a sensitivity analysis is provided to show that this potential risk is low.

d. Potential increase in the probability of stuck open SRV during ATWS due to a potential initial increase in power. This is evaluated as a potential change not an actual change for a number of reasons as summarized below:

- A comparison between the EPU ATWS analysis and the MELLLA+ ATWS analysis indicates that the integrated SRV flow is about the same. For example, a comparison with MSIVC (all MSIVs close initiating event) for the end of cycle indicated the integrated flow is similar for the first minute, MELLLA+ is slightly higher between 1 and 5 minutes and then MELLLA+ integrated flow is less than EPU from 5 to 10 minutes due to the higher boron enrichment which results in slightly quicker shutdown (confirmed by neutron flux comparison). See Figure 1 below.

- The impact of a stuck open SRV is relatively minor. RCIC is not credited as an injection system with a stuck open SRV, but RCIC is already not credited in the NMP2 ATWS model. Also, a stuck open SRV would eventually lead to MSIV closure; the NMP2 PRA models a stuck open SRV as a contributor to loss of main condenser making it more difficult to prevent HCTL and requiring an eventual emergency RPV blowdown. This does not result in core damage, but does provide additional challenges to the operators and success criteria for preventing HCTL.

- Based on the present NMP2 PRA ATWS Model, an increased probability of a stuck open SRV has no impact because of conservative modeling. The main condenser is already assumed to be unavailable based on the assumed reliability of operator actions to prevent MSIV closure on Level 1 (see operator actions ZFW01 and ZMO01). Also, without the main condenser available the model does not credit the operators preventing HCTL with both trains of RHR suppression pool cooling and both trains of SLCS (see operator action ZOH02). This is not an operator reliability issue as there was not sufficient analysis at the time to show crediting both trains of RHR and SLCS as preventing HCTL; setting operator to failure is a convenient method for not taking this credit for the time being.

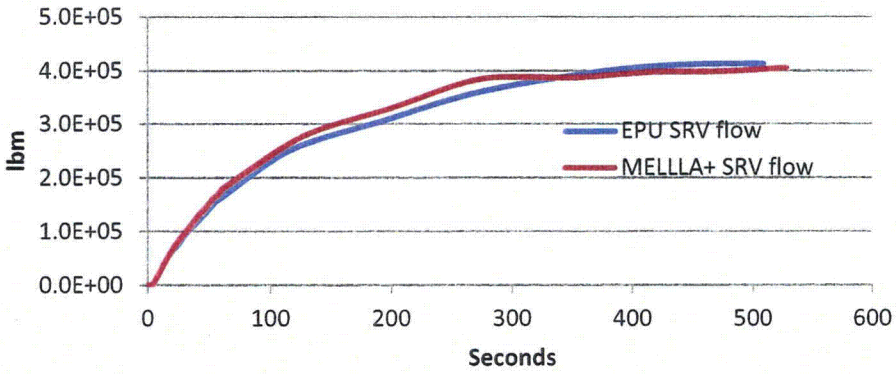
- Increased probability of stuck open SRV is not significant for a number of reasons: (1) duration that SRVs are open is not significantly increased as a result of increased boron 10 enrichment and (2) additional SRV cycles should also be insignificant for the same reason and once the valve closes successfully it is more likely to succeed the second or third time. Additional SRVs may be challenged

initially due to potentially higher power during a portion of the transient; however, the peak RPV pressure calculated for MELLLA+ is not increased so this may not be significant for the MSIV closure event. For a less significant transient such as turbine trip with turbine bypass available (~18.5% of rated steam flow), the increase in power could result in additional SRVs opening. Thus, the probability of a stuck open SRV is likely dominated by the increased number of SRVs that open. The probability that an SRV fails to close in the PRA is 4.6E-3 (e.g., basic event MSSVPZXPSV120XB1) and each SRV flow rate is equivalent to ~5.7% of power (i.e., USAR Table 15.0-3a with 2 of 18 SRVs assumed failed; 91%/16).

In summary, the potential risk increase due to increased probability of stuck open SRV during an ATWS is not easy to calculate because of the conservative modeling would presently indicate 0.0 impact. Also, the increased probability of a stuck open SRV is not significant and it would not be a significant event as it does not lead directly to core damage. In order to show this quantitatively, the ATWS contribution to CDF and LERF for the representative sequences that would be affected by a stuck open SRV is summed recognizing that the increase in risk would be less than these results (a 10% increase is assumed to be bounding).

2. Real change – ATWS human reliability impacts due to slightly higher power post recirculation pump trip must be taken into consideration in the PRA. As described above, the integrated SRV flow MSIVC is similar to EPU for the first minute and then MELLLA+ shows an increase until about 5 minutes into the transient; as expected neutron flux response is similar where neutron flux for MELLLA+ appears slightly higher until 5 minutes into the transient then the higher boron enrichment for MELLLA+ results in a lower neutron flux for MELLLA+. This slight increase in power during the first 5 minutes post trip (MSIVC) results in more power being transferred to the suppression pool at least initially and RPV level will drop quicker requiring operators to restore feedwater level potentially faster. Each operator action is evaluated below for the MELLLA+ condition to support a PRA calculation with these updated HEPs.

Figure 1: Integrated SRV Flow



Potential Change – Sensitivity Analysis on Plant Trip Frequency
An additional reactor trip is assumed to occur in the first year after MELLLA+ implementation. Although no such increase is expected, this potential increase in risk is evaluated to show the low risk of such an event in case such an event does occur. In the NMP2 PRA, the most recent 10 years of plant data is used in the Bayesian analysis as being representative. Thus, the change in long term average plant trip initiating event frequency is 0.1/yr.

Potential Change – Sensitivity Analysis on Partial Loss of Feedwater Trip Frequency
Since MELLLA+ has no impact on the feedwater pumps and their likelihood of failure, this sensitivity must account for the feedwater pump unchanged failure probability contribution to the partial loss of feedwater initiating event (PLOF) and then the increased probability of a scram due to MELLLA+. Based on the analysis performed for EPU, this PLOF initiating event is modeled conservatively because it always assumes that a scram occurs. Thus, based on the conservative modeling in the PRA, there is no impact to calculate. Instead, the present contribution of PLOF is provided to show that even if there was a small increase it would be insignificant.

Potential Change – Sensitivity Analysis on Loss of Main Condenser Frequency
An additional loss of condenser is assumed to occur within a few years after MELLLA+ implementation. Although no such increase is expected, this potential increase in risk is evaluated to show the low risk of such an event in case such an event does occur. In the NMP2 PRA, the most recent 10 years of plant data is used in the Bayesian analysis as being representative. Thus, the change in long term average plant trip initiating event frequency is 0.1/yr.

Potential Change – Sensitivity Analysis on Stuck Open SRV during ATWS
As described above, the CDF and LERF contribution from ATWS event tree top event EPC failures (sequences ATWS-006, 007, 014 and 015) is quantified because these are the sequences that are affected by a stuck open SRV. Since these sequences are presently modeled conservatively, these results provide an upper bound for estimating any risk increase. That is, it assumes that first the risk of these sequences is reduced significantly and then a stuck open SRV results in a risk increase to the present risk level. This result is multiplied by 10% to account for the increased probability of a stuck open SRV due to MELLLA+.

ATWS HEP Changes
The steam flow due to ATWS power levels drives the amount of RPV level reduction which is a key factor in ATWS HEPs. The integrated inventory loss driven by ATWS power can be seen in Figure 1 which shows the increased steam flow for MELLLA+ conditions versus non-MELLLA+ conditions for one specific ATWS scenario considered in the PRA (MSIV closure failure to SCRAM). Note that steam flow is similar for MELLLA+ and non-MELLLA+ over the first 60 seconds or so. From 1 to 2 minutes, the integrated steam flow for the MELLLA+ case is approximately 5% higher (~2.6E5 lbm versus ~2.7E5 lbm). At about 4 minutes, it is approximately 10% higher (~3.2E5 lbm versus ~3.5E5 lbm). Note that after 6 minutes, the power and steam mass decrease for the MELLLA+ conditions. This is attributable to the increased power reduction associated with the enhanced Boron enrichment associated with the MELLLA+ improvement. The reduced available action time for operator actions would be

proportional to the inventory lost due to the increase in ATWS power associated with MELLLA+ conditions. Baseline Human Error Probabilities (HEPs) and timing are derived in the NMP2 Human Reliability Analysis (HRA) notebook. Appendix B of this document provides a summary of simulator walkdowns. For MELLLA+ evaluation, impacts of the proposed change are considered in relation to the baseline HEP values. The human actions summarized in Table 1 (Section 3.2.5) are evaluated below with regard to new HEP values for MELLLA+:

- ZFW01 (restore feedwater before Level 1, 0.5): level 1 could be reached slightly faster given MELLLA+ configuration. According to MAAP run U2AT1PA, RPV Level 1 would be reached within 2 minutes of a full-power ATWS given pre-EPU as well as post EPU conditions. The ATWS Feedwater runback is locked-out for ~55 seconds (RRCS Feedwater Runback occurs 25 seconds after RPV high pressure (ATWS) signal and runback is locked-in for 30 seconds) so operators have a fairly small time window for action. Since operators are expected to prevent/secure HPCS upon ATWS EOP entry, HPI will be limited to CRD and RCIC. The action requires between 15-30 seconds. 0.5 was assigned for post-EPU conditions and this is deemed reasonable for MELLLA+ conditions as well since the impact is judged to be on the order of seconds and is within the uncertainty of all inputs in such a time-sensitive action. Also, combined with the 0.5 assigned for ZMO01, below, the impact is that for all full-power ATWS events, the PRA is assuming the operators will not be able to maintain the MSIVs open. Thus, this action cannot be penalized any further.

- ZMO01 (disable Level 1 MSIV Isolation, 0.5): level 1 could be reached slightly faster given MELLLA+ configuration. According to MAAP run U2AT1PA, RPV Level 1 would be reached within 2 minutes of a full-power ATWS given pre-EPU as well as post EPU conditions. The action requires between 45-60 seconds. 0.5 was assigned for post-EPU conditions and this is deemed reasonable for MELLLA+ conditions as well since the impact is judged to be on the order of seconds and is within the uncertainty of all inputs in such a time-sensitive action. Also, combined with the 0.5 assigned for ZFW01, above, the impact is that for all full-power ATWS events, the PRA is assuming the operators will not be able to maintain the MSIVs open. Thus, this action cannot be penalized any further.

- ZMS01 (put mode switch in SHUTDOWN, 1.2E-3): given success of ZFW01 and ZMO01, this action is dependent on pressure not RPV level and would not be impacted by MELLLA+.

- ZAI01 (ADS Inhibit, 6.4E-3): this is an immediate response action and not judged to be significantly impacted by MELLLA+. According to MAAP run U2AT1PA, RPV Level 1 would be reached within 2 minutes of a full-power ATWS given pre-EPU as well as post EPU conditions. The action requires less than 30 seconds. Also, the ADS timer is 120 seconds and this gives operators a secondary cue (i.e., in addition to ATWS EOP entry) for the action as well as significantly more time for the action. While level 1 may be reached slightly more quickly in MELLLA+ configurations, the 120 second ADS time delay is unaffected. Since the time for this action is driven primarily by the time to reach RPV level 1 (plus the 2 minute time delay) and reaching level 1 occurs within the first 2

minutes, the time window is taken as 5% reduced, proportionate with the MELLLA+ increased power over the first 2 minutes. Thus, the 4 minute time window for this action has been reduced by 5% to 228 seconds. Changing this system window in the HRA calculation for this action increases the failure probability from 6.4E-3 to 8.4E-3. As sensitivity, the 4 minute time window for this action is reduced by 10% to 216 seconds. For the 10% sensitivity, the probability for this action is increased from 6.4E-3 to 1.2E-2.

- ZEPC1 (secure HPCS, 2.4E-2): RPV level 2 (HPCS Initiation) is reached within about 1 minute. HPCS initiation can cause significant positive reactivity after a period of operation. This period has not been developed in detail but 2 minutes is judged a survivable duration. Thus, 3 minutes was judged a reasonable system window for this action. This action is directed immediately upon entry to the ATWS EOP. Thus, it is less time-critical than other ATWS response actions. Also, HPCS can run for a small period of time without significantly impacting level or power. However, this time is difficult to estimate and determining the impact of MELLLA+ conditions within this uncertainty is difficult. Since the time for this action is driven primarily by the time to reach RPV level 2 and reaching level 2 occurs within the first 2 minutes, the time window is taken as 5% reduced, proportionate with the MELLLA+ increased steam flow over the first 2 minutes. Thus, the 3 minute time window for this action has been reduced by 5% to 171 seconds. Changing this system window in the HRA calculation for this action increases the failure probability from 2.4E-2 to 2.8E-2. As sensitivity, the 3 minute time window for this action is reduced by 10% to 162 seconds. For the 10% sensitivity, the probability for this action is increased from 2.4E-2 to 3.3E-2.

- ZEPC2 (control FW and RPV level, 6.8E-2): This action is to control Feedwater, post-ATWS-runback so that power is maintained below 4% (EOP specification). Due to the 55 second time lockout for ATWS Feedwater control, power should be drastically reduced before operator intervention is allowed and this action is not driven in large measure by time constraints. This is more of a "failure on demand" type action. However, time pressure is noticeable as ATWS events can precede quickly. Operators need to manually manipulate feedwater FVCs when the runback clears (55 seconds). Time can be significant in that an RPV level over-response can only be tolerated for a short duration before RPV overpower can cause a significant RPV challenge. The timing depends on several factors such as the amount of rods which fail to SCRAM, one versus two SLC pump success, as well as the status of the core cycle. Based on a review of ATWS MAAP runs (MAAP U2AT1PA, U2AT2PA, and U2AT3P), 3 minutes is judged a reasonable time window for this action. Since the time for this action is driven primarily by the time to reach RPV level 1 and reaching level 1 occurs within the first 2 minutes, the time window is taken as 5% reduced, proportionate with the MELLLA+ increased steam flow over the first 2 minutes. Thus, the 3 minute time window for this action has been reduced by 5% to 171 seconds. Changing this system window in the HRA calculation for this action increases the failure probability from 6.8E-2 to 8.4E-2. As sensitivity, the 3 minute time window for this action is reduced by 10% to 162 seconds. For the 10% sensitivity, the probability for this action is increased from 6.8E-2 to 1.0E-1.

- ZOH02 (align both trains of suppression pool cooling early to prevent HCTL, 0.98): since this action is presently not credited in the PRA model, there is no change due to MELLLA+.

- ZOD03 (RPV blowdown for LPI (loss of FW), 1.2E-2): this action is to blowdown the RPV to provide LPI given that RPV water level cannot be maintained above -39". A significant power reduction would occur, regardless of other equipment, if RPV level drops to -39". While the time to reach this level can be impacted to some degree by MELLLA+ conditions, the continuation to core damage would be driven by boil-off from caused by decay heat at low power. Since the time for this action is 6 minutes, the time window is taken as 10% reduced proportionate with the MELLLA+ increased steam flow over that time frame. Thus, the 6 minutes time window for this action has been reduced by 10% to 324 seconds. Changing this system window in the HRA calculation for this action increases the failure probability from 1.2E-2 to 1.5E-2.

- ZCH01 (terminate and prevent overfilling, 3.9E-2): This action is to secure LPI to prevent unintended injection from flushing Boron or otherwise introducing positive reactivity. LPI can only inject once RPV pressure has been reduced. Presumably this would only occur once power had been significantly reduced. Since this occurs following successful initial ATWS response, the challenge is not tied to MELLLA+ versus non-MELLLA+ time but rather following through with EOP directions once initial response actions are complete. Therefore, any minor timing impacts caused by MELLLA+ conditions would be a minimal impact on this action. The value assigned for post-EPU conditions is deemed reasonable for MELLLA+ conditions.

- ZCH02 (slowly raise level with LPI (loss of FW), 3.4E-2): This action is taken following injection of SLC and is intended to address the potential for overfilling which would introduce positive reactivity. This action is taken after power has been reduced and MELLLA+ and non-MELLLA+ challenges would be similar. The value assigned for post-EPU conditions is deemed reasonable for MELLLA+ conditions.

In summary, the following HEP changes will be included in the MELLLA+ PRA model:

- Increase ZAI01 HEP from 6.4E-3 to 8.4E-3
- Increase ZEPC1 HEP from 2.4E-2 to 2.8E-2
- Increase ZEPC2 HEP from 6.8E-2 to 8.4E-2
- Increase ZOD03 HEP from 1.2E-2 to 1.5E-2

3.3.2 Risk Calculations

The updated HEP values for the ATWS MELLLA+ model were added to the NMP2 EPU PRA model (Reference 1) so that the risk increase from MELLLA+ operation can be calculated. This is considered the real risk increase that will result from the PRA update for MELLLA+ (all other results presented below are for sensitivity analysis only). The following summarizes the results:

Risk Metric	NMP2 EPU	MELLLA+	Risk Increase
CDF	3.77E-6	3.78E-6	1E-8
LERF	3.92E-7	3.95E-7	3E-9

Note that the above results conservatively assume the plant operates in the MELLLA+ domain during the whole cycle.

Sensitivity analyses were performed to demonstrate that some of the uncertainties with this evaluation have a minor impact on risk. A sensitivity mentioned above applies to the potential for an increase in initiating event frequency. While the MELLLA+ improvement should reduce initiating event frequency, it is possible that some of the changes could result in an additional unplanned event, particularly early in the implementation period. As sensitivity, a higher SCRAM challenge frequency is assumed, in addition to the new baseline HEP impacts discussed above. If 1 additional event were to occur over the standard 10 year initiating event data collection window, the impact is an increase of 0.1 per year. Based on the discussion above, a spurious SCRAM is the more likely event to occur; however a real RPS challenge initiating event is also considered as an extreme sensitivity. For these sensitivity cases, the following changes were made:

- Increase ZAI01 HEP from 6.4E-3 to 8.4E-3 (new baseline MELLLA+ HEP value)
- Increase ZEPC1 HEP from 2.4E-2 to 2.8E-2 (new baseline MELLLA+ HEP value)
- Increase ZEPC2 HEP from 6.8E-2 to 8.4E-2 (new baseline MELLLA+ HEP value)
- Increase ZOD03 HEP from 1.2E-2 to 1.5E-2 (new baseline MELLLA+ HEP value)
- Increase %SCRAM IE Frequency from 9.26E-2/yr to 0.1926/yr

The following summarizes the results:

Sensitivity (1) for %SCRAM Impact and Adjusted Baseline MELLLA+ HEPs			
Risk Metric	NMP2 EPU	MELLLA+	Risk Increase
CDF	3.77E-6	3.78E-6	1E-8
LERF	3.92E-7	3.95E-7	3E-9

The %RPS sensitivity case is the same as %SCRAM above except %RPS is increased from 0.275/yr to 0.375/yr instead of the %SCRAM increase. The following summarizes the results:

Sensitivity (2) for %RPS Impact and Adjusted Baseline MELLLA+ HEPs			
Risk Metric	NMP2 EPU	MELLLA+	Risk Increase
CDF	3.77E-6	3.80E-6	3E-8
LERF	3.92E-7	4.00E-7	8E-9

Also, %LOC sensitivity case was performed the same as above to simulate a potential increase in frequency due to increased moisture content. %LOC is increased from 0.067 to 0.167. The following summarizes the results:

Sensitivity (3) for %LOC Impact and Adjusted Baseline MELLLA+ HEPs			
Risk Metric	**NMP2 EPU**	**MELLLA+**	**Risk Increase**
CDF	3.77E-6	3.88E-6	1E-7
LERF	3.92E-7	3.99E-7	7E-9

The potential for loss of a feedwater pump to cause a scram was identified although the PRA model already includes this event (PLOF) and conservatively assumes the scram occurs. Although a sensitivity analysis was not necessary, the contribution to CDF and LERF from PLOF is provided here to show that even if a small increase (e.g., 10%) was assumed the risk would be very low:

PLOF CDF = 1.2E-8
PLOF LERF = 3.0E-9

The MELLLA+ model is also used to calculate CDF and LERF for sequences ATWS-006, 007, 014 and 015 to provide an extreme sensitivity on the impact of a stuck open SRV. Total CDF for these sequences is 1.07E-7/yr and total LERF for these sequences is 3.09E-8/yr. The following summarizes the results after multiplying these totals by 110% to account for the increased probability of a stuck open SRV:

ΔCDF = 1.07E-8
ΔLERF = 3.09E-9

Sensitivity was also considered assuming a bounding 10% time window reduction to all impacted ATWS HEPs. This assumes that the 10% reduction applies even for ATWS actions driven by early event response even though the MELLLA+ versus non-MELLLA+ conditions are very similar in the initial 2 minutes of an event. For this sensitivity, the following changes were made:

- Increase ZAI01 HEP from 6.4E-3 to 1.2E-2
- Increase ZEPC1 HEP from 2.4E-2 to 3.3E-2
- Increase ZEPC2 HEP from 6.8E-2 to 1.0E-1
- Increase ZOD03 HEP from 1.2E-2 to 1.5E-2

The following summarizes the results:

Sensitivity (4) for Higher MELLLA+ HEP Impact			
Risk Metric	**NMP2 EPU**	**MELLLA+**	**Risk Increase**
CDF	3.77E-6	3.80E-6	3E-8
LERF	3.92E-7	4.00E-7	8E-9

Finally, sensitivity is performed where the increased initiating event impact is applied along with the higher (i.e., 10%) reduction in time for the ATWS HEPs. For this sensitivity, the following changes were made:

- Increase ZAI01 HEP from 6.4E-3 to 1.2E-2
- Increase ZEPC1 HEP from 2.4E-2 to 3.3E-2

- Increase ZEPC2 HEP from 6.8E-2 to 1.0E-1
- Increase ZOD03 HEP from 1.2E-2 to 1.5E-2
- Increase %SCRAM IE Frequency from 9.26E-2/yr to 0.1926/yr

The following summarizes the results:

Sensitivity (5) for %SCRAM IE Impact and Higher MELLLA+ HEP Impact			
Risk Metric	**NMP2 EPU**	**MELLLA+**	**Risk Increase**
CDF	3.77E-6	3.8E-6	3E-8
LERF	3.92E-7	4.0E-7	8E-9

4 Results and Conclusion

The proposed MELLLA+ operating region for NMP2 has been reviewed to determine the impact on the PRA. The PRA is based on the EPU MELLLA operating region and includes internal events as well as fire and seismic initiating events. The impact of MELLLA+ on the PRA is very low and meets NRC guidelines in Regulatory Guide 1.174 for core damage frequency (CDF) and large early release frequency (LERF). MELLLA+ has no impact on the risk associated with accidents initiated during shutdown conditions.

The estimated risk increase for at-power events due to MELLLA+ is a delta CDF of 1E-8 and delta LERF of 3E-9. This represents a very small risk change in Regulatory Guide 1.174. Based on these results, the proposed MELLLA+ operating region is acceptable on a risk basis.

Sensitivity analyses results also demonstrate a low risk with CDF and LERF changes no more than 1E-7.

References

1. NMP2 PRA (U2MODEL2012R0, EPU and External Events)

2. Regulatory Guide 1.200, Rev 2 "An Approach For Determining The Technical Adequacy of Probabilistic Risk Assessment Results For Risk-Informed Activities."

3. ASME/ANS RA-Sa-2009 "Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications"

4. Amendment No. 138, Renewed License No. NPF-69 "Nine Mile Point Nuclear Station, LLC (NMPNS), Docket No 50-410, Nine Mile Point Nuclear Station, Unit No 2 Amendment to Facility Operating License"

5. NEDC-33006P-A "Maximum Extended Load Line Limit Analysis Plus" Revision 3, June 2009

6. NEDC-33004P-A "Constant Pressure Power Uprate" Revision 4, July 2003

7. NEDC-32424P-A "Generic Guidelines for General Electric Boiling Water Reactor Extended Power Uprate" February 1999

8. NEDC-32523P-A "Generic Evaluations for General Electric Boiling Water Reactor Extended Power Uprate" February 2000

9. Regulatory Guide 1.174 "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis"

10. RS-001, Rev. 0 "U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation Review Standard for Extended Power Uprates" December 2003

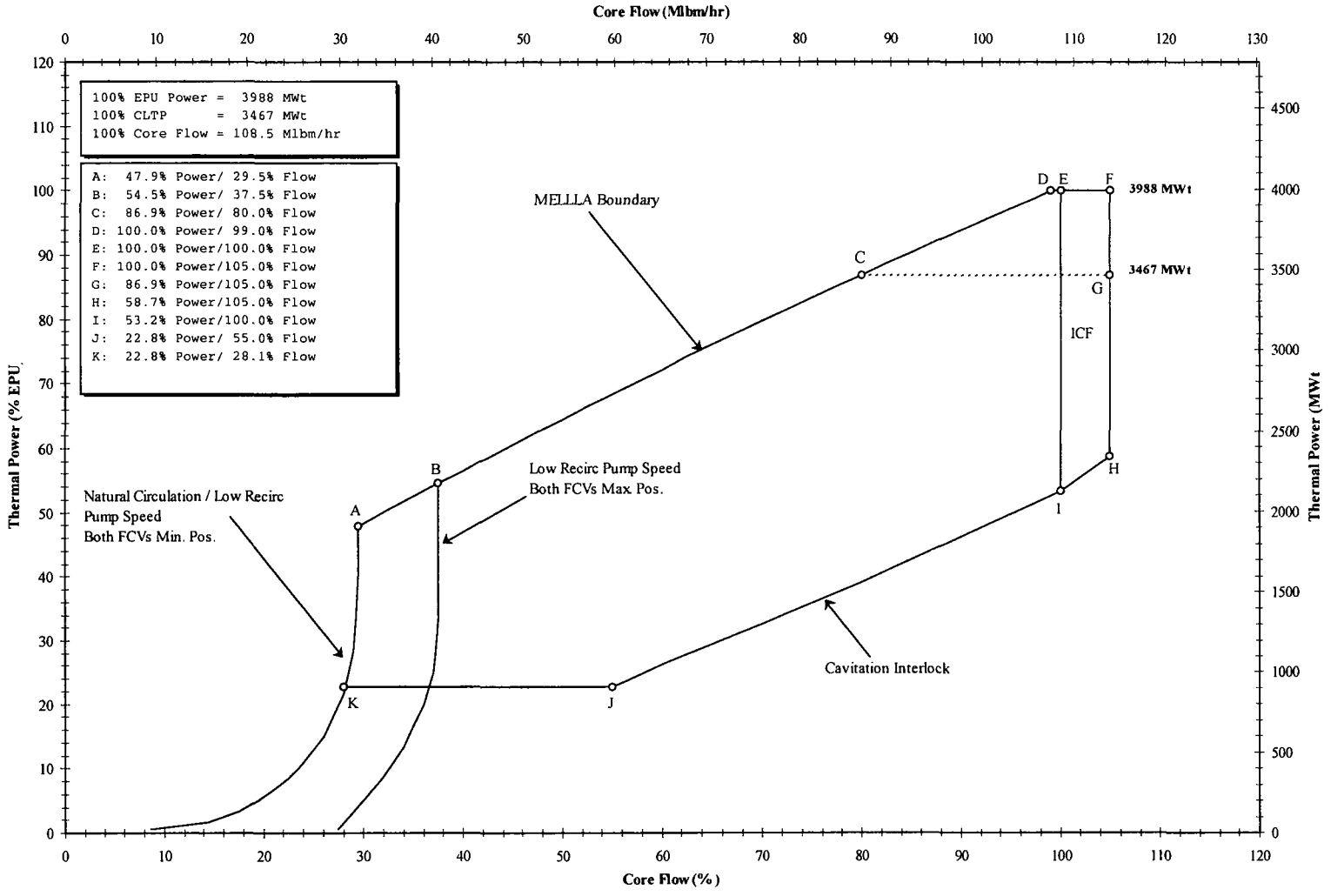
11. NEDC-33576P "Safety Analysis Report For Nine Mile Point Nuclear Station Unit 2 Maximum Extended Load Line Limit Analysis Plus" Draft B Rev 0, February 2012

ATTACHMENT 5

NINE MILE POINT UNIT 2
POWER/FLOW OPERATING MAP FOR CURRENT CYCLE

Nine Mile Point Nuclear Station, LLC
November 1, 2013

ATTACHMENT 5
Nine Mile Point Unit 2 Power/Flow Operating Map for Current Cycle



ATTACHMENT 6

GENERAL ELECTRIC – HITACHI
AFFIDAVIT JUSTIFYING WITHHOLDING
PROPRIETARY INFORMATION IN NEDC-33576P

Nine Mile Point Nuclear Station, LLC
November 1, 2013

GE-Hitachi Nuclear Energy Americas LLC

AFFIDAVIT

I, Peter M. Yandow, state as follows:

(1) I am the Manager, NPP/Services Licensing, Regulatory Affairs, GE-Hitachi Nuclear Energy Americas LLC (GEH), and have been delegated the function of reviewing the information described in paragraph (2) which is sought to be withheld, and have been authorized to apply for its withholding.

(2) The information sought to be withheld is contained in GEH proprietary report NEDC-33576P, "Safety Analysis Report for Nine Mile Point Unit 2 Maximum Extended Load Line Limit Analysis Plus," Revision 0, dated October 2013. GEH proprietary information in NEDC-33576P is identified by a dotted underline inside double square brackets. [[This sentence is an example.{3}]]. Figures and large equation objects containing GEH proprietary information are identified with double square brackets before and after the object. In each case, the superscript notation {3} refers to Paragraph (3) of this affidavit, which provides the basis for the proprietary determination.

(3) In making this application for withholding of proprietary information of which it is the owner or licensee, GEH relies upon the exemption from disclosure set forth in the *Freedom of Information Act* ("FOIA"), 5 U.S.C. §552(b)(4), and the *Trade Secrets Act*, 18 U.S.C. §1905, and NRC regulations 10 CFR 9.17(a)(4), and 2.390(a)(4) for trade secrets (Exemption 4). The material for which exemption from disclosure is here sought also qualifies under the narrower definition of trade secret, within the meanings assigned to those terms for purposes of FOIA Exemption 4 in, respectively, Critical Mass Energy Project v. Nuclear Regulatory Commission, 975 F.2d 871 (D.C. Cir. 1992), and Public Citizen Health Research Group v. FDA, 704 F.2d 1280 (D.C. Cir. 1983).

(4) The information sought to be withheld is considered to be proprietary for the reasons set forth in paragraphs (4)a and (4)b. Some examples of categories of information that fit into the definition of proprietary information are:

a. Information that discloses a process, method, or apparatus, including supporting data and analyses, where prevention of its use by GEH's competitors without a license from GEH constitutes a competitive economic advantage over other companies;

b. Information that, if used by a competitor, would reduce its expenditure of resources or improve its competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing of a similar product;

c. Information that reveals aspects of past, present, or future GEH customer-funded development plans and programs, resulting in potential products to GEH;

d. Information that discloses trade secret or potentially patentable subject matter for which it may be desirable to obtain patent protection.

(5) To address 10 CFR 2.390(b)(4), the information sought to be withheld is being submitted to NRC in confidence. The information is of a sort customarily held in confidence by GEH, and is in fact so held. The information sought to be withheld has, to the best of my knowledge and belief, consistently been held in confidence by GEH, not been disclosed publicly, and not been made available in public sources. All disclosures to third parties, including any required transmittals to the NRC, have been made, or must be made, pursuant to regulatory provisions for proprietary or confidentiality agreements or both that provide for maintaining the information in confidence. The initial designation of this information as proprietary information, and the subsequent steps taken to prevent its unauthorized disclosure, are as set forth in the following paragraphs (6) and (7).

(6) Initial approval of proprietary treatment of a document is made by the manager of the originating component, who is the person most likely to be acquainted with the value and sensitivity of the information in relation to industry knowledge, or who is the person most likely to be subject to the terms under which it was licensed to GEH. Access to such documents within GEH is limited to a "need to know" basis.

(7) The procedure for approval of external release of such a document typically requires review by the staff manager, project manager, principal scientist, or other equivalent authority for technical content, competitive effect, and determination of the accuracy of the proprietary designation. Disclosures outside GEH are limited to regulatory bodies, customers, and potential customers, and their agents, suppliers, and licensees, and others with a legitimate need for the information, and then only in accordance with appropriate regulatory provisions or proprietary and/or confidentiality agreements.

(8) The information identified in paragraph (2) is classified as proprietary because it contains detailed results and conclusions regarding supporting evaluations of the safety-significant changes necessary to demonstrate the regulatory acceptability of the Maximum Extended Load Line Limit Analysis Plus analysis for a GEH Boiling Water Reactor ("BWR"). The analysis utilized analytical models and methods, including computer codes, which GEH has developed, obtained NRC approval of, and applied to perform evaluations of Maximum Extended Load Line Limit Analysis Plus for a GEH BWR.

The development of the evaluation processes along with the interpretation and application of the analytical results is derived from the extensive experience and information databases that constitute major GEH assets.

(9) Public disclosure of the information sought to be withheld is likely to cause substantial harm to GEH's competitive position and foreclose or reduce the availability of profit-

making opportunities. The information is part of GEH's comprehensive BWR safety and technology base, and its commercial value extends beyond the original development cost. The value of the technology base goes beyond the extensive physical database and analytical methodology and includes development of the expertise to determine and apply the appropriate evaluation process. In addition, the technology base includes the value derived from providing analyses done with NRC-approved methods.

The research, development, engineering, analytical and NRC review costs comprise a substantial investment of time and money by GEH. The precise value of the expertise to devise an evaluation process and apply the correct analytical methodology is difficult to quantify, but it clearly is substantial. GEH's competitive advantage will be lost if its competitors are able to use the results of the GEH experience to normalize or verify their own process or if they are able to claim an equivalent understanding by demonstrating that they can arrive at the same or similar conclusions.

The value of this information to GEH would be lost if the information were disclosed to the public. Making such information available to competitors without their having been required to undertake a similar expenditure of resources would unfairly provide competitors with a windfall, and deprive GEH of the opportunity to exercise its competitive advantage to seek an adequate return on its large investment in developing and obtaining these very valuable analytical tools.

I declare under penalty of perjury that the foregoing affidavit and the matters stated therein are true and correct to the best of my knowledge, information, and belief.

Executed on this 17th day of October 2013.

Peter M. Yandow
Manager, NPP/Services Licensing
Regulatory Affairs
GE-Hitachi Nuclear Energy Americas LLC

ATTACHMENT 7

GLOBAL NUCLEAR FUEL
AFFIDAVIT JUSTIFYING WITHHOLDING
PROPRIETARY INFORMATION IN GNF-0000-0156-7490-R0-P

Nine Mile Point Nuclear Station, LLC
November 1, 2013

Global Nuclear Fuel – Americas
AFFIDAVIT

I, Russell M. Fawcett, state as follows:

(1) I am Manager, Core, Fuel and Advanced Design, Global Nuclear Fuel – Americas, LLC ("GNF-A"), and have been delegated the function of reviewing the information described in paragraph (2) which is sought to be withheld, and have been authorized to apply for its withholding.

(2) The information sought to be withheld is contained in GNF-A proprietary report GNF-0000-0156-7490-R0-P, entitled *GNF Additional Information Regarding the Requested Changes to the Technical Specification SLMCPR,* Revision 0, August 26, 2013. GNF-A proprietary information in *GNF Additional Information Regarding the Requested Changes to the Technical Specification SLMCPR,* Revision 0, August 26, 2013 is identified by a dotted underline within double square brackets. [[This sentence is an example.{3}]] Figures and large objects are identified with double square brackets before and after the object. In all cases, the superscript notation {3} refers to Paragraph (3) of the enclosed affidavit, which provides the basis for the proprietary determination.

(3) In making this application for withholding of proprietary information of which it is the owner or licensee, GNF-A relies upon the exemption from disclosure set forth in the Freedom of Information Act ("FOIA"), 5 USC Sec. 552(b)(4), and the Trade Secrets Act, 18 USC Sec. 1905, and NRC regulations 10 CFR 9.17(a)(4), and 2.390(a)(4) for "trade secrets" (Exemption 4). The material for which exemption from disclosure is here sought also qualify under the narrower definition of "trade secret", within the meanings assigned to those terms for purposes of FOIA Exemption 4 in, respectively, Critical Mass Energy Project v. Nuclear Regulatory Commission, 975F2d871 (DC Cir. 1992), and Public Citizen Health Research Group v. FDA, 704F2d1280 (DC Cir. 1983).

(4) Some examples of categories of information which fit into the definition of proprietary information are:

a. Information that discloses a process, method, or apparatus, including supporting data and analyses, where prevention of its use by GNF-A's competitors without license from GNF-A constitutes a competitive economic advantage over other companies;

b. Information which, if used by a competitor, would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing of a similar product;

c. Information which reveals aspects of past, present, or future GNF-A customer-funded development plans and programs, resulting in potential products to GNF-A;

d. Information which discloses patentable subject matter for which it may be desirable to obtain patent protection.

The information sought to be withheld is considered to be proprietary for the reasons set forth in paragraphs (4)a. and (4)b. above.

(5) To address 10 CFR 2.390 (b) (4), the information sought to be withheld is being submitted to NRC in confidence. The information is of a sort customarily held in confidence by GNF-A, and is in fact so held. The information sought to be withheld has, to the best of my knowledge and belief, consistently been held in confidence by GNF-A, no public disclosure has been made, and it is not available in public sources. All disclosures to third parties including any required transmittals to NRC, have been made, or must be made, pursuant to regulatory provisions or proprietary agreements which provide for maintenance of the information in confidence. Its initial designation as proprietary information, and the subsequent steps taken to prevent its unauthorized disclosure, are as set forth in paragraphs (6) and (7) following.

(6) Initial approval of proprietary treatment of a document is made by the manager of the originating component, the person most likely to be acquainted with the value and sensitivity of the information in relation to industry knowledge, or subject to the terms under which it was licensed to GNF-A. Access to such documents within GNF-A is limited on a "need to know" basis.

(7) The procedure for approval of external release of such a document typically requires review by the staff manager, project manager, principal scientist or other equivalent authority, by the manager of the cognizant marketing function (or his delegate), and by the Legal Operation, for technical content, competitive effect, and determination of the accuracy of the proprietary designation. Disclosures outside GNF-A are limited to regulatory bodies, customers, and potential customers, and their agents, suppliers, and licensees, and others with a legitimate need for the information, and then only in accordance with appropriate regulatory provisions or proprietary agreements.

(8) The information identified in paragraph (2) is classified as proprietary because it contains details of GNF-A's fuel design and licensing methodology. The development of this methodology, along with the testing, development and approval was achieved at a significant cost to GNF-A or its licensor.

The development of the fuel design and licensing methodology along with the interpretation and application of the analytical results is derived from an extensive experience database that constitutes a major GNF-A asset.

(9) Public disclosure of the information sought to be withheld is likely to cause substantial harm to GNF-A's competitive position and foreclose or reduce the availability of profit-making opportunities. The information is part of GNF-A's comprehensive BWR safety and technology base, and its commercial value extends beyond the original development cost. The value of the technology base goes beyond the extensive physical database and analytical methodology and includes development of the expertise to determine and apply the appropriate evaluation process. In addition, the technology base includes the value derived from providing analyses done with NRC-approved methods.

The research, development, engineering, analytical, and NRC review costs comprise a substantial investment of time and money by GNF-A.

The precise value of the expertise to devise an evaluation process and apply the correct analytical methodology is difficult to quantify, but it clearly is substantial.

GNF-A's competitive advantage will be lost if its competitors are able to use the results of the GNF-A experience to normalize or verify their own process or if they are able to claim an equivalent understanding by demonstrating that they can arrive at the same or similar conclusions.

The value of this information to GNF-A would be lost if the information were disclosed to the public. Making such information available to competitors without their having been required to undertake a similar expenditure of resources would unfairly provide competitors with a windfall, and deprive GNF-A of the opportunity to exercise its competitive advantage to seek an adequate return on its large investment in developing and obtaining these very valuable analytical tools.

I declare under penalty of perjury that the foregoing affidavit and the matters stated therein are true and correct to the best of my knowledge, information, and belief.

Executed on this 28th day of August 2013.

Russell M. Fawcett
Manager, Core, Fuel and Advanced Design
Global Nuclear Fuel – Americas, LLC