UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

March 28, 2014

Karen D. Fili
Site Vice President
Monticello Nuclear Generating Plant
Northern States Power Company - Minnesota
2807 West County Road 75
Monticello, MN 55362-9637

SUBJECT: MONTICELLO NUCLEAR GENERATING PLANT - ISSUANCE OF AMENDMENT NO. 180 TO RENEWED FACILITY OPERATING LICENSE REGARDING MAXIMUM EXTENDED LOAD LINE LIMIT ANALYSIS PLUS (TAC NO. ME3145)

Dear Mrs. Fili:

The U.S. Nuclear Regulatory Commission (NRC) has issued the enclosed Amendment No. 180 to Renewed Facility Operating License No. DPR-22 for the Monticello Nuclear Generating Plant (MNGP). This amendment consists of changes to the Renewed Facility Operating License and the Technical Specifications in response to your application dated January 21, 2010, as supplemented by letters dated March 4, 2010, September 28, 2010, November 11, 2010, June 27, 2012, September 28, 2012, November 30, 2012 (2 letters), December 21, 2012, March 21, 2013, May 13, 2013, June 26, 2013, July 8, 2013, July 31, 2013, August 14, 2013, October 4, 2013, December 20, 2013, and February 24, 2014.

The amendment revises the MNGP technical specifications to allow plant operation from the currently licensee Maximum Extended Load Line Limit Analysis (MELLLA) operating domain to operation in the expanded MELLLA Plus (MELLLA+) operating domain under the previously approved extended power uprate conditions of 2004 megawatts thermal rated core thermal power.

A copy of our related safety evaluation (SE) is also enclosed. The NRC has determined that the related SE contains proprietary information pursuant to Title 10 of the *Code of Federal Regulations,* Section 2.390, "Public Inspections, Exemptions, Requests for Withholding." Accordingly, the NRC staff has also prepared a redacted, publicly-available, non-proprietary version of the SE. Copies of the proprietary and non-proprietary versions of the SE are enclosed.

A Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,

Terry A. Beltz, Senior Project Manager
Plant Licensing Branch III-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-263

Enclosures:
1. Amendment No. 180 to License No. DPR-22
2. Safety Evaluation (non-proprietary): ML14087A013
3. Safety Evaluation (proprietary): ML13317A866

cc w/o Enclosure 3: Distribution via Listserv

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

NORTHERN STATES POWER COMPANY

DOCKET NO. 50-263

MONTICELLO NUCLEAR GENERATING PLANT

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 180
License No. DPR-22

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Northern States Power Company (NSPM, the licensee), dated January 21, 2010, as supplemented by letters dated March 4, 2010, September 28, 2010, November 11, 2010, June 27, 2012, September 28, 2012, November 30, 2012 (2 letters), December 21, 2012, March 21, 2013, May 13, 2013, June 26, 2013, July 8, 2013, July 31, 2013, August 14, 2013, October 4, 2013, December 20, 2013, and February 24, 2014, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this license amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.2 of Renewed Facility Operating License No. DPR-22 is hereby amended to read as follows:

   Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 180, are hereby incorporated in the license. NSPM shall operate the facility in accordance with the Technical Specifications.

3. The license amendment is effective as of its date of issuance and shall be implemented within 120 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert D. Carlson, Chief
Plant Licensing Branch III-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to Renewed Facility
Operating License DPR-22 and
Technical Specifications

Date of Issuance: March 28, 2014

ATTACHMENT TO LICENSE AMENDMENT NO. 180

RENEWED FACILITY OPERATING LICENSE NO. DPR-22

DOCKET NO. 50-263

Replace the following pages of the Renewed Facility Operating License No. DPR-22 and the Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain vertical lines indicating the areas of change.

License Pages

REMOVE	INSERT
- 3 -	- 3 -

Technical Specifications (Appendix A) Pages

REMOVE	INSERT
3.3.1.1-3	3.3.1.1-3
3.3.1.1-4	3.3.1.1-4
3.3.1.1-5	3.3.1.1-5
3.3.1.1-6	3.3.1.1-6
3.3.1.1-7	3.3.1.1-7
3.3.1.1-8	3.3.1.1-8
3.3.1.1-9	3.3.1.1-9
3.3.1.1-10	3.3.1.1-10
3.4.1-1	3.4.1-1
5.6-2	5.6-2
5.6-3	5.6-3

2. Pursuant to the Act and 10 CFR Part 70, NSPM to receive, possess, and use at any time special nuclear material as reactor fuel, in accordance with the limitations for storage and amounts required for reactor operations, as described in the Final Safety Analysis Report, as supplemented and amended, and the licensee's filings dated August 16, 1974 (those portions dealing with handling of reactor fuel);

3. Pursuant to the Act and 10 CFR Parts 30, 40 and 70, NSPM to receive, possess, and use at any time any byproduct, source and special nuclear material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

4. Pursuant to the Act and 10 CFR Parts 30, 40 and 70, NSPM to receive, possess, and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components; and

5. Pursuant to the Act and 10 CFR Parts 30 and 70, NSPM to possess, but not separate, such byproduct and special nuclear material as may be produced by operation of the facility.

C. This renewed operating license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission, now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

1. Maximum Power Level

   NSPM is authorized to operate the facility at steady state reactor core power levels not in excess of 2004 megawatts (thermal).

2. Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 180, are hereby incorporated in the license. NSPM shall operate the facility in accordance with the Technical Specifications.

3. Physical Protection

   NSPM shall implement and maintain in effect all provisions of the Commission-approved physical security, guard training and qualification, and safeguards contingency plans including amendments made pursuant to provisions of the Miscellaneous Amendments and Search

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
I. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	I.1	Initiate action to implement the Manual BSP Regions defined in the COLR.	Immediately
	AND		
	I.2	Implement the Automated BSP Scram Region using the modified APRM Simulated Thermal Power – High scram setpoints defined in the COLR.	12 hours
	AND		
	I.3	Initiate action in accordance with Specification 5.6.6.	Immediately
J. Required Action and associated Completion Time of Condition I 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 < 20% RTP.	4 hours

SURVEILLANCE REQUIREMENTS

-----NOTES-----

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	-----NOTE----- Not required to be performed until 12 hours after THERMAL POWER ≥ 25% RTP. ----- Verify the absolute difference between the average power range monitor (APRM) channels and the calculated power is ≤ 2% RTP while operating at ≥ 25% RTP.	7 days
SR 3.3.1.1.3	-----NOTE----- Not required to be performed when entering MODE 2 from MODE 1 until 12 hours after entering MODE 2. ----- Perform CHANNEL FUNCTIONAL TEST.	7 days
SR 3.3.1.1.4	Perform a functional test of each RPS automatic scram contactor.	7 days
SR 3.3.1.1.5	Perform CHANNEL FUNCTIONAL TEST.	31 days
SR 3.3.1.1.6	Calibrate the local power range monitors.	1000 megawatt days per ton
SR 3.3.1.1.7	Perform CHANNEL FUNCTIONAL TEST.	92 days

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
SR 3.3.1.1.8	Calibrate the trip units.	92 days
SR 3.3.1.1.9	Perform CHANNEL CALIBRATION.	92 days
SR 3.3.1.1.10	Perform CHANNEL FUNCTIONAL TEST.	24 months
SR 3.3.1.1.11	-----NOTES----- 1. Neutron detectors are excluded. 2. For Function 1, not required to be performed when entering MODE 2 from MODE 1 until 12 hours after entering MODE 2. 3. For Functions 2.b and 2.f, the recirculation flow transmitters that feed the APRMs are included. ---------- Perform CHANNEL CALIBRATION.	24 months
SR 3.3.1.1.12	Perform LOGIC SYSTEM FUNCTIONAL TEST.	24 months
SR 3.3.1.1.13	Verify Turbine Stop Valve - Closure and Turbine Control Valve Fast Closure, Acceleration Relay Oil Pressure - Low Functions are not bypassed when THERMAL POWER is > 40% RTP.	24 months

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.1.14	------------------------------NOTES------------------------------ 1. For Function 2.e, "n" equals 8 channels for the purpose of determining the STAGGERED TEST BASIS Frequency. Testing of APRM and OPRM outputs shall alternate. 2. For Function 5, "n" equals 4 channels for the purpose of determining the STAGGERED TEST BASIS Frequency. --- Verify the RPS RESPONSE TIME is within limits.	24 months on a STAGGERED TEST BASIS
SR 3.3.1.1.15	------------------------------NOTES------------------------------ 1. For Function 2.a, not required to be performed when entering MODE 2 from MODE 1 until 12 hours after entering MODE 2. 2. For Functions 2.b and 2.f, the CHANNEL FUNCTIONAL TEST includes the recirculation flow input processing, excluding the flow transmitters. --- Perform CHANNEL FUNCTIONAL TEST.	184 days

Table 3.3.1.1-1 (page 1 of 4)
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 – High High	2	3	G	SR 3.3.1.1.1 SR 3.3.1.1.3 SR 3.3.1.1.4 SR 3.3.1.1.11 SR 3.3.1.1.12 SR 3.3.1.1.14	≤ 122/125 divisions of full scale
	5[(a)]	3	H	SR 3.3.1.1.1 SR 3.3.1.1.3 SR 3.3.1.1.4 SR 3.3.1.1.11 SR 3.3.1.1.12 SR 3.3.1.1.14	≤ 122/125 divisions of full scale
b. Inop.	2	3	G	SR 3.3.1.1.3 SR 3.3.1.1.4 SR 3.3.1.1.12	NA
	5[(a)]	3	H	SR 3.3.1.1.3 SR 3.3.1.1.4 SR 3.3.1.1.12	NA
2. Average Power Range Monitors					
a. Neutron Flux – High, (Setdown)	2	3[(c)]	G	SR 3.3.1.1.1 SR 3.3.1.1.4 SR 3.3.1.1.6 SR 3.3.1.1.11 SR 3.3.1.1.15	≤ 20% RTP
b. Simulated Thermal Power – High	1	3[(c)]	F	SR 3.3.1.1.1 SR 3.3.1.1.2 SR 3.3.1.1.4 SR 3.3.1.1.6 SR 3.3.1.1.11 SR 3.3.1.1.15	≤ 0.61W + 67.2% RTP[(b)(h)] and ≤ 116% RTP

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

(b) ≤ 0.55 (W – Delta W) + 61.5% RTP when reset for single loop operation per LCO 3.4.1, "Recirculation Loops Operating." The cycle-specific value for Delta W is specified in the COLR.

(c) Each APRM / OPRM channel provides inputs to both trip systems.

(h) With OPRM Upscale (Function 2.f) 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 I of this specification.

Table 3.3.1.1-1 (page 2 of 4)
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
c. Neutron Flux – High	1	3[c]	F	SR 3.3.1.1.1 SR 3.3.1.1.2 SR 3.3.1.1.4 SR 3.3.1.1.6 SR 3.3.1.1.11[f][g] SR 3.3.1.1.15	≤ 122% RTP
d. Inop.	1, 2	3[c]	G	SR 3.3.1.1.4 SR 3.3.1.1.15	NA
e. 2-Out-Of-4 Voter	1, 2	2	G	SR 3.3.1.1.1 SR 3.3.1.1.4 SR 3.3.1.1.12 SR 3.3.1.1.14 SR 3.3.1.1.15	NA
f. OPRM Upscale[e]	≥ 20% RTP	3[c]	I	SR 3.3.1.1.1 SR 3.3.1.1.4 SR 3.3.1.1.6 SR 3.3.1.1.11 SR 3.3.1.1.15	
3. Reactor Vessel Steam Dome Pressure – High	1, 2	2	G	SR 3.3.1.1.4 SR 3.3.1.1.7 SR 3.3.1.1.9 SR 3.3.1.1.12 SR 3.3.1.1.14	≤ 1075 psig
4. Reactor Vessel Water Level – Low	1, 2	2	G	SR 3.3.1.1.1 SR 3.3.1.1.4 SR 3.3.1.1.7 SR 3.3.1.1.8 SR 3.3.1.1.11 SR 3.3.1.1.12 SR 3.3.1.1.14	≥ 7 inches

(c) Each APRM / OPRM channel provides inputs to both trip systems.

(e) 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.

(f) If the as-found channel setpoint is not the Nominal Trip Setpoint but is conservative with respect to the Allowable Value, then the channel shall be evaluated to verify that it is functioning as required before returning the channel to service.

(g) The instrument channel setpoint shall be reset to the Nominal Trip Setpoint (NTSP) at the completion of the surveillance; otherwise, the channel shall be declared inoperable. The NTSP and the methodology used to determine the NTSP are specified in the Technical Requirements Manual.

Table 3.3.1.1-1 (page 3 of 4)
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
5. Main Steam Isolation Valve – Closure	1, 2[(d)]	8	F	SR 3.3.1.1.4 SR 3.3.1.1.7 SR 3.3.1.1.11 SR 3.3.1.1.12 SR 3.3.1.1.14	≤ 10% closed
6. Drywell Pressure – High	1, 2	2	G	SR 3.3.1.1.4 SR 3.3.1.1.7 SR 3.3.1.1.9 SR 3.3.1.1.12	≤ 2.0 psig
7. Scram Discharge Volume Water Level – High					
a. Resistance Temperature Detector	1, 2	2	G	SR 3.3.1.1.4 SR 3.3.1.1.7 SR 3.3.1.1.9 SR 3.3.1.1.12	≤ 56.0 gallons
	5[(a)]	2	H	SR 3.3.1.1.4 SR 3.3.1.1.7 SR 3.3.1.1.9 SR 3.3.1.1.12	≤ 56.0 gallons
b. Float Switch	1, 2	2	G	SR 3.3.1.1.4 SR 3.3.1.1.7 SR 3.3.1.1.9 SR 3.3.1.1.12	≤ 56.0 gallons
	5[(a)]	2	H	SR 3.3.1.1.4 SR 3.3.1.1.7 SR 3.3.1.1.9 SR 3.3.1.1.12	≤ 56.0 gallons
8. Turbine Stop Valve – Closure	> 40% RTP	4	E	SR 3.3.1.1.4 SR 3.3.1.1.7 SR 3.3.1.1.11 SR 3.3.1.1.12 SR 3.3.1.1.13 SR 3.3.1.1.14	≤ 10% closed

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

(d) With reactor pressure ≥ 600 psig.

Table 3.3.1.1-1 (page 4 of 4)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
9. Turbine Control Valve Fast Closure, Acceleration Relay Oil Pressure – Low	> 40% RTP	2	E	SR 3.3.1.1.4 SR 3.3.1.1.7 SR 3.3.1.1.9 SR 3.3.1.1.12 SR 3.3.1.1.13 SR 3.3.1.1.14	≥ 167.8 psig
10. Reactor Mode Switch – Shutdown Position	1, 2	1	G	SR 3.3.1.1.10 SR 3.3.1.1.12	NA
	5[(a)]	1	H	SR 3.3.1.1.10 SR 3.3.1.1.12	NA
11. Manual Scram	1, 2	1	G	SR 3.3.1.1.5 SR 3.3.1.1.12	NA
	5[(a)]	1	H	SR 3.3.1.1.5 SR 3.3.1.1.12	NA

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

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 may be in operation provided the plant is not operating in the MELLLA+ domain defined in the COLR and provided the following limits are 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 Monitor Simulated Thermal Power - High), Allowable Value of Table 3.3.1.1-1 is reset for single loop operation.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Requirements of the LCO not met.	A.1 Satisfy the requirements of the LCO.	24 hours
B. Required Action and associated Completion Time of Condition A not met. OR No recirculation loops in operation.	B.1 Be in MODE 3.	12 hours

5.6 Reporting Requirements

5.6.3 CORE OPERATING LIMITS REPORT (COLR) (continued)

4. Control Rod Block Instrumentation Allowable Value for the Table 3.3.2.1-1 Rod Block Monitor Functions 1.a, 1.b, and 1.c and associated Applicability RTP levels;

5. Reactor Protection System Instrumentation Delta W value for Table 3.3.1.1-1, Function 2.b, APRM Simulated Thermal Power – High, Note b; and

6. 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.f), Automated BSP Scram Region, and the BSP Boundary for Specification 3.3.1.1.

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, "General Electric Standard Application for Reactor Fuel"; and

2. (Not Used.)

3. (Not Used.)

4. NEDO-33075-A, Revision 6, "General Electric Boiling Water Reactor Detect and Suppress Solution - Confirmation Density," January 2008.

The COLR will contain the complete identification for each of the Technical Specification referenced topical reports used to prepare the COLR (i.e., report number, title, revision, date, and any supplements).

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

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

5.6 Reporting Requirements

5.6.4 Post Accident Monitoring 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.

5.6.5 Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

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

   1. Limiting Conditions for Operation Section 3.4.9, "RCS Pressure and Temperature (P/T) Limits"

   2. Surveillance Requirements Section 3.4.9, "RCS Pressure and Temperature (P/T) Limits"

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

   1. SIR-05-044-A, "Pressure-Temperature Limits Report Methodology for Boiling Water Reactors," Revision 0, dated April 2007.

c. The PTLR shall be provided to the NRC upon issuance for each reactor vessel fluence period and for any revision or supplements thereto.

5.6.6 OPRM Report

When a report is required by Condition I of LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation," a report shall be submitted within the following 90 days. The report shall outline the preplanned means to provide backup stability protection, the cause of the inoperability, and the plans and schedule for restoring the required instrumentation channels to OPERABLE status.

Enclosure 2

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 180 TO

RENEWED FACILITY OPERATING LICENSE NO. DPR-22

NORTHERN STATES POWER COMPANY

MONTICELLO NUCLEAR GENERATING PLANT

DOCKET NO. 50-263

(Non-Proprietary - Redacted Version)

ADAMS Accession Number: ML14087A013

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

REGARDING LICENSE AMENDMENT NO. 180 TO

RENEWED FACILITY OPERATING LICENSE NO. DPR-22

FOR MAXIMUM EXTENDED LOAD LINE LIMIT ANALYSIS PLUS (MELLLA+)

NORTHERN STATES POWER COMPANY

MONTICELLO NUCLEAR GENERATING PLANT

DOCKET NO. 50-263

This document contains proprietary information pursuant to Title 10 of the *Code of Federal Regulations* Section 2.390.

Proprietary information is identified by text enclosed within double brackets ([[ ]]).

ADAMS Accession No. ML14087A013

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REGARDING LICENSE AMENDMENT NO. 180 TO

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FOR MAXIMUM EXTENDED LOAD LINE LIMIT ANALYSIS PLUS (MELLLA+)

NORTHERN STATES POWER COMPANY

MONTICELLO NUCLEAR GENERATING PLANT

DOCKET NO. 50-263

1.0 INTRODUCTION

By letter dated January 21, 2010[1], as supplemented by additional letters[2], Northern States Power Company, a Minnesota corporation (NSPM or the licensee), submitted an application to amend the Monticello Nuclear Generating Plant (MNGP) technical specifications (TSs). The license amendment request (LAR) proposes a revision to the MNGP TSs to allow plant operation from the currently licensed Maximum Extended Load Line Limit Analysis (MELLLA) domain to operation in the expanded MELLLA Plus (MELLLA+) domain under the previously approved Extended Power Uprate (EPU) conditions of 2004 megawatts thermal (MWt) rated core thermal power.

The supplemental letters received from March 4, 2010, to February 24, 2014, contained clarifying information that did not expand the scope of the application as originally noticed, and did not change the NRC staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on September 21, 2010 (75 FR 57527).

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[1] Agencywide Documents Access and Management System (ADAMS) Accession No. ML100280558.

[2] March 4, 2010 (ADAMS Accession No. ML100710445); September 28, 2010 (ADAMS Accession No. ML102790375); November 11, 2011 (ADAMS Accession No. ML11321A332); June 27, 2012 (ADAMS Accession No. ML12192A104); September 28, 2012 (ADAMS Accession No. ML12276A057); November 30, 2012 (ADAMS Accession Nos. ML123380345 and ML12340A432); December 21, 2012 (ADAMS Accession No. ML13002A261); March 21, 2013 (ADAMS Accession No. ML13085A033); May 13, 2013 (ADAMS Accession No. ML13134A301); June 26, 2013 (ADAMS Accession No. ML13191B126 and ML13191B128); July 8, 2013 (ADAMS Accession No. ML13191A568); July 31, 2013 (ADAMS Accession No. ML13217A373); August 14, 2013 (ADAMS Accession No. ML13246A080); October 4, 2013 (ADAMS Accession No. ML13282A122); December 20, 2013 (ADAMS Accession No. ML13358A357); and February 24, 2014 (ADAMS Accession No. ML14057A526).

2.0 REGULATORY EVALUATION

2.1 Introduction

In its January 21, 2010, application to operate MNGP in the MELLLA+ expanded operating domain, NSPM submitted Licensing Topical Report (LTR) NEDC-33435P, Revision 1, "Safety Analysis Report [SAR] for Monticello Maximum Extended Load Line Limit Analysis Plus" (Reference 1). The proposed MELLLA+ domain supports operation of MNGP at EPU rated core thermal power with rated core flow as low as 80 percent. This 20 percent flow-control window provides a reduced number of control rod movements than in the MELLLA domain, providing greater operating flexibility. The NRC staff's review is based on the submitted LTR, the information obtained during a number of meetings and conference calls with the licensee, and formal requests for additional information (RAI). To evaluate the impact of operation in the expanded operating domain, the NRC staff performed this review using relevant sections of the review guidance contained in Review Standard 001 (RS-001), Revision 0, "Review Standard for Extended Power Uprates," (Reference 2); relevant sections of the Standard Review Plan (Reference 3); and the findings of the staff's MELLLA+ Safety Evaluation Report (SER), NEDC-33006P-A, Revision 3, "General Electric [GE] Boiling Water Reactor Maximum Extended Load Line Limit Analysis Plus" (Reference 5).

In NEDC-33435P, Rev. 1 (Reference 1), the licensee documents the results of all significant safety evaluations (SEs) performed to justify the expansion of the core flow operating domain for MNGP to MELLLA+. These analyses support operation of MNGP at the post-EPU, current licensed thermal power (CLTP) of 2004 MWt with core flow as low as 80 percent (%) of rated.

These analyses are based on the approved methodologies described in NEDC-33006P-A (Reference 5) and NEDC-33173P, "Applicability of GE Methods to Expanded Operating Domains" (Reference 6). All limitations from the approved methodology have been addressed.

The main conclusion from this review is that the broadening of the MNGP operating domain by lowering the flow at high powers without additional limitations would reduce the safety margin, but the solutions proposed in the MNGP Safety Analysis Report (SAR) are technically acceptable to satisfy the regulatory criteria. The following solutions are proposed to maintain the same safety margin under MELLLA+ as under the current licensed thermal power:

- Operation in the MELLLA+ domain will preclude the flexibility of having safety relief valves (SRV) out of service. This restriction is implemented through administrative controls and is necessary to demonstrate compliance to peak vessel pressure limits during anticipated transient without scram (ATWS) events.

- Feedwater heater out-of-service operation will not be allowed in the MELLLA+ domain because analyses have not been performed to demonstrate compliance with applicable criteria under these conditions.

- Single-loop operation (SLO) is not allowed in the MELLLA+ domain.

- The Linear Heat Generation Rate (LHGR) setdown value will be increased an additional 2.3 percent (from 10 percent under EPU conditions to a MELLLA+ setdown of 12.3 percent) in the MELLLA+ domain to maintain equivalent Peak Clad Temperature (PCT) performance during loss-of-coolant accident (LOCA) events. This setdown value will be implemented in the core operating limit report (COLR) and confirmed for future cycles.

- The Maximum Average Planar LHGR (MAPLHGR) limit will be reduced by 2.6 percent for operation in the MELLLA+ domain to maintain equivalent PCT performance during LOCA events. This setdown will be implemented in the COLR and confirmed for future cycles.

- To provide additional protection against spurious, noise-induced scrams of the Detect and Suppress Solution - Confirmation Density (DSS-CD) system, the amplitude discriminator setpoint (S_{AD}) will be increased **[[ ]]**. To compensate for the increased setpoint value, **[[ ]]**.

The licensee's analyses results are summarized in Table 9-1 of NEDC-33435P, Revision 1 (Reference 1), and indicate that the limiting Anticipated Operational Occurrences (AOOs) result in larger delta-CPR when initiated at nominal conditions than inside the MELLLA+ domain; therefore, additional OLPMCPR margin is not required for operation in the MELLLA+ domain.

The NRC staff placed special emphasis on its review of ATWS, and particularly ATWS with instability (ATWSI). The staff's review includes a review of code-features and confirmatory calculations. Specifically, the staff performed preliminary reviews of the TRACG04 methodologies for the determination of the minimum temperature for stable film boiling (T_{min}) and the quenching model for post-T_{min} operating conditions. The review of the two TRACG04 methodologies has not been completed at this time; however, the application of TRACG04 for ATWSI calculation at MNGP is acceptable because the licensee has demonstrated that the T_{min} criteria is not reached at MNGP even when using the most conservative version of the T_{min} correlation. Thus, the MNGP TRACG04 ATWSI calculations essentially have the T_{min} and quench models "turned off" and review of the T_{min} and quenching methodologies is not required to guarantee that MNGP satisfies the core coolability ATWS criteria.

The MNGP MELLLA+ ATWSI calculation satisfies the ATWS acceptance criteria (Reference 3), Section 15.8), in part by taking two deviations from the standard methodology in the MELLLA+ SER, NEDC-33006P-A (Reference 5):

- Operator actions to reduce reactor vessel water have been assumed to occur within 90 seconds of the ATWS initiation. This is faster than the recommended value of 120 seconds in NEDC-33006P-A, and faster than the conservative time used for past ATWS analyses of 250 seconds.

- The peak rod power has been set at 95 percent of the LHGR limit. In past ATWS analyses, peak rod power was set at LHGR limits.

The licensee proposed the following two deviations, which the NRC staff finds acceptable because:

- The licensee provided in its response to RAI 9/11/13-1 (Reference 17) a commitment to train and test licensed reactor operators to initiate ATWS actions within the allotted time, and an adequate margin exists to mitigate ATWSI oscillations in the MELLLA+ domain. Calculations indicate that ATWS acceptance criteria continue to be satisfied even in the presence of unstable power oscillations when the MNGP-specific timing for operator actions is used.

- The licensee reviewed MNGP design limits and verified that the core is designed with more than 5 percent margin to the LHGR limit (typically a 10 percent margin); thus, the 95 percent setting is conservative for an ATWS best-estimate calculation.

The NRC staff determined that the licensee adequately accounted for the effects of the proposed MELLLA+ operating domain extension on the nuclear design and demonstrated that the fuel design limits will not be exceeded during normal or anticipated operational transients, and that the effects of postulated reactivity accidents will not cause significant damage to the pressure vessel or impair the capability to cool the core.

The NRC staff considered the following regulatory requirements:

- Title 10 of the *Code of Federal Regulations* (10 CFR), Section 50.36, "Technical specifications," in which the Commission established its regulatory requirements related to the contents of the TS. Specifically, 10 CFR 50.36(a)(1) states that "Each applicant for a license authorizing operation of a production or utilization facility shall include in its application proposed technical specifications in accordance with the requirements of this section."

- In 10 CFR 50.36(c)(3) it states that, "Surveillance requirements are requirements relating to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions of operation will be met."

- Title 10 of the *Code of Federal Regulations* (10 CFR), Section 50.46, “Acceptance criteria for emergency core cooling systems for light-water nuclear power sources,” which establishes standards for the calculation of emergency core cooling system (ECCS) performance and acceptance criteria for that calculated performance.

- 10 CFR 50.62, “Requirements for reduction of risk from anticipated transients without scram (ATWS) events for light-water-cooled nuclear power plants,” insofar as licensees provide the means to address an ATWS event, an anticipated operational occurrence defined in Appendix A of this part, followed by the failure of the reactor trip portion of the protection system specified in GDC 20 of Appendix A.

- 10 CFR 50.63, “Loss of all alternating current power,” insofar as it requires that the plant withstand and recover from a station blackout (SBO) event of a specified duration.

- Appendix A to 10 CFR Part 50, "General Design Criteria [GDC] for Nuclear Power Plants," was published in 1971. The applicable MNGP principal design criteria predate these Appendix A criteria. The MNGP principal design criteria are listed in the MNGP Updated Safety Analysis Report (USAR), Section 1.2, "Principal Design Criteria." In 1967, the Atomic Energy Commission (AEC) issued for public comment a revised set of proposed GDC (32 FR 10213, dated July 11, 1967). An evaluation comparing the MNGP design basis to the AEC-proposed GDCs of 1967 is presented in the MNGP USAR, Appendix E, "Plant Comparative Evaluation with the Proposed AEC 70 Design Criteria."

  The associated Appendix A GDCs applicable to the NRC staff's review are further discussed in the respective sections of this safety evaluation (SE).

- Appendix K to 10 CFR Part 50, "ECCS Evaluation Models," which establishes required and acceptable features of evaluation models for heat removal by the ECCS after the blowdown phase of a LOCA.

2.2 Reactor Systems

2.2.1 Summary of Technical Information

The MNGP SAR, NEDC-33435P, Revision 1 (Reference 1), contains information divided into the following eleven sections:

- Section 1.0 - Introduction
- Section 2.0 - Reactor Core and Fuel Performance
- Section 3.0 - Reactor Coolant and Connected Systems
- Section 4.0 - Engineered Safety Features
- Section 5.0 - Instrumentation and Control
- Section 6.0 - Electrical Power and Auxiliary Systems
- Section 7.0 - Power Conversion Systems
- Section 8.0 - Radwaste Systems and Radiation Sources
- Section 9.0 - Reactor Safety Performance Evaluations
- Section 10.0 - Other Evaluations
- Section 11.0 - Licensing Evaluations

The SAR also includes four appendices that evaluate the disposition of limitations of applicable SERs. A complete listing of the required limitations and conditions is presented in Appendices A, B, C, and D of the SAR. These appendices address the limitations from the MELLLA+ SER (Reference 5), the Methods SER 9 (Reference 6), the DSS-CD SER (Reference 7), and the DSS-CD TRACG SER (Reference 8).

This SE addresses only the Reactor System Sections, which correspond to Section 2.8 of the "Review Standard for Extended Power Uprates," RS-001 (Reference 2). Therefore, the SAR sections addressed in this SE – see below – are Sections 1.0, 2.0, 3.0, 4.0, and 9.0. Information in these five SAR sections is summarized below.

Section 1.0 - Introduction

Section 1 of the SAR describes the report approach, as well as the differences between generic and plant-specific assessments. Generic assessments are those safety evaluations that can be disposed of by either: (1) a reference bounding calculation, (2) demonstration negligible impact of MELLLA+ operation, (3) or deferring to the plant-specific analyses during the reload process. Plant specific evaluations are provided for those items where a generic assessment is not applicable.

The licensee committed to supplement the SAR with the fuel and cycle dependent analysis including the plant-specific thermal limits assessment. The NRC staff reviewed the Supplemental Reload Licensing Report (SRLR) (Reference 14) and identified that the analyses documented in the SRLR support the conclusions in this SE. The final SRLR has been submitted by MNGP and confirms the conclusions.

Table 1-1 of the SAR lists all the computer codes used for this evaluation. Figure 1-1 of the SAR (reproduced below as Figure 1) defines the MELLLA+ operating domain. **[[**

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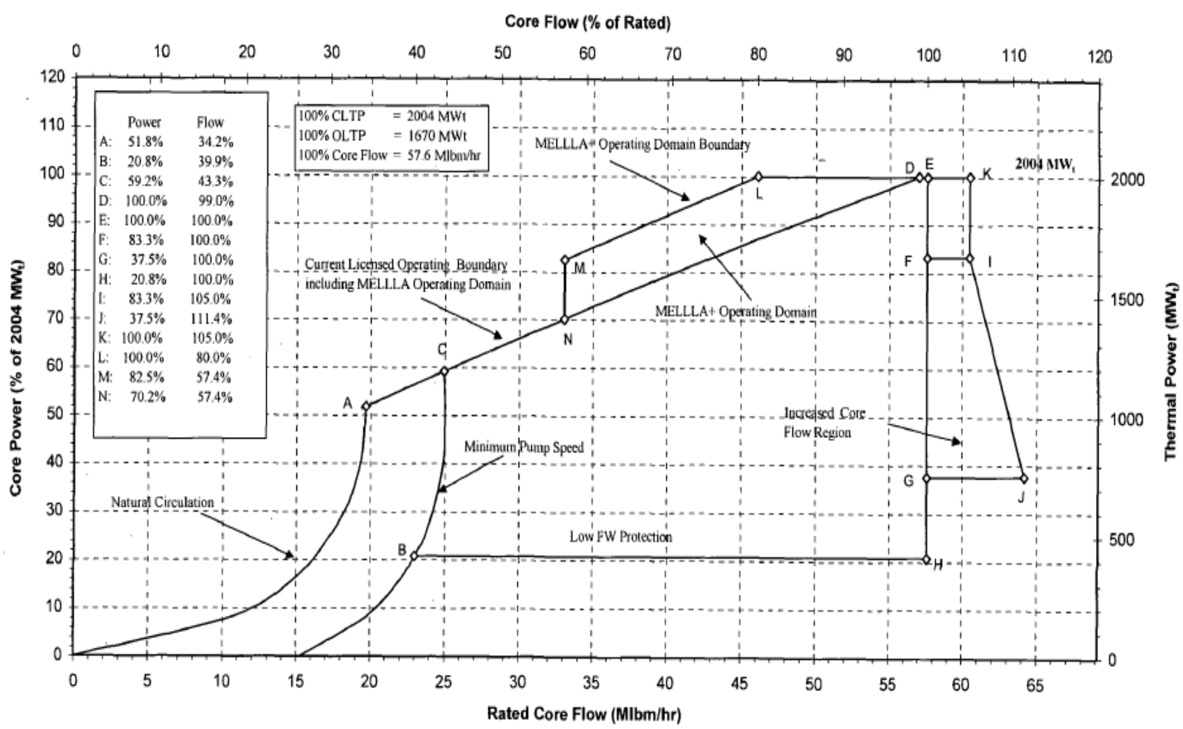


Figure 1 - MELLLA+ Operating Domain

Section 1.2.4 of the SAR describes the allowed operational enhancements, which are covered by the approved MELLLA+ SER. The following enhancements are allowed:

1. Increased core flow
2. Average Power Range Monitor and Rod Block Monitor Technical Specifications (APRM/RBM/ARTS) improvements

The following enhancements are not allowed in the MELLLA+ domain:

1. Safety Relief Valve (SRV) Out-of-Service
2. Feedwater Heater Out-of-Service
3. Single-Loop Operation

Section 2.0 - Reactor Core and Fuel Performance

Following the approved methodology in the MELLLA+ SER (Reference 5), Section 2.0 of the SAR evaluates MNGP on a plant-specific basis for the following topics:

1. Fuel Design and Operation

    a. Fuel Product Line Design
    b. Core Design
    c. Fuel Thermal Margin Monitoring Threshold

2. Thermal Limits Assessment

    a. Safety Limit MCPR
    b. Operating Limit MCPR
    c. MAPLHGR Limit
    d. LHGR Limit

3. Reactivity Characteristics

    a. Hot Excess Reactivity
    b. Strong Rod Out Shutdown Margin
    c. Standby Liquid Control System (SLCS) Shutdown Margin

4. Stability

    a. DSS-CD Setpoints
    b. Armed Region
    c. Backup Stability Protection (BSP)

5. Reactivity Control

    a. Scram Time Response
    b. Control Rod Drive (CRD) Positioning and Cooling

c. CRD Integrity

No additional plant-specific evaluations are required for the reactor core area.

The SAR states that because MNGP currently uses only GE14 fuel, the following limitations and conditions from the Methods LTR SER, M+LTR SER, and DSS-CD SER are not applicable to the MNGP:

1. Methods LTR SER Limitations and Conditions

    a. APPLICATION OF 10 WEIGHT PERCENT GD: Limitation and Condition 9.13
    b. MIXED CORE METHOD 1: Limitation and Condition 9.21
    c. MIXED CORE METHOD 2: Limitation and Condition 9.22

2. M+ LTR SER Limitations and Conditions

    a. CONCURRENT CHANGES: Limitation and Condition 12.3.d, 12.3.e, and 12.3.f
    b. APPENDIX A RAI 14-9: Limitation and Condition 12.23.6
    c. APPENDIX A RAI 14-10: Limitation and Condition 12.23.7

3. DSS-CD SER Limitations and Conditions

    a. Limitation and Condition 4.5

Section 2 of the SAR addresses additional limitations and conditions related to reactor core and fuel performance, including the following:

1. TGBLA/PANAC Version. The most recent version of the time of analyses was used.

2. M+LTR SER Limitation and Condition 12.24.1. The TRACG supporting analyses used the detailed calculation of bundle flow, as required by this condition.

Section 2 of the SAR provides power distributions, LHGR, and critical power ratios (CPR) at three points during the operating cycle. In response to RAI-02, GEH provided similar information for a representative EPU (pre-MELLLA+) cycle. The RAI response valuation in Appendix A of this SE provides a summary comparison of the two cores (see Table A-2). The NRC staff finds the differences are not significant.

The responses to RAI-03 through RAI-10 provide additional details about the transients analyzed to confirm the generic disposition of the above items for MNGP.

Section 3.0 - Reactor Coolant and Connected Systems

Following the approved methodology in the MELLLA+ SER (Reference 5), Section 3.0 of the SAR evaluates MNGP on a plant-specific basis for the following topics:

1. Nuclear System Pressure Relief And Overpressure Protection

    a. Overpressure Relief Capacity

The evaluation concludes that, for MNGP, the limiting overpressure event is the Main Steam Isolation Valve Closure with Scram on High Flux. The peak RPV bottom head pressure is unchanged and remains less than the ASME limit of 1375 pounds per square inch gauge (psig) for AOOs. The SRV tolerance assumed in the MNGP ASME overpressure event analysis is 3 percent. The NRC staff finds this tolerance to be consistent with the actual SRV performance testing at MNGP.

For non-AOO transients, the ATWS analysis concludes that an increase in the number of SRVs credited in the analysis is required to demonstrate acceptable results; this is accomplished by eliminating the flexibility of having a single SRV out of service during operation in the MELLLA+ region.

2. Reactor Vessel

    a. Fracture Toughness

MELLLA+ operation results in slightly higher fluxes at the vessel because of the reduced moderation. These higher fluxes are comparable to the ones approved for the EPU license amendment. The SAR estimates that the change in peak fluence is **[[ ]]** negligible, and does not require changes to the current operating practices.

3. Reactor Internals

    a. Reactor Internals Pressure Differences (Acoustic and Flow-Induced Loads) for faulted conditions.

The plant-specific LOCA analysis concludes that the loads **[[ ]]** as a result of MELLLA+ operating domain expansion.

    b. Reactor Internals Structural Evaluation for Normal, Upset, and Emergency Conditions

The analysis concludes that there is **[[ ]]**.

    c. Reactor Internals Structural Evaluation for Faulted Conditions

[[ ]].

    d. Steam Dryer Separator Performance

The analysis indicates that there will be increased moisture in the steam dryer separators, but it is within acceptable steam separator-dryer performance. The separator moisture content is monitored during operation to ensure it remains within acceptable limits.

Following the approved methodology in the MELLLA+ SER (Reference 5), Section 3.0 of the SAR evaluates MNGP on a plant-specific basis for the following topics:

1. Nuclear System Pressure Relief And Overpressure Protection
    a. Flow-Induced Vibration
2. Reactor Vessel
    a. Reactor Vessel Structural Evaluation
3. Reactor Internals
    a. Fuel Assembly and Control Rod Guide Tube Lift Forces
    b. Reactor Internals Pressure Differences for Normal, Upset, and Emergency Conditions
4. Flow Induced Vibration (FIV)
    a. Piping FIV Evaluation
    b. RPV Internals FIV Evaluation
5. Piping Evaluation
    a. Reactor Coolant Pressure Boundary Piping
    b. Balance-of-Plant Piping
6. Reactor Recirculation System
    a. System Evaluation
    b. Net Positive Suction Head (NPSH)
    c. Single-Loop Operation
    d. Flow Mismatch
7. Main Steam Line Flow Restrictors
    a. Structural Integrity
8. Main Steam Isolation Valves
    a. Isolation Performance
    b. Valve Pressure Drop

9. Reactor Core Isolation Cooling

    a. System Hardware
    b. System Initiation
    c. Net Positive Suction Head
    d. Inventory Makeup Level Margin to Top of Active Fuel (TAF)

10. Residual Heat Removal System

    a. Low Pressure Coolant Injection Mode
    b. Suppression Pool and Containment Spray Cooling Modes
    c. Shutdown Cooling Mode
    d. Steam Condensing Mode (not applicable to MNGP)
    e. Fuel Pool Cooling Assist

11. Reactor Water Cleanup System

    a. System Performance
    b. Containment Isolation

<u>Section 4.0 - Engineered Safety Features</u>

Following the approved methodology in the MELLLA+ SER (Reference 5), Section 4.0 of the SAR evaluates MNGP on a plant-specific basis for the following topics:

1. Containment System Performance

    a. Short-Term Pressure and Temperature Response
    b. Containment Dynamic Loads, LOCA Loads & Sub-compartment Pressurization

2. Emergency Core Cooling System Performance

    a. Large Break Peak Clad Temperature

The licensee's evaluation concludes that MELLLA+ primarily affects the first peak PCT, therefore the limiting single failure is not affected by MELLLA+. The limiting Appendix K large break single failure for MNGP remains the LPCI Injection Valve (LPCIIV) failure. The evaluation concludes that a 12.3 percent LHGR setdown is required to maintain PCT performance, an increase of 2.3 percent with respect to the 10 percent setdown required for MELLLA+ operation. LOCA analyses are presented to demonstrate the PCT performance.

    b. Small Break Peak Clad Temperature. No small break calculations were performed at MELLLA+ conditions because the PCT results at full flow are significantly lower than the limiting Appendix K PCT numbers.

Following the approved methodology in the MELLLA+ SER (Reference 5), Section 3.0 of the SAR evaluates MNGP generically and confirms the following topics:

1. Containment System Performance

    a. Long-Term Suppression Pool Temperature Response
    b. Containment Dynamic Loads, SRV Loads
    c. Containment Isolation
    d. Generic Letter 89-10
    e. Generic Letter 89-16
    f. Generic Letter 95-07
    g. Generic Letter 96-06

2. Emergency Core Cooling Systems

    a. High Pressure Coolant Injection
    b. Core Spray
    c. Low Pressure Coolant Injection Mode of the RHR System
    d. Automatic Depressurization System
    e. ECCS Net Positive Suction Head (contingent on resolution of the use of containment accident pressure)

3. Emergency Core Cooling System Performance

    a. Local Cladding Oxidation
    b. Core Wide Metal Water Reaction
    c. Coolable Geometry
    d. Long-Term Cooling
    e. Flow Mismatch Limits

4. Main Control Room Atmosphere Control System

    a. Iodine Intake

5. Standby Gas Treatment System

    a. Flow Capacity
    b. Iodine Removal Capacity

6. Main Steam Isolation Valve Leakage Control System. (Not applicable to MNGP)

7. Post-LOCA Combustible Gas Control System

    a. Hydrogen and Oxygen Production

<u>Section 9.0 - Reactor Safety Performance Evaluations</u>

Following the approved methodology in the MELLLA+ SER (Reference 5), Section 9.0 of the SAR evaluates MNGP on a plant-specific basis for the following topics:

1. Anticipated Operational Occurrences

    a. Fuel Thermal Margin Events Table 9-1 of the SAR presents the calculated response for the six limiting AOO's (TTWBP, TTNBP, LRNBP, FWCF, HPCIL8, and LFWH). A comparison is provided between the CLTP/80 percent flow point and CLTP/105 percent flow conditions. For all cases analyzed, the CLTP/105 percent ICF point is more limiting. The response to RAIs 11 through 13, detailed in Appendix A of this SE, provides additional details.

    b. Power and Flow Dependent Limits

        The evaluation concludes that current LHGR and MAPHGR limits are adequate under MELLLA+ conditions. The response to RAI 14 provides additional details of the slow flow increase transient.

2. Design Basis Accidents And Events Of Radiological Consequence

    a. Control Rod Drop Accident (CRDA). The evaluation concludes that this event is bounded by the analysis for the current licensed operating domain.

3. Special Events

    a. ATWS (Overpressure)
    b. ATWS (Suppression Pool Temperature and Containment pressure)
    c. ATWS (Peak Cladding Temperature and Oxidation)
    d. ATWS with Core Instability

Following the approved methodology in the MELLLA+ SER (Reference 5), Section 9.0 of the SAR evaluates MNGP on a plant-specific basis for the following topics:

1. Anticipated Operational Occurrences

    a. Non-limiting Events

2. Design Basis Accidents And Events Of Radiological Consequence

    a. Main Steam Line Break Accident (MSLBA)
    b. Loss of Coolant Accident (LOCA)
    c. Fuel Handling Accident (FHA)

3. Special Events

    a. Station Blackout

2.2.2 Regulatory Review Criteria

The NRC staff regulatory criteria for this review are based on the following sources:

1. Review Standard 001 (RS-001) (Reference 2), and

2. Relevant sections of the Standard Review Plan (SRP) (Reference 3), specifically:

   a. SRP Section 4 and, in particular, Sections 4.2 “Fuel System Design;” 4.3 “Nuclear Design;” 4.4 “Thermal and Hydraulic Design;” and 4.6 “Emergency Systems.”

   b. SRP Section 15 and, in particular, Sections 15.1 “Increase in heat removal by the secondary system;” 15.2 “Decrease in heat removal by the secondary system;” 15.3 “Decrease in RCS flow rate;” 15.4 “Reactivity and power distribution anomalies;” 15.5 “Increase in reactor coolant inventory;” 15.6 “Decrease in reactor coolant inventory;” 15.7 “Radioactive release from a subsystem or component;” 15.8 “Anticipated Transients without Scram;” and 15.9 “Boiling Water Reactor Stability.”

3. Title 10 of the *Code of Federal Regulations* (10 CFR), Section 50.46, “Acceptance criteria for emergency core cooling systems for light-water nuclear power sources,” which establishes standards for the calculation of emergency core cooling system (ECCS) performance and acceptance criteria for that calculated performance.

4. 10 CFR Section 50.62, “Requirements for reduction of risk from anticipated transients without scram (ATWS) events for light-water-cooled nuclear power plants,” insofar as licensees provide the means to address an ATWS event, an anticipated operational occurrence defined in Appendix A of this part, followed by the failure of the reactor trip portion of the protection system specified in GDC 20 of Appendix A.

5. 10 CFR Part 50 (10 CFR 50), Appendix K, “ECCS Evaluation Models,” which establishes required and acceptable features of evaluation models for heat removal by the ECCS after the blowdown phase of a LOCA.

The NRC staff acceptance criteria are based on the following GDCs in Appendix A of 10 CFR Part 50:

1. GDC-4, “Environmental and dynamic effects design bases,” insofar as it requires that structures, systems, and components (SSCs) important to safety must be protected against dynamic effects associated with flow instabilities and loads such as those resulting from water hammer.

2. GDC-5, “Sharing of structures, systems, and components,” insofar as it requires that SSCs important to safety not be shared among nuclear power units unless it can be shown that sharing will not significantly impair their ability to perform their safety functions.

3. GDC-10, “Reactor design,” insofar as the Reactor Protection System must be designed with appropriate margin to assure that specified fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences.

4. GDC-11, “Reactor inherent protection,” insofar as the reactor core must be designed so that the net effect of the prompt inherent nuclear feedback characteristics tends to compensate for a rapid increase in reactivity.

5. GDC-12, “Suppression of reactor power oscillations,” insofar as unstable oscillations with the potential of violating specified acceptable fuel design limits (SAFDLs) are either not possible or can be reliably and readily detected and suppressed.

6. GDC-13, “Instrumentation and control,” insofar as instrumentation and controls must be provided to monitor variables and systems affecting the fission process over anticipated ranges for normal operation, AOOs and accident conditions, and to maintain the variables and systems within prescribed operating ranges.

7. GDC-15, “Reactor coolant system design,” insofar as it requires that the RCS and associated auxiliary, control, and protection systems be designed with sufficient margin to assure that the design conditions of the RCPB are not exceeded during any condition of normal operation, including AOOs

8. GDC-20, “Protection system functions,” insofar as the Reactor Protection System must be designed to initiate the reactivity control systems automatically to assure that acceptable fuel design limits are not exceeded as a result of AOOs, and to sense accident conditions and initiate the operation of systems and components important to safety.

9. GDC-23, “Protection system failure modes,” insofar as it requires that the protection system be designed to fail into a safe state.

10. GDC-25, “Protection system requirements for reactivity control malfunctions,” insofar as the Reactor Protection System must be designed to assure that SAFDLs are not exceeded for any single malfunction of the reactivity control systems.

11. GDC-26, “Reactivity control system redundancy and capability,” insofar as two independent reactivity control systems must be provided, with both systems capable of reliably controlling the rate of reactivity changes resulting from planned, normal power changes, including AOOs, so that SAFDLs are not exceeded.

12. GDC-28, “Reactivity limits,” insofar as the reactivity control systems must be designed with appropriate limits on the potential amount and rate of reactivity increase to assure that the effects of postulated reactivity accidents can neither (1) result in damage to the reactor coolant pressure boundary (RCPB) greater than limited local yielding nor (2) sufficiently disturb the core, its support structures, or other reactor vessel internals so as to significantly impair the capability to cool the core.

13. GDC-31, "Fracture prevention of reactor coolant pressure boundary," insofar as it requires that the RCPB be designed with sufficient margin to assure that it behaves in a non-brittle manner and that the probability of rapidly propagating fracture is minimized.

14. GDC-33, "Reactor coolant makeup," insofar as it requires that a system to provide reactor coolant makeup for protection against small breaks in the RCPB be provided so the fuel design limits are not exceeded.

15. GDC-35, "Emergency core cooling," insofar as an emergency system to provide abundant emergency core cooling must be provided to transfer heat from the reactor core following any LOCA.

16. GDC-54, "Piping systems penetrating containment," insofar as it requires that piping systems penetrating containment be designed with the capability to periodically test the operability of the isolation valves to determine if valve leakage is within acceptable limits.

The GDCs discussed above are those currently specified in Appendix A of 10 CFR Part 50. The applicable MNGP principal design criteria predate these Appendix A criteria. The MNGP principal design criteria are listed in the MNGP Updated Safety Analysis Report (USAR), Section 1.2, "Principal Design Criteria." In 1967, the Atomic Energy Commission (AEC) issued for public comment a revised set of proposed GDC (32 FR 10213, dated July 11, 1967). An evaluation comparing the MNGP design basis and Appendix A GDCs to the AEC-proposed GDCs is presented in the MNGP USAR, Appendix E, "Plant Comparative Evaluation with the Proposed AEC 70 Design Criteria."

- The current GDC-4 is comparable to AEC-proposed GDCs 40 and 42
- The current GDC-5 is comparable to AEC-proposed GDC 4.
- The current GDC-10 is comparable to AEC-proposed GDC 6, as further described in USAR Section 14.4.
- The current GDC-11 is comparable to AEC-proposed GDC 8.
- The current GDC-12 is comparable to AEC-proposed GDC 7, as further described in USAR Section 14.6.
- The current GDC-13 is comparable to AEC-proposed GDC 12, as further described in USAR Section 14.7.4.
- The current GDC-15 is comparable to AEC-proposed GDC 9, as further described in USAR Section 14.4.
- The current GDC-20 is comparable to AEC-proposed GDCs 14 and 15, as further described in USAR Section 14.4.
- The current GDC-23 is comparable to AEC-proposed GDC 26.
- The current GDC-25 is comparable to AEC-proposed GDC 31, as further described in USAR Section 14.4.

- The current GDC-26 is comparable to AEC-proposed GDC 27, as further described in USAR Section 14.4.
- The current GDC-28 is comparable to AEC-proposed GDC-32.
- The current GDC-31 is comparable to AEC-proposed GDCs 33, 34 and 35.
- The current GDC-33 is comparable to AEC-proposed GDC-37.
- The current GDC-35 is comparable to AEC-proposed GDCs 37, 42 and 44.
- The current GDC-54 is comparable to AEC-proposed GDC-57.

2.3 Containment and Ventilation

2.3.1 Regulatory Review Criteria

The NRC staff acceptance criteria are based on the following GDC in Appendix A of 10 CFR Part 50:

1. GDC-4, "Environmental and dynamic effects design bases," insofar as it requires that SSCs important to safety be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents and that such SSCs be protected against dynamic effects.

2. GDC-16, "Containment design," insofar as it requires that the containment and associated systems be designed to establish an essentially leak tight barrier against the uncontrolled release of radioactivity to the environment, and to assure that the containment design conditions important to safety are not exceeded as long as postulated accident conditions require.

3. GDC-19, "Control room," insofar as it requires that adequate radiation protection be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of five rem [rem is defined in 10 CFR 20.1004, "Units of radiation dose] whole body, or its equivalent to any part of the body, for the duration of the accident.

4. GDC-38, "Containment Heat Removal," insofar as it requires that a containment heat removal system be provided and that its function shall be to reduce rapidly the containment pressure and temperature following a LOCA and maintain them at acceptably low levels.

5. GDC-41, "Containment atmosphere cleanup," insofar as it requires systems to (1) control fission products, hydrogen, oxygen and other substances which may be released into the reactor containment; (2) reduce the concentration and quality of fission products released to the environment following postulated accidents; and (3) control the concentration of hydrogen or oxygen and other substances in the containment

atmosphere following postulated accidents to assure that containment integrity is maintained.

6. GDC-50, "Containment Design Basis," insofar as it requires that the containment and its associated heat removal systems be designed so that the containment structure can accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated temperature and pressure conditions resulting from any LOCA.

The NRC staff acceptance criteria are also based on 10 CFR 50.44, "Combustible gas control for nuclear power reactors," insofar as it requires that plants be provided with the capability of controlling combustible gas concentrations in the containment atmosphere.

Specific review criteria is contained in NUREG-0800, Sections 6.2.1.1.C, 6.2.1.2, 6.2.5, and 6.2.2, as supplemented by Draft Guide 1107, "Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident."

The GDCs discussed above are those currently specified in Appendix A of 10 CFR Part 50. The applicable MNGP principal design criteria predate these Appendix A criteria. The MNGP principal design criteria are listed in the MNGP Updated Safety Analysis Report (USAR), Section 1.2, "Principal Design Criteria." In 1967, the Atomic Energy Commission (AEC) issued for public comment a revised set of proposed GDC (32 FR 10213, dated July 11, 1967). An evaluation comparing the MNGP design basis and Appendix A GDCs to the AEC-proposed GDCs is presented in the MNGP USAR, Appendix E, "Plant Comparative Evaluation with the Proposed AEC 70 Design Criteria."

- The current GDC-4 is comparable to AEC-proposed GDCs 40 and 42.
- The current GDC-19 is comparable to AEC-proposed GDC 11, as further described in USAR Sections 5.3.5, 6.7.3, 12.3.1.6, and 14.7.
- The current GDC-16 is comparable to AEC-proposed GDC 49.
- The current GDC-38 is comparable to AEC-proposed GDCs 41 and 52.
- The intent of current GDC-41 is described in USAR Section 5.3.4.1.
- The current GDC-50 is comparable to AEC-proposed GDC 49.

3.0 TECHNICAL EVALUATION

3.1 Reactor Systems

The NRC staff used RS-001, "Review Standard for Extended Power Uprates" (Reference 2) as a reference in conducting the MELLLA+ review. Although MELLLA+ is not a power uprate and RS-001 guidance is not wholly applicable, RS-001 provides a good framework for review of this application. The staff recognizes that there are sections in RS-001 that are unnecessary for the MELLLA+ application review. RS-001 specifies the following review areas that were evaluated for the proposed extension of the operating domain:

- Materials and Chemical Engineering
- Mechanical and Civil Engineering
- Electrical Engineering
- Instrumentation and Controls
- Plant Systems
- Containment Review Considerations
- Habitability, Filtration, and Ventilation
- Reactor Systems
- Source Terms and Radiological Consequences Analyses
- Health Physics
- Human Performance
- Power Ascension and Testing Plan
- Risk Evaluation

The NRC staff's SE is based on a partial evaluation of NEDC-33435P, Rev. 1 (Reference 1), that only addresses the impact of the MELLLA+ extended operating domain on Item 8 - Reactor Systems. The staff used only the framework in Matrix 8 of RS-001 to evaluate the applicable portions of NEDC-33435P, Rev. 1.

RS-001 specifies that the following Reactor Systems areas should be reviewed:

- Fuel System Design
- Nuclear Design
- Thermal and Hydraulic Design
- Emergency Systems
- Accident and Transient Analyses
- Fuel Storage

The NRC staff's evaluation of these Reactor Systems areas is provided in the following sections.

3.1.1 Fuel System Design

Regulatory Evaluation

The fuel system consists of arrays of fuel rods, burnable poison rods, spacer grids and springs, end plates, channel boxes, and reactivity control rods. The NRC staff reviewed the fuel system

to ensure that (1) the fuel system is not damaged as a result of normal operation and AOOs, (2) fuel system damage is never so severe as to prevent control rod insertion when it is required, (3) the number of fuel rod failures is not underestimated for postulated accidents, and (4) coolability is always maintained. The NRC staff's review covered fuel system damage mechanisms, limiting values for important parameters, and performance of the fuel system during normal operation, AOOs, and postulated accidents. The NRC's acceptance criteria are based on (1) 10 CFR 50.46, insofar as it establishes standards for the calculation of ECCS performance and acceptance criteria for that calculated performance; (2) GDC-10, insofar as it requires that the reactor core be designed with appropriate margin to assure that SAFDLs are not exceeded during any condition of normal operation, including the effects of AOOs; and (3) GDC-35, insofar as it requires that a system to provide abundant emergency core cooling be provided to transfer heat from the reactor core following any LOCA. Specific review criteria are contained in SRP Section 4.2 and other guidance provided in Matrix 8 of RS-001.

Technical Evaluation

The NRC staff reviewed the impact on the fuel system of the proposed MELLLA+ operating domain extension based on the licensee-provided analyses for normal operation, AOOs, infrequent and special events. The complete staff evaluation of these results is documented below in Section 3.1.5, "Accident and Transient Analyses." As stated in that evaluation, operation at the lower MELLLA+ flows has an impact on transient response, and the effect on fuel is marginally more severe for some events. To mitigate these events, the licensee proposes to use more restrictive setpoints so that the final safety limit minimum critical power ratio (SLMCPR) is maintained constant. The licensee's analyses demonstrate that, with the proposed MNGP MELLLA+ setpoints, significant fuel damage is very unlikely for any AOO or the analyzed infrequent or special events, and core coolability is maintained. Therefore, the NRC staff concludes that the impact on fuel of operation with the more restrictive setpoints at the lower MELLLA+ flows is minimal.

Conclusions

The NRC staff has reviewed the licensee's analyses related to the effects of the proposed operating domain extension on the fuel system design of the fuel assemblies, control systems, and reactor core. The NRC staff concludes that the licensee adequately accounted for the effects of the proposed operating domain extension on the fuel system and demonstrated that (1) fuel system damage is very unlikely as a result of normal plant operation and AOOs, (2) fuel system damage, should it happen, will not be so severe as to prevent control rod insertion when it is required, (3) the number of fuel rod failures has not been underestimated for postulated accidents, and (4) core coolability is likely to be maintained.

Based on the above, the NRC staff concludes that the fuel system and associated analyses will continue to meet the requirements of 10 CFR 50.46, GDC-10, and GDC-35 following implementation of the proposed operating domain extension. Therefore, the NRC staff concludes that the proposed operating domain extension acceptable with respect to the fuel system design.

3.1.2 Nuclear Design

Regulatory Evaluation

The NRC staff reviewed the nuclear design of the fuel assemblies, control systems, and reactor core to ensure that fuel design limits will not be exceeded during normal operation and anticipated operational transients, and that the effects of postulated reactivity accidents will not cause significant damage to the Reactor Coolant Pressure Boundary (RCPB) or impair the capability to cool the core. The NRC staff's review covered core power distribution, reactivity coefficients, reactivity control requirements and control provisions, control rod patterns and reactivity worths, criticality, burnup, and vessel irradiation. The NRC's acceptance criteria are based on (1) GDC-10, insofar as it requires that the reactor core be designed with appropriate margin to assure that SAFDLs are not exceeded during any condition of normal operation, including the effects of AOOs; (2) GDC-11, insofar as it requires that the reactor core be designed so that the net effect of the prompt inherent nuclear feedback characteristics tends to compensate for a rapid increase in reactivity; (3) GDC-12, insofar as it requires that the reactor core be designed to assure that power oscillations, which can result in conditions exceeding SAFDLs, are not possible or can be reliably and readily detected and suppressed; (4) GDC-13, insofar as it requires that instrumentation and controls be provided to monitor variables and systems affecting the fission process over anticipated ranges for normal operation, AOOs and accident conditions, and to maintain the variables and systems within prescribed operating ranges; (5) GDC-20, insofar as it requires that the protection system be designed to initiate the reactivity control systems automatically to assure that acceptable fuel design limits are not exceeded as a result of AOOs and to automatically initiate operation of systems and components important to safety under accident conditions; (6) GDC-25, insofar as it requires that the protection system be designed to assure that SAFDLs are not exceeded for any single malfunction of the reactivity control systems; (7) GDC-26, insofar as it requires that two independent reactivity control systems be provided, with both systems capable of reliably controlling the rate of reactivity changes resulting from planned, normal power changes; and (8) GDC-28, insofar as it requires that the reactivity control systems be designed to assure that the effects of postulated reactivity accidents can neither result in damage to the RCPB greater than limited local yielding, nor disturb the core, its support structures, or other reactor vessel internals so as to significantly impair the capability to cool the core. Specific review criteria are contained in SRP Section 4.3 and other guidance provided in Matrix 8 of RS-001.

Technical Evaluation

Operating Limits

GDC-10 specifies the requirements for core operating limits. GDC-10 is met by operating the plant within established operating limits. The OLMCPR and LHGR limit are designed to protect the fuel during normal operation, as well as during anticipated transients, from exceeding SAFDLs.

The NRC staff reviewed the design changes between the MNGP EPU core design and a reference MELLLA+ core design in terms of its impact on compliance with GDC-10. The staff

notes that the core and fuel design remain unchanged, and a full load of GE14 fuel is used for both cores.

The SLMCPR is calculated based on the actual core loading pattern for each reload core, and the results are reported in the SRLR. In the event that the cycle-specific SLMCPR is not bounded by the current MNGP TS value, the licensee must implement a license amendment to change the TS. As required by the MELLLA+ SER (Reference 5), the SLMCPR is calculated at different operating conditions for every reload core. The specified conditions include the following: 120%OLTP/100%Flow, 120%OLTP/80%Flow, 120%OLTP/105%Flow, and 99%OLTP/57.4%Flow. The calculated SLMCPR values include the 0.03 adder required by the Methods SER (Reference 6) for operation in the MELLLA+ domain. The licensee will evaluate the final SLMCPR value for the reload core prior to MELLLA+ implementation and provide this information to the NRC as specified in Reference 5. The final SLMCPR value is also referenced in the Cycle 27 COLR, which is provided to the NRC in accordance TS 5.6.3, "Core Operating Limits Report (COLR)."

The Operating Limit Minimum Critical Power Ratio (OLMCPR) is calculated by adding the change in MCPR due to the limiting AOO event to the SLMCPR. The OLMCPR for MNGP is determined on a cycle-specific basis from the results of the reload transient analysis, which are documented in the SRLR. Because TRACG04 methods (as opposed to PANACEA/ODYN/ISCORT/TASC) were used for the MELLLA+ Cycle 27 analyses, the 0.01 adder to the resulting OLMCPR required by the Methods SER (Reference 6), when the old methods are used, was not applied. The final value of the OLMCPR is documented in the COLR. Based on the generic results documented in the MELLLA+ SER (Reference 5), and the reference transient analyses documented in Section 9 of the MNGP SAR (Reference 1), the OLMCPR is not expected to change significantly for MELLLA+ operation.

The LHGR limits ensure that the plant does not exceed fuel thermal-mechanical design limits. These limits are evaluated in the cycle-specific reload analysis and documented in the COLR. LOCA analysis confirmed that MELLLA+ operation has an impact on the PCT. To maintain the same Appendix K results for MELLLA+ as for EPU, MNGP proposes to increase the LHGR setdown value from 10 percent for EPU conditions to 12.3 percent under MELLLA+.

The MAPLHGR operating limit is calculated for each reload fuel bundle design. The limits are documented in the cycle-specific COLR.

Section 4.3 of the SAR (Reference 1) presents results for a LOCA analysis at different initial conditions. The evaluation concludes that MELLLA+ primarily affects the first peak PCT, therefore the limiting single failure is not affected by MELLLA+, and the limiting Appendix K large break single failure for MNGP remains the LPCIIV failure. These LOCA analyses demonstrate that, with the proposed 12.3 percent LHGR setdown, the limiting PCT values at MELLLA+ are bounded by those under EPU.

Monitoring and Control

GDC-13 specifies the requirements for instrumentation to monitor variables affecting the fission process. Maneuvering within the MELLLA+ operating domain is performed by either controlling

the recirculation flow, or moving control rods. GDC-13 requires that instrumentation be provided to ensure that the operation is within prescribed operating ranges.

The design changes to incorporate MELLLA+ do not include any changes to the neutron monitoring system (NMS) or the flow instrumentation. Nevertheless, the NRC staff reviewed the effects of operation in the expanded domain on instrumentation performance, and therefore, the adequacy of the NMS to meet the requirements of GDC-13.

At the MELLLA+ low corner (Point M of Figure 1), the power-to-flow ratio is maximized and there is the potential to encounter void formation in the bypass region. In RAI-03, the NRC staff requested that the licensee provide an assessment of bypass void formation at point M. Furthermore, the staff requested that the licensee determine the effects of bypass void formation on local power range monitor (LPRM). The evaluation was performed and the bypass void fraction is expected to be less than 3.4 percent. This value is calculated using the ISCOR hot channel methodology computer code, which is conservative because it neglects cross flow between bundles in the bypass region; thus, the bypass voids are expected to be lower than the 3.4 percent value calculated. Proper LPRM calibration requires that the bypass void be maintained to a value less than 5 percent; as such, the NRC staff concludes that bypass voiding under MELLLA+ conditions in MNGP do not affect the LPRM instrumentation adversely.

Therefore, the NRC staff finds that the MNGP instrumentation and control systems are adequate to fulfill the requirements of GDC 13 under MELLLA+.

Reactivity Control

GDCs 20, 25, 26, and 28 specify the requirements for the reactivity control systems.

Changes to reactor power are achieved in the expanded operating domain by controlling core reactivity with control rod blades, in addition to adjusting reactor recirculation flow.

GDCs 20 and 25 are met by the reactor protection system and the SCRAM function of the control rod system. These are unaffected by the implementation of the MELLLA+ domain.

GDC-26 is met by the control rod system and the standby liquid control system. These systems are unaffected by the implementation of the MELLLA+ domain.

Compliance with GDC-28 is assured by demonstrating acceptable radiological consequences and barrier integrity during postulated control rod drop accidents. The most limiting conditions occur during low power operation and, therefore, are unaffected by MELLLA+ implementation.

Based on the above, the NRC staff finds that GDCs 20, 25, 26, and 28 continue to be met.

Conclusions

The NRC staff has reviewed the licensee's analyses related to the effect of the proposed operating domain extension on the nuclear design of the fuel assemblies, control systems, and reactor core. The NRC staff concludes that the licensee adequately accounted for the effects of

the proposed operating domain extension on the nuclear design and has demonstrated that the fuel design limits will not be exceeded during normal or anticipated operational transients, and that the effects of postulated reactivity accidents will not cause significant damage to the RCPB or impair the capability to cool the core. Based on this evaluation and in coordination with the reviews of the fuel system design, thermal and hydraulic design, and transient and accident analyses, the NRC staff concludes that the nuclear design of the fuel assemblies, control systems, and reactor core will continue to meet the applicable requirements of GDCs 10, 11, 12, 13, 20, 25, 26, and 28. Therefore, the NRC staff finds the proposed operating domain extension acceptable with respect to the nuclear design.

3.1.3 Thermal and Hydraulic Design

Regulatory Evaluation

The NRC staff reviewed the thermal and hydraulic design of the core and the RCS to confirm that the design (1) has been accomplished using acceptable analytical methods, (2) is equivalent to or a justified extrapolation from proven designs, (3) provides acceptable margins of safety from conditions which would lead to fuel damage during normal reactor operation and AOOs, and (4) is not susceptible to thermal-hydraulic instability. The review also covered hydraulic loads on the core and RCS components during normal operation and DBA conditions and core thermal-hydraulic stability under normal operation and ATWS events. The NRC's acceptance criteria are based on (1) GDC-10, insofar as it requires that the reactor core be designed with appropriate margin to assure that SAFDLs are not exceeded during any condition of normal operation, including the effects of AOOs; and (2) GDC-12, insofar as it requires that the reactor core and associated coolant, control, and protection systems be designed to assure that power oscillations, which can result in conditions exceeding SAFDLs, are not possible or can reliably and readily be detected and suppressed. Specific review criteria are contained in SRP Section 4.4, and other guidance provided in Matrix 8 of RS-001.

Technical Evaluation

Analytical Methods

The staff has reviewed the analytical methods utilized by the licensee. A comprehensive list of the codes used to support the SAR analyses is documented in Table 1-1 of NEDC-33435P, Rev. 1 (Reference 1). As described in the footnotes to Table 1-1, not all the codes being used have an explicit staff SER associated with them; however, sufficient regulatory basis are provided for their use. The following exceptions are noted:

1. The ISCOR code does not have an explicitly approved SER; however, the approval SER for NEDE-24011-P, Rev. 0, mentions a "digital computer code" that is an acceptable method. General Electric - Hitachi (GEH) states that the digital computer code referred to in the SER for NEDE-24011-P, Rev. 0, is indeed ISCOR.

2. A similar situation to Item 1, above, occurs with STEMP code, which is used to calculate the suppression pool temperature using basic energy conservation equations. STEMP was referenced in the approval of NEDE-24222.

3. The LAMB code is explicitly approved for use in ECCS-LOCA applications, but it is not explicitly approved for use in reactor internal pressure differences and containment response; however, this is simply an extension of the approved use, and the models used are those of the approved ECCS-LOCA application.

4. GEH uses two main versions of the TRACG computer code. TRACG02, which is approved for DSS-CD and ATWS analysis, and TRACG04, which is also approved for DSS-CD. The main difference between the two versions is that TRACG04 uses the new-version PANAC11 cross section libraries, whereas TRACG02 requires the old PANAC10 cross sections. TRACG04 has been used as a best-estimate calculation; however, the licensing basis analyses are: ODYN for ATWS and TRACG02 for DSS-CD. Both of these codes are approved.

5. The licensee used TRACG04 for the ATWSI transient. The NRC staff performed preliminary reviews of the TRACG04 methodologies for the determination of the minimum temperature for stable film boiling (T_{min}) and the quenching model for post-T_{min} operating conditions. The review of these two TRACG04 methodologies has not been completed at this time; however the application of TRACG04 for ATWSI calculation in MNGP is acceptable because the licensee demonstrated that the T_{min} criteria is not reached at MNGP even when using the most conservative version of the T_{min} correlation. Because the MNGP TRACG04 ATWSI calculations essentially have the T_{min} and quench models "turned off," review of the T_{min} and quenching methodologies is not required to provide assurance that MNGP satisfies the core coolability ATWS criteria.

Based on the above, the NRC staff concludes that all the analytical methods used in the SAR are either approved or provide an acceptable extension of the use of an approved code.

Equivalency to Proven Designs

The proposed MELLLA+ operating domain is similar in design to the power-flow operating domain currently in use by MNGP. The primary difference is the higher power-to-flow ratio in the MELLLA+ corner, which results in higher operating void fraction and higher operating power when the recirculation pumps are tripped, which affect ATWS performance.

Transient Response

The licensee provided analyses for normal operation, AOOs, infrequent and special events. The complete staff evaluation of these results is documented Section 4.6, "Accident and Transient Analyses." As documented in that evaluation, operation at lower flows in the MELLLA+ domain has an impact on transient response. The licensee proposes to increase the operating limit margin when entering the MELLLA+ region so the final thermal hydraulic performance during transients will be maintained.

In particular, the licensee proposes to implement a LHGR setdown of 12.3 percent while in the MELLLA+ region. Through analysis, the licensee has demonstrated that with this setdown value, the LOCA results are consistent with pre-MELLLA+ values and margins are preserved.

The OLMCPR steady-state limits are calculated on a cycle specific basis to maintain the same margin to the SLMCPR during transients. The transient delta-CPR, which defines the OLMCPR, is calculated for all transients affected by the MELLLA+ extension. In this way, the limiting transient event initiating from inside the MELLLA+ region has the same margin to the SLMCPR as before the MELLLA+ domain was implemented.

Stability

MNGP will implement the DSS-CD solution consistent with the M+LTR. DSS-CD implementation includes any limitations and conditions in the applicable DSS-CD SER.

In the responses to RAI-06, 07, and 08, GEH provided a detailed presentation of the DSS-CD methodology application for MNGP. The presentation slides provided an adequate description of the (Reference 11) of the DSS-CD methodology for MNGP. In order to use a **[[**

]]. For the reference DSS-CD design **[[**

]]. The numbers for **[[**

]], in the demonstrated implementation the ultimate safety limit (SLMCPR) remains unchanged **[[** **]]**. The approach **[[**

]] is acceptable.

In its response to RAI-09, GEH states that MNGP will implement the Automated Backup Stability Protection (ABSP) option. Technical Specifications require restoration of the primary DSS-CD instrumentation within 120 days if both OPRM system and ABSP function are inoperable. Additionally, new TS 3.3.1.1, Required Action I.3, and TS 5.6.6, “OPRM Report,” requires a special report be submitted to the NRC staff within 90 days, detailing the licensee’s plan and schedule for restoration of the primary stability licensing option. The ABSP option provides acceptable stability protection while the primary DSS-CD option is declared inoperable.

<u>Conclusions</u>

The NRC staff has reviewed the licensee’s analyses related to the effects of the proposed operating domain extension on the thermal and hydraulic design of the core and the RCS. The NRC staff concludes that the licensee adequately accounted for the effects of the proposed operating domain extension on the thermal and hydraulic design and demonstrated that the design (1) has been accomplished using acceptable analytical methods, (2) is equivalent to proven designs, (3) provides acceptable margins of safety from conditions that would lead to fuel damage during normal reactor operation and AOOs, and (4) is not susceptible to thermal-hydraulic instability. The NRC staff further concludes that the licensee adequately accounted

for the effects of the proposed operating domain extension on the hydraulic loads on the core and RCS components. Based on this, the NRC staff concludes that the thermal and hydraulic design will continue to meet the requirements of GDCs 10 and 12 following implementation of the proposed operating domain extension. Therefore, the NRC staff finds the proposed operating domain extension acceptable with respect to thermal and hydraulic design.

3.1.4 Emergency Systems

RS-001 (Reference 2) provides guidance for the review of emergency systems for operating domain extensions. The NRC staff reviewed the following systems:

- Control rod drive system
- Overpressure protection for the RCPB during power operation
- Reactor core isolation cooling
- Reactor heat removal system
- Standby liquid control system

3.1.4.1 Control Rod Drive System

Regulatory Evaluation

The NRC staff's review covered the functional performance of the control rod drive system (CRDS) to confirm that the system can affect a safe shutdown, respond within acceptable limits during AOOs, and prevent or mitigate the consequences of postulated accidents. The review also covered the CRDS cooling system to ensure that it will continue to meet its design requirements. The NRC's acceptance criteria are based on (1) GDC-4, insofar as it requires that SSCs important to safety be designed to accommodate the effects of the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents; (2) GDC-23, insofar as it requires that the protection system be designed to fail into a safe state; (3) GDC-25, insofar as it requires that the protection system be designed to assure that SAFDLs are not exceeded for any single malfunction of the reactivity control systems; (4) GDC-26, insofar as it requires that two independent reactivity control systems be provided, with both systems capable of reliably controlling the rate of reactivity changes resulting from planned, normal power changes; (5) GDC-28, insofar as it requires that the reactivity control systems be designed to assure that the effects of postulated reactivity accidents can neither result in damage to the RCPB greater than limited local yielding, nor disturb the core, its support structures, or other reactor vessel internals so as to significantly impair the capability to cool the core; and (6) 10 CFR 50.62(c)(3), insofar as it requires that all BWRs have an alternate rod injection (ARI) system diverse from the reactor trip system, and that the ARI system have redundant scram air header exhaust valves. Specific review criteria are contained in SRP Section 4.6.

Technical Evaluation

The control rod design has not been modified relative to the baseline. Therefore, the NRC staff concludes that the regulatory requirements in GDCs 4, 23, 25, 26, 28, and 10 CFR 50.62(c)(3) continue to be satisfied by the design.

3.1.4.2 Overpressure Protection for the RCPB During Power Operation

Regulatory Evaluation

Overpressure protection for the RCPB during power operation is provided by relief and safety valves and the reactor protection system. The NRC staff's review covered relief and safety valves on the main steam lines and piping from these valves to the suppression pool. The NRC's acceptance criteria are based on (1) GDC-15, insofar as it requires that the RCS and associated auxiliary, control, and protection systems be designed with sufficient margin to assure that the design conditions of the RCPB are not exceeded during any condition of normal operation, including AOOs; and (2) GDC-31, insofar as it requires that the RCPB be designed with sufficient margin to assure that it behaves in a non-brittle manner and that the probability of rapidly propagating fracture is minimized. Specific review criteria are contained in SRP Section 5.2.2.

Technical Evaluation

The licensee has evaluated the impact of the proposed operating domain extension on overpressure protection. The evaluation is documented in Section 3.1 of the SAR. Two items have been evaluated:

1. Flow induced vibrations, and
2. Overpressure relief

Flow-induced vibrations are disposed of generically in the MELLLA+ SER because the pressure remains unchanged; therefore, the steam flow during normal operation or through a relief valve or break remains unchanged, and there is no significant effect on flow induced vibrations.

For MNGP, the limiting overpressure event is the main steam isolation valve closure followed by a High-Flux Scram. The analyses in Section 3.1 of the SAR indicate that the peak vessel pressure remains unchanged, and it is below the 1375 psig limit.

For the ATWS analyses (Section 9.3.1 of the SAR), the licensee concludes that the required SRV capacity must be increased to demonstrate acceptable ATWS results. To satisfy this requirement, the licensee proposed a restriction to the number of SRVs that can be out-of-service in order to operate in the MELLLA+ domain. As such, the number of operable SRVs required in the MELLLA+ domain is effectively increased by one. The NRC staff finds this operating restriction on the SRVs to be acceptable for maintaining overpressure protection of the RCPB if operating in the MELLLA+ domain.

3.1.4.3 Reactor Core Isolation Cooling

Regulatory Evaluation

The reactor core isolation cooling (RCIC) system serves as a standby source of cooling water to provide a limited decay heat removal capability whenever the main feedwater system is isolated from the reactor vessel. In addition, the RCIC system may provide decay heat removal necessary for coping with a station blackout. The water supply for the RCIC system comes from the condensate storage tank, with a secondary supply from the suppression pool. The NRC staff's review covered the effect of the proposed MELLLA+ expanded operating domain (EOD) on the functional capability of the system. The NRC's acceptance criteria are based on (1) GDC-4, insofar as it requires that SSCs important to safety be protected against dynamic effects; (2) GDC-5, insofar as it requires that SSCs important to safety not be shared among nuclear power units unless it can be demonstrated that sharing will not impair its ability to perform its safety function; (3) GDC-33, insofar as it requires that a system to provide reactor coolant makeup for protection against small breaks in the RCPB be provided so the fuel design limits are not exceeded; (4) GDC-54, insofar as it requires that piping systems penetrating containment be designed with the capability to periodically test the operability of the isolation valves to determine if valve leakage is within acceptable limits; and (5) 10 CFR 50.63, insofar as it requires that the plant withstand and recover from an SBO of a specified duration. Specific review criteria are contained in SRP Section 5.4.6.

Technical Evaluation

The RCIC design has not been modified relative to the baseline and the expanded operating domain does not have an impact on the gross thermal power. As such, the requirements of GDCs 4, 5, 33, 54, and 10 CFR 50.63 continue to be satisfied.

3.1.4.4 Residual Heat Removal System

Regulatory Evaluation

The residual heat removal (RHR) system is used to cool down the RCS following shutdown. The RHR system is typically a low pressure system which takes over the shutdown cooling function when the RCS temperature is reduced. The NRC staff's review covered the effect of the proposed MELLLA+ EOD on the functional capability of the RHR system to cool the RCS following shutdown and provide decay heat removal. The NRC's acceptance criteria are based on (1) GDC-4, insofar as it requires that SSCs important to safety be protected against dynamic effects; and (2) GDC-5, insofar as it requires that SSCs important to safety not be shared among nuclear power units unless it can be shown that sharing will not significantly impair their ability to perform their safety functions. Specific review criteria are contained in SRP Section 5.4.7 and other guidance provided in Matrix 8 of RS-001.

Technical Evaluation

The RHR system design has not been modified relative to the baseline, and the expanded operating domain does not have an impact on the decay heat. Thus, the requirements of GDCs 4 and 5 continue to be satisfied.

3.1.4.5 Standby Liquid Control System

Regulatory Evaluation

The standby liquid control system (SLCS) provides backup capability for reactivity control independent of the control rod system. The SLCS functions by injecting a boron solution into the reactor to affect shutdown. The NRC staff's review covered the effect of the proposed MELLLA+ EOD on the functional capability of the system to deliver the required amount of boron solution into the reactor. The NRC's acceptance criteria are based on (1) GDC-26, insofar as it requires that two independent reactivity control systems of different design principles be provided, and that one of the systems be capable of holding the reactor subcritical in the cold condition; and (2) 10 CFR 50.62(c)(4), insofar as it requires that the SLCS be capable of reliably injecting a borated water solution into the reactor pressure vessel at a boron concentration, boron enrichment, and flow rate that provides a set level of reactivity control. Specific review criteria are contained in SRP Section 9.3.5 and other guidance provided in Matrix 8 of RS-001.

Technical Evaluation

The Hot Shutdown Boron Weight (HSBW) is calculated on generic bases for each fuel line (e.g. GE14 in the case of MNGP). The HSBW is confirmed effective on plant- and cycle-specific basis with ODYN and TRACG ATWS calculations. Section 9.3.1 of the SAR documents these calculations. Both the licensing bases and the best-estimate ATWS calculations show that the generic HSBW is effective to shutdown the MNGP core under MELLLA+ initial conditions. Therefore, no modification to the SLCS design is required for MELLLA+.

The HSBW value was analyzed in the MELLLA+ SER NEDC-33006P-A, Rev. 3 (Reference 5). In the response to RAI 5 of NEDC-33006P-A, GEH evaluated the impact of different HSBW values on the final suppression pool temperature. In the response to RAI 1.10 of NEDC-33006PA, GEH provided the HSBW in parts per million (ppm) as a function of core exposure where the average value is very close to the analysis value of 522 ppm. The NRC staff concluded that the acceptability of 522 ppm would need to be evaluated on a plant-specific basis. This was done for MNGP through ODYN simulation and the best estimate simulation with a 3D TRACG simulation for the ATWS analysis. Thus, the NRC staff finds that the ODYN and TRACG ATWS analyses confirm the applicability of the generic HSBW value for MNGP under MELLLA+ conditions.

The SLCS design has not been modified relative to the baseline, the reactor pressure has not been modified and the SLCS boron inventory shutdown margin has been evaluated for the initial core in the SAR (Reference 1). Therefore, the NRC staff finds that sufficient information has

been provided to review the SLCS, and the requirements of GDC-26 and 10 CFR 50.62(c)(4) continue to be satisfied.

Conclusions

The NRC staff has reviewed the licensee's analyses related to the effects of the proposed operating domain extension on the functional design of the CRDS. The NRC staff concludes that the design has not been modified relative to the baseline. The regulatory requirements in GDC-4, 23, 25, 26, 28, and 10 CFR 50.62(c)(3) continue to be satisfied by the design.

The NRC staff has reviewed the licensee's analyses related to the effects of the proposed operating domain extension on the overpressure protection capability of the plant during power operation. The NRC staff concludes that the licensee (1) adequately accounted for the effects of the proposed operating domain extension on pressurization events and overpressure protection features and (2) demonstrated that the plant will continue to have sufficient pressure relief capacity to ensure that pressure limits are not exceeded. Based on this, the NRC staff concludes that the overpressure protection features will continue to meet GDCs 15 and 31 following implementation of the proposed operating domain extension. Therefore, the NRC staff finds the proposed operating domain extension acceptable with respect to overpressure protection during power operation.

The NRC staff has reviewed the licensee's analyses related to the effects of the proposed operating domain extension on the ability of the RCIC system to provide decay heat removal following an isolation of main feedwater event and provide makeup to the core following a small break in the RCPB. The NRC staff concludes that the design has not been modified relative to the baseline, and the expanded operating domain does not have an impact on the gross thermal power. Therefore, the licensee adequately accounted for the effects of the proposed operating domain extension on these events and demonstrated that the RCIC system will continue to provide sufficient decay heat removal and makeup for these events following implementation of the proposed operating domain extension. Based on this, the NRC staff concludes that the RCIC system will continue to meet the requirements of GDCs 4, 5, 33, and 54, and 10 CFR 50.63 following implementation of the proposed operating domain extension. Therefore, the NRC staff finds the proposed operating domain extension acceptable with respect to the RCIC system.

The NRC staff has reviewed the licensee's analyses related to the effects of the proposed operating domain extension on the RHR system. The NRC staff concludes that the design has not been modified relative to the baseline, and the expanded operating domain does not have an impact on the decay heat. Therefore, the licensee adequately accounted for the effects of the proposed operating domain extension on the system and demonstrated that the RHR system will maintain its ability to cool the RCS following shutdown and provide decay heat removal. Based on this, the NRC staff concludes that the RHR system will continue to meet the requirements of GDCs 4 and 5 following implementation of the proposed operating domain extension. Therefore, the NRC staff finds the proposed operating domain extension acceptable with respect to the RHR system.

The NRC staff has reviewed the licensee's analyses related to the effects of the proposed operating domain extension on the SLCS and concludes that the design has not been modified relative to the baseline, the reactor pressure has not been modified and the SLCS boron inventory shutdown margin has been evaluated for the initial core in NEDC-33326P, therefore, the licensee adequately accounted for the effects of the proposed operating domain extension on the system and demonstrated that the system will continue to provide the function of reactivity control independent of the control rod system following implementation of the proposed operating domain extension. Based on this, the NRC staff concludes that the SLCS will continue to meet the requirements of GDC-26, and 10 CFR 50.62(c)(4) following implementation of the proposed operating domain extension. Therefore, the NRC staff finds the proposed operating domain extension acceptable with respect to the SLCS.

3.1.5 Accident and Transient Analyses

Anticipated Operational Occurrences

The licensee has performed a review of AOO transients and reported the results in Chapter 9 of the SAR (Reference 1). The result of this evaluation is that most transient analyses are either unaffected by the MELLLA+ operating domain extension or are bounded by other analyses. Table 1 (provided below) contains a summary of the AOO analysis evaluation. The AOOs analyzed in the SAR for the MELLLA+ domain extension include the following:

- Generator Load Rejection Without Bypass (LRNBP)
- Turbine Trip with Bypass (TTWBP)
- Turbine Trip Without Bypass (TTNBP)
- Feedwater Controller Failure (Maximum Demand) (FWCF)
- Loss of Feedwater Heater (LFWH)

As shown in Table 1, these AOOs were evaluated at the current licensed power (CLTP), which is equivalent to 120 percent of OLTP, and two flows: the increased core flow (ICF) limit of 105 percent and the MELLLA+ reduced core flow limit of 80 percent. The comparisons show that for all cases, the ICF conditions are more limiting, indicating no impact of MELLLA+ operation on delta-CPR.

Table 1 - Comparison of AOO Analyses Results at 80% and 105% Core Flow

Event	Parameter	Units	CLTP ICF (105%) Rated Core Flow	CLTP 80% Rated Core Flow
TTWBP				
	Peak Neutron Flux	% Initial	634	423
	Peak Heat Flux	% Initial	137	127
	Peak Vessel Pressure	psig	1272	1260
	ΔCPR Option B	NA	0.44	0.40

Event	Parameter	Units	CLTP ICF (105%) Rated Core Flow	CLTP 80% Rated Core Flow
TTNBP				
	Peak Neutron Flux	% Initial	565	350
	Peak Heat Flux	% Initial	132	120
	Peak Vessel Pressure	psig	1256	1245
	ΔCPR Option B	NA	0.40	0.33
LRNBP				
	Peak Neutron Flux	% Initial	458	270
	Peak Heat Flux	% Initial	123	109
	Peak Vessel Pressure	psig	1247	1234
	ΔCPR Option B	NA	0.36	0.25
FWCF				
	Peak Neutron Flux	% Initial	609	361
	Peak Heat Flux	% Initial	140	126
	Peak Vessel Pressure	psig	1252	1241
	ΔCPR Option B	NA	0.43	0.36
HPCIL8				
	Peak Neutron Flux	% Initial	549	339
	Peak Heat Flux	% Initial	139	126
	Peak Vessel Pressure	psig	1242	1231
	ΔCPR Option B	NA	0.43	0.36
LFWH	ΔCPR	NA	0.16 @ 99% RCF	0.13

The operating limits to CPR and LHGR are adjusted upwards when operating at off-rated conditions by power- and flow-dependent factors. The licensee has calculated the slow recirculation flow increase under MELLLA+ conditions to evaluate the power- and flow-dependent limits. The results of these analyses are documented in Section 9.1.2 of the SAR. These results indicate that the existing limits in MNGP are adequate for MELLLA+ operation.

In Section 9.1.3 of the SAR, the licensee has evaluated the impact of MELLLA+ on non-limiting events, including:

1. Inadvertent HPCI start
2. Slow recirculation flow increase
3. Fast recirculation flow increase

The evaluation of the non-limiting events in MNGP confirms the generic disposition of these events in the MELLLA+ SER.

Design Basis Accidents and Events of Radiological Consequence

The licensee has evaluated the impact of MELLLA+ operation on the radiological consequences of design bases events (DBA). Loss-of-coolant events and fuel handling accidents were resolved generically in the MELLLA+ SER, and have been confirmed for the MNGP application. The radiological consequences of the control rod drop accident have been evaluated in Section 9.2.1 of the SAR. The evaluation concludes that MELLLA+ operation is bounded by the existing analyses because MELLLA+ does not change the source term, removal, or transport mechanisms.

Special Events

The evaluation of MELLLA+ impact on special events is documented in Section 9.3 of the SER. Station blackout was evaluated generically in the MELLLA+ SER and the conclusions were confirmed by the NRC staff for MNGP. The remaining special events are ATWS events, including ATWS-Instability (ATWSI).

<u>Regulatory Evaluation</u>

ATWS is defined as an AOO followed by the failure of the reactor portion of the protection system. The regulations at 10 CFR 50.62 require that:

- Each BWR has an ARI system that is designed to perform its function in a reliable manner and be independent (from the existing reactor trip system) from sensor output to the final actuation device.

- Each BWR has a standby liquid control system (SLCS) with the capability of injecting into the reactor vessel a borated water solution with reactivity control at least equivalent to the control obtained by injecting 86 gallons per minute of a 13 weight-percent sodium pentaborate decahydrate solution at the natural boron-10 isotope abundance into a 251-inch inside diameter reactor vessel.

- Each BWR has equipment to trip the reactor coolant recirculation pumps automatically under conditions indicative of an ATWS.

The NRC staff's review was conducted to ensure that (1) the above requirements are met, (2) sufficient margin is available in the setpoint for the SLCS pump discharge relief valve such that SLCS operability is not affected by the proposed MELLLA+ EOD, and (3) operator actions specified in the plant's Emergency Operating Procedures are consistent with the generic EPGs/SAGs, insofar as they apply to the plant design. In addition, the NRC staff reviewed the licensee's ATWS analysis to ensure that (1) the peak vessel bottom pressure is less than the ASME Service Level C limit of 1500 psig; (2) the peak clad temperature is within the 10 CFR 50.46 limit of 2200 Fahrenheit (°F); (3) the peak suppression pool temperature is less than the design limit; and (4) the peak containment pressure is less than the containment design

pressure. The NRC staff also evaluated the potential for thermal-hydraulic instability in conjunction with ATWS events using the methods and criteria approved by the NRC staff. For this analysis, the NRC staff reviewed the limiting event determination, the sequence of events, the analytical model and its applicability, the values of parameters used in the analytical model, and the results of the analyses.

Technical Evaluation

The licensee reviewed the family of ATWS in the MNGP SAR (Reference 1). Two types of ATWS events are analyzed: isolation ATWS that may lead to emergency depressurization (ATWSED), and ATWS with instability (ATWSI). Based on this evaluation, the licensee concludes that the ATWS logic and setpoints remain unchanged for the proposed operating domain extension. The NRC staff finds that the licensee's evaluation is acceptable.

ATWSED Evaluation

The limiting ATWSED events as specified in the MELLLA+ SER are:

1. Main steam isolation valve closure (MSIVC)
2. Pressure regulator failure open (PRFO)
3. Loss of offside power (LOOP)

Two analysis methods are used: the licensing methodology, which uses ODYN, and a best-estimate methodology, which uses TRACG04 with input data from TGBLA06/PANAC11. For the licensing basis ATWSED calculation (ODYN), the water level is controlled to TAF+5 feet, and the suppression pool is allowed to heat up even after the heat capacity temperature limit (HCTL) is reached. For the best-estimate ATWSED calculation (TRACG), emergency depressurization is assumed within 60 seconds after the HCTL is reached.

The ODYN calculation indicates that overpressure limits would be violated if SRVs were allowed to be out of service. Therefore, the licensee imposes a limitation to require all SRVs operable while operating in the MELLLA+ domain. With those assumptions, the peak vessel pressure is calculated to reach 1452 psig, which is below the 1500 psig ASME Service Level C limit. The calculations also show that MELLLA+ operation has a negligible effect on PCT and clad oxidation because the peak channel power and limits remain unchanged.

The ODYN calculation indicates that, without depressurization, the suppression pool temperature would reach a temperature of 197ºF, which is below the containment limit of 281°F. The licensee notes that the ODYN calculations for EPU conditions resulted in suppression pool temperatures that violate the HCTL and would have required depressurization; therefore, MELLLA+ does not present a new condition.

The best-estimate ATWSED TRACG calculations demonstrate that, depending on initial conditions, the HCTL may or may not be reached and emergency depressurization may not be required. The HCTL is a function of the reactor operating pressure and the suppression pool water level. For this reason, the licensee performed the best-estimate analysis for bounding

assumptions of HCTL of 150°F to 175°F. For the low HCTL value, depressurization is required, but not for the high level.

Section 9.3.1 of the SAR presents the results of these analyses. For all cases analyzed, the ATWSED acceptance criteria are satisfied. The NRC staff reviewed the ATWSED data presented and finds the licensee's evaluation acceptable.

ATWSI Evaluation

The licensee evaluated stability during ATWSI events and the results are documented in Section 9.3.1 of the SAR and RAI 9/11/13-2 (Reference 17). The results of the ATWSI analysis show that the mitigation actions in the MNGP EOP procedures (flow runback to uncover the feedwater spargers and early boron injection) are effective in the MELLLA+ operating domain when the MNGP-specific timing for operator actions are used. The TRACG04 calculations indicate that all applicable ATWS criteria are satisfied for ATWSI.

The NRC staff performed preliminary reviews of the TRACG04 methodologies for determination of the minimum temperature for stable film boiling (T_{min}) and the quenching model for post-T_{min} operating conditions. The review of these two TRACG04 methodologies has not been completed at this time; however the application of TRACG04 for ATWSI calculation in MNGP is acceptable because the licensee has demonstrated that the T_{min} criteria is not reached in MNGP even when using the most conservative version of the T_{min} correlation. Because the MNGP TRACG04 ATWSI calculations essentially have the T_{min} and quench models "turned off," review of the T_{min} and quenching methodologies is not required to guarantee that MNGP satisfies the core coolability ATWS criteria.

As required by the MELLLA+ SER limitation, the SAR lists the key operator actions credited. MNGP has taken two deviations from the standard operator-action methodology in the MELLLA+ SER, NEDC-33006P-A (Reference 5):

- Operator actions to reduce water level have been assumed to occur within 90 seconds of the ATWS initiation. This is faster than the recommended value of 120 seconds in NEDC-33006P-A, and faster than the conservative time used for past ATWS analyses of 250 seconds. Boron injection is assumed to occur within 120 seconds, and has not changed from previous analyses.

- The peak rod power has been set at 95 percent of the Linear Heat Generation Rate (LHGR) limit. In past ATWS analyses, peak rod power was set at LHGR limits.

The NRC staff finds the two proposed deviations acceptable because:

- In response to RAI 9/11/13-1 (Reference 17), the licensee provided a commitment to train and test licensed reactor operators to initiate ATWS actions within the allotted time. This commitment is entered in the MNGP commitment tracking system and becomes part of the licensing basis. The NRC staff will verify commitment implementation. Any subsequent changes to the operator action time will require licensee review and evaluation.

Prompt manual feedwater flow runback and early boron injection are adequate to mitigate the ATWSI oscillations, and are still effective in the MELLLA+ domain. The calculations indicate that ATWS acceptance criteria continue to be satisfied even in the presence of unstable power oscillations when the MNGP-specific timing for operator actions is used.

- The licensee reviewed the MNGP MELLLA+ core design and found that it designed with more than a 5 percent margin to the LHGR limit, and the 95 percent setting is conservative for an ATWS best-estimate calculation.

Post-CHF Heat Transfer Models in TRACG04

From October 23 through 25, 2012, the NRC staff conducted an audit of the T_{min} and quench-front GEH methodologies and issued follow-up RAIs. A major audit finding was the lack of experimental data to validate the zirconium (Zr) beta-term credit in the Shumway T_{min} correlation (Reference 15), which may result in T_{min} values approximately 150 degrees Kelvin higher for Zr than stainless steel (SS). Most available T_{min} data has been collected with SS tubes. Experimental data was also missing to justify the use of the void-fraction term in the correlation. For this reason, the staff requested that the MNGP ATWSI calculation be re-run without void-fraction credit, and with two bounding assumptions for T_{min} correlation (with and without the Zr beta-term credit). The NRC staff's expectation is that the actual T_{min} for Zr tubes will lie between these two bounding values. In an RAI response (Reference 16), the licensee indicated that the choice of T_{min} correlation does not significantly affect the power oscillations for core average or hot channel. However, removing the Zr beta-term credit in the Shumway correlation significantly reduces the value of T_{min}. In the MNGP ATWSI calculations, T_{min} is never reached during the oscillations when the best-estimate operation action timing is used.

In addition, as a result of these analyses, GEH reviewed the quench model methods in TRACG04 and found a documentation error in the original correlation for quench front heat transfer coefficient for bottom reflooding that had propagated to the TRACG04 coding. The error was corrected in the code, and GEH has committed to update the documentation. As a result of this coding error, quench front propagation is significantly slower and PCT values increase significantly for transients that require quenching.

The NRC staff has not reviewed in detail the Shumway T_{min} correlation or the new quenching model at this time. The licensee committed to enhance licensed operator training to ensure that actions to reduce water level occurs within 90 seconds of an ATWS event initiation. The use of best-estimate values is acceptable for beyond-design basis events, and the operators are trained and tested to the more stringent times. Since TRACG04 calculations in the response to RAI 9/11/13-2 (Reference 17) indicate that with the 90-second and the 95 percent LHGR assumptions, an ATWSI event is controlled before unstable oscillations have time to develop. Therefore, the NRC staff finds the 90-second action time is acceptable.

The licensee concluded that the TTWBP ATWSI event is no longer limiting for PCT when the MNGP-specific timing for operator actions are used. With respect to unstable power oscillations, the limiting ATWSI event in MNGP becomes a two recirculation pump trip (2RPT). The 2RPT event shows relatively small oscillations, that don't challenge acceptable limits

because the event involves only a flow reduction, and not the significant subcooling event induced by the turbine trip and the associated loss of extraction steam for the feedwater heaters. Even though 2RPT has traditionally not considered an ATWS event because there is no immediate automatic scram signal that could fail, the NRC staff finds the conclusion of the MNGP evaluation reasonable and accepts 2RPT as the limiting ATWSI event for MNGP.

Based on the above data, we conclude that the ATWS mitigation features (i.e., prompt manual FW flow runback and early boron injection) are adequate to mitigate the ATWSI oscillations, and are still effective in the MELLLA+ domain. The calculations indicate that the ATWS acceptance criteria are satisfied even in the presence of unstable power oscillations when the MNGP-specific timing for operator actions is used.

Conclusions

The NRC staff reviewed the information submitted by the licensee related to ATWS and concludes that the licensee adequately accounted for the effects of the proposed operating domain extension on ATWS. The NRC staff concludes that the licensee demonstrated that ARI and SLCS have been installed and that they will continue to meet the requirements of 10 CFR 50.62 and the analysis acceptance criteria following implementation of the proposed operating domain extension. The MNGP commitment to train the operators to take immediate actions (within 90 seconds) to reduce reactor water level ensures the power level during an ATWS event is promptly reduced so that the probability of unstable power oscillations is minimized.

The commitment has been entered into the licensee's commitment tracking system, and will continue to remain part of the licensing basis. The NRC staff will review this commitment to verify implementation. Any future changes to this operator response time will require the licensee to perform a review and evaluation (e.g., 10 CFR 50.59).

Based on the above, the NRC staff finds the proposed operating domain extension acceptable with respect to ATWS.

3.1.6 Fuel Storage

This is an EPU issue under RS-001, and is not applicable to the MELLLA+ application.

3.1.7 Net Positive Suction Head (NPSH)

The available NPSH is a critical parameter for the successful operation of ECCS equipment during ATWS and LOCA events. Evaluations by MNGP indicate that to maintain NPSH during limiting transients, the plant requires credit for Containment Accident Pressure (CAP). This issue has been addressed by the Commission in SECY-11-0014 (Reference 18), which provides CAP guidance.

The CAP issue was resolved during the MNGP EPU application and reviewed by the Advisory Committee on Reactor Safeguards (ACRS) (Reference 19), which summarized the issue as follows:

> This application is the first EPU request using SECY-11-0014 CAP guidance, as well as the BWR Owners Group (BWROG) guidance. NSPM evaluated NPSH margin using conservative assumptions for the limiting DBLOCA, and realistic assumptions for non-design basis events, such as Appendix R fire, anticipated transient without scram (ATWS), and station blackout (SBO) events. The licensee's analyses for each event consisted of the following steps: (a) containment analysis using the Super HEX (SHEX) computer code to calculate the transient wetwell pressure and the corresponding transient suppression pool temperature, (b) calculation of the NPSHa at the inlet of the RHR and CS pumps using the transient suppression pool temperature with varying transient wetwell pressure as inputs, and (c) evaluation of NPSH margin. These deterministic calculations were performed using conservative assumptions consistent with Regulatory Guide 1.82, Revision 3.

Additional evaluation of NPSH margin is provided by the NRC staff in Section 3.2.6 of this safety evaluation.

3.1.8 Disposition of Limitations in Applicable SERs

The SAR (Reference 1) appendices summarize the disposition of the limitations in the applicable SERs, including:

1. The Methods SER, NEDC-33173P-A (Reference 6)
2. The MELLLA+ SER, NEDC-33006P-A, Rev. 3 (Reference 5)
3. The DSS-CD SER, NEDC-33075P-A (Reference 7), and
4. The DSS-CD TRACG Application SER, NEDC-33147P (Reference 8)

NEDC-33173P-A Limitations

Appendix A summarizes the disposition of limitations in the Methods SER, NEDC-33173P-A (Reference 6). The licensee states that the Methods SER limitations discussed below do not apply to MNGP based on the following justifications:

- 9.2 - The limitation is applicable only to TGBLA04/PANAC10 applications.
- 9.4 - The SLMCPR limitation is applied to EPU conditions and is not increased further for MELLLA+ operation.
- 9.12 - The Thermal-Mechanical evaluation was performed using GESTR code. A sensitivity analysis was included using PRIME.
- 9.13 - MNGP MELLLA+ uses less than 10 percent Gd.
- 9.15, 9.16, 9.20 – TRACG04/PANAC11 was used for MELLLA+ and applicable void biases and uncertainties are included.

- 9.18 - Because the significant conservatisms in the current licensing methodology and associated MCPR margins are more than sufficient to compensate for the overall uncertainty in the OPRM instrumentation.

- 9.21 - MNGP MELLLA+ is not based on a mixed core.

- 9.22 - GE14 is an approved fuel line in the methods SER.

- 9.23 - GEH has a standing commitment to submit eigenvalue and power distribution tracking data following implementation of MELLLA+. This limitation cannot be satisfied prior to implementation and is, thus, not applicable to the MNGP SAR.

The disposition of the limitations applicable to MNGP is summarized on a table in the SAR appendix, and discussed with additional detail in the body of the report. These limitations and their resolution are as follows:

- 9.1 - TGBLA06/PANAC11 methods are used.

- 9.3 - MNGP MELLLA+ power density is 34.8 MW/(Mlbm/hr), which does not exceed the 50 MW/(Mlbm/hr) limit.

- 9.5 - A 0.03 penalty is added to the MNGP SLMCPR.

- 9.6 - The R-factors are consistent with the axial void profiles expected in MNGP.

- 9.7 - The MNGP ECCS LOCA analyses include an evaluation for top-peaked and mid-peaked axial power profiles.
- 9.8 - MNGP ECCS LOCA calculations have been performed at the MELLLA+ corner (100% CLTP 80% Flow). As a result, the LHGR setdown has been increased from 10 percent to 12.3 percent to maintain PCT performance.

- 9.9 and 9.10 - Calculations documented in the SRLR confirm the evaluation of the LHGR and AOOs documented in Section 9 of the SAR. T-M limits are satisfied with the proposed operating margins.

- 9.11 - The results in Section 9 of the SAR demonstrate a 10 percent margin to T-M limits.

- 9.14 - Following NRC review, the 10 CFR Part 21 concern addressed by this limitation was resolved, but the 350 psi penalty was applied at MNGP.

- 9.17 - Bypass voiding is conservatively estimated at 3.4 percent, which satisfies the limitation.

- 9.19 - TRACG04 methods were used for the MELLLA+ Cycle 27 analysis.

- 9.24 - The bundle power, operating LHGR, and MCPR have been provided for the first MELLLA+ MNGP cycle. All limits are satisfied.

NEDC-33006P-A Limitations

Appendix B summarizes the disposition of limitations in the MELLLA+ SER, NEDC-33006P-A, Rev. 3 (Reference 5). The licensee states that the following MELLLA+ SER limitations do not apply to MNGP based on the following justifications:

- 12.3d, 12.3e, 12.3f, 12.23.6, and 12.23.7 - MNGP MELLLA+ uses a full load of GE14 fuel, which is already approved.

- 12.10c - MNGP MELLLA+ takes credit for off-rated limits at minimum core flow state point; therefore, core monitoring is required per limitation 12.10.d.

- 12.20 - MNGP does not use the generic ATWS/Stability analysis and has performed a plant-specific ATWS Instability evaluation.

The disposition of the limitations applicable to MNGP is summarized on a table in the SAR appendix and discussed in more detail in the body of the report. These limitations and their resolution are as follows:

- 12.1 - GEXL-Plus applicability has been confirmed in Section 1.1.3 of the SAR.

- 12.2 - The limitations from NEDC-33173P-A, NEDC-33075P-A, and NEDC-33147P-A are specifically addressed in Appendices A, C, and D of the SAR.

- 12.3a - As addressed in Section 1.1.2 of the SAR, concurrent changes have been taken into account in the evaluation

- 12.3b - As addressed in Section 1.1.1 of the SAR, all generic dispositions have been reviewed for applicability.

- 12.3c - As addressed in Section 1.1.1 of the SAR, generic bounding sensitivities have been reviewed for applicability.

- 12.3g - DSS-CD will be employed in MNGP to address possible instabilities. DSS-CD has been approved for MELLLA+ applications.

- 12.4 - The MNGP application has provided the plant-specific thermal limits and transient assessment in the SRLR.

- 12.5a - SLO operation and SRV OOS are not allowed in MELLLA+. The MNGP TS have been updated.

- 12.5b – Operation with a feedwater heater out-of-service is not allowed at MNGP.

- 12.5c - The licensee has committed to provide the power-flow map in the COLR.

- 12.6 - The licensee has evaluated the SLMCPR at off-rated conditions and has reported it in the SRLR.

- 12.7 - The DSS-CD automated backup stability option will be implemented in MNGP.

- 12.8 - The change of vessel effective full power years is estimated to be less than 1 percent under MELLLA+ conditions. Up to date approved methodology was used for the estimation.

- 12.9 - A discussion of non-category-A materials is presented in Section 3.5.1.4 of the SAR. The Augmented Inspection Program at MNGP is considered as still applicable under MELLLA+ conditions.

- 12.10a - The MNGP-specific Appendix K ECCS LOCA calculations were provided in the SAR. The PCT results are determined to be bound by the high-flow PCT values. A plant-specific LHGR setdown has been calculated for MNGP.

- 12.10b, and d - MNGP has opted for monitoring off-rated LOCA limits and taking credit for them during the analysis. The licensee has committed to confirm these off-rated limits for every future reload.

- 12.11 - Top peaked and mid-peaked power shapes have been used for the LOCA analyses.

- 12.12a and b - Both the nominal and the Appendix K LOCA results have been reported in the SAR. The current uncertainty method was used.

- 12.13, 12.14 - No small-break LOCA PCT calculations were required for MNGP because the rated small break PCT is significantly lower than the limiting Appendix K PCT values. A number of small-break sizes were evaluated to determine the limiting event for the MNGP EPU application.

- 12.15 - Bypass voiding has been calculated to be **[[ ]]**, which is lower than the **[[ ]]** limit.

- 12.16 - A plant-specific RWE analysis was performed using PANACEA to confirm the validity of the RBM setpoints.

- 12.17 - ATWS calculations were performed in Section 9.3.1 of the SAR using the licensing basis (ODYN) and a best-estimate code (TRACG).

- 12.18a, and c - TRACG ATWS calculations were performed to demonstrate compliance with ATWS criteria because: (1) the licensing bases ODYN calculation showed that

HCTL limit would be reached, and (2) the licensee opted not to increase the Boron 10 concentration.

- 12.18b - MNGP MELLLA+ employs the best-estimate TRACG analysis to confirm ODYN calculations.

- 12.18d - MNGP Administrative Procedures will implement the limitation of no SRVs out of service in the MELLLA+ region.

- 12.18e and f - The key assumptions used for the ATWS analyses and the treatment of uncertainties are documented in Section 9.3.1 of the SAR.

- 12.19 - The licensee has provided a best-estimate ATWS/Stability calculation using TRACG04 to demonstrate compliance with limits.

- 12.21 - A plant-specific probabilistic risk assessment was included in Section 10.5 of the SAR. Based on these analyses, the licensee concludes that the risk increase lies in within Region III (i.e., changes that represent very small risk changes).

- 12.22 - Fluence calculations indicate that the top guide and shroud exceed the 5×10^{20} n/cm^2 [neutrons per square centimeter] threshold. The inspection strategies in place are considered sufficient.

- 12.23.2 - The ATWS calculations key parameters were provided.

- 12.23.3 - The SRV tolerances were included in the ATWS analyses.

- 12.23.4 - The MNGP Emergency Operating Procedures (EOP) were reviewed and sensitivity analyses performed for different water level control strategies. The EOPs require operators to lower reactor vessel water level to TAF [top of active fuel] (unless the transient terminates early) and control within a band between the minimum steam cooling water level and 2 feet below the feedwater spargers. A wide band is necessary because manual level control during an AWTS cannot be accomplished accurately. The sensitivity calculations indicate that the EOP strategy is adequate to satisfy the ATWS criteria.

- 12.23.5 - The MNGP MELLLA+ power density is 43.5 MW/(Mlbm/hr), which does not exceed the 52.5 MW/(Mlbm/hr) limit.

- 12.23.8 - The ATWS calculations accounted for all MNGP specific features.

- 12.23.9 – The plant-specific ATWS calculations accounted for the physical limitations of HPCI and RCIC. NPSH limitations are described separately in Section 3.9.3.

- 12.23.10 - The containment pressure calculated by the best estimate TRACG analysis is **[[ ]]**, which is under the containment limit of **[[ ]]** for MNGP. All safety grade equipment will function under containment overpressure conditions.

- 12.23.11 - The HCTL values used for ATWs calculations are the nominal values. They are a function of vessel pressure and suppression pool water level.

- 12.24.1 - The TRACG MNGP-specific calculations **[[ ]].**

- 12.24.2 - The core exit void fraction is presented in table 1-2 of the SAR for a MELLLA, and MELLLA+. The highest void fraction under MELLLA+ corresponds to the low flow point (82.5% CLTP, 57.4% flow) and has a value of 73 percent, compared to 69 percent for the nominal MELLLA condition (100% CLTP, 99% flow).

NEDC-33075P-A Limitations

Appendix C in the SAR summarizes the disposition of limitations in the DSS-CD SER, NEDC-33075P-A (Reference 7). The licensee states that the following DSS-CD SER limitations do not apply to MNGP based on the following:

- 4.3 - The applicability checklist, which is used to confirm that the generic DSS-CD calculations are directly applicable to MNGP, confirms the generic applicability.

- 4.5 - MNGP uses only GE14 fuel, which is already been analyzed in the generic calculations.

The disposition of the limitations applicable to MNGP is summarized on a table in the SAR appendix and discussed in more detail in the body of the report. The limitations and their resolution as follows:

- 4.1 - DSS-CD will be implemented in the already approved GE Option III platform.

- 4.2 - The DSS-CD applicability checklist was followed for MNGP and the results indicate that MNGP satisfies all the criteria for the generic analyses.

- 4.4 - MNGP TS have been updated to reflect the DSS-CD related items, including backup stability implementation.

- 4.6 - MNGP has submitted information to support the use of plant-specific CDA setpoints.

- 4.7 - MNGP will implement DSS-CD on the existing Option III hardware.

- 4.8 - The plant-specific setting values established by GEH and documented in the plant-specific DSS-CD Setting Report.

- 4.9 - The licensee has reviewed the applicability of the DSS-CD trip function and concluded that it is applicable.

NEDC-33147P Limitations

Appendix D in the SAR summarizes the disposition of limitations in the DSS-CD TRACG Application SER, NEDC-33147P (Reference 8). The only limitation in this SER is Limitation 4.1, which requires a submittal for review if a change on uncertainty evaluation is proposed. The licensee states that no change is proposed and no submittal is necessary.

1. The TRACG calculations to support the plant-specific DSS-CD setpoints have used the existing uncertainty methodology.

The NRC staff finds the licensee's evaluation of the above limitations acceptable.

Conclusion

The NRC staff has reviewed the proposed MNGP MELLLA+ operating domain extension as documented in NEDC-33435P, Rev. 1 (Reference 1). The staff reviewed the licensee's analyses related to the effect of the proposed extension on the nuclear design of the fuel assemblies, control systems, and reactor core. The main conclusion from this review is that the broadening of the MNGP operating domain by lowering the flow at high powers without additional limitations would reduce the safety margin, but the solutions proposed in the MNGP SAR are technically acceptable to satisfy the regulatory criteria. The following solutions are proposed to maintain the same safety margin under MELLLA+ than under the current operating domain:

- Operation in the MELLLA+ domain will preclude the flexibility of having SRV out of service during operation in the MELLLA+ region. This restriction is implemented by administrative controls and is necessary to demonstrate compliance to peak vessel pressure limits during isolation ATWS events. As such, the licensee will revise the MNGP Technical Requirements Manual prior to MELLLA+ implementation.

- Operation with a feedwater heater out-of-service is not allowed at MNGP in the MELLLA+ domain, and analyses have not been performed to demonstrate compliance with applicable criteria under these conditions.

- Single Loop Operation (SLO) is not allowed in the MELLLA+ domain.

- The LHGR setdown value will be increased by an additional 2.3 percent in the MELLLA+ domain to maintain equivalent Peak Clad Temperature (PCT) performance during LOCA events. This setdown value will be implemented in the COLR and confirmed for future cycles.

- The MAPLHGR limit will be reduced by an additional 2.6 percent for operation in the MELLLA+ domain to maintain equivalent PCT performance during LOCA events. This setdown will be implemented in the COLR and confirmed for future cycles.

- To provide additional protection against spurious, noise-induced scrams, the amplitude discriminator setpoint (SAD) of the DSS-CD system **[[**

  ]].

- The licensee provided in the response to RAI 9/11/13-1 (Reference 17) a commitment to train and test licensed reactor operators to initiate ATWS actions within the allotted time to satisfy all acceptance criteria for ATWSI events.

The results of the licensee analyses (summarized in Table 9-1 of NEDC-33435P, Rev. 1, (Reference 1) indicate that the limiting AOOs result in larger delta-CPR when initiated at nominal conditions than inside the MELLLA+ domain; therefore, additional OLPMCPR margin is not required for operation in the MELLLA+ domain.

The NRC staff concludes that the licensee adequately accounted for the effects of the proposed MELLLA+ operating domain extension on the nuclear design and demonstrated that the fuel design limits will not be exceeded during normal or anticipated operational transients, and that the effects of postulated reactivity accidents will not cause significant damage to the pressure vessel or impair the capability to cool the core.

Based on the above evaluation, the NRC staff concludes that the nuclear design of the fuel assemblies, control systems, and reactor core will continue to meet the applicable regulatory requirements. Therefore, the NRC staff finds the proposed MELLLA+ operating domain extension acceptable.

3.2 Containment and Ventilation

The MNGP is a Boiling Water Reactor (BWR)-3 with a Mark I pressure suppression type primary containment. As described in Section 5.1 of the MNGP USAR, Revision 24, the primary containment encloses the reactor vessel (RV), the reactor coolant recirculation loops, and other branch connections of the reactor coolant system (RCS). The major elements of the primary containment are the drywell, the pressure suppression chamber (or wetwell) that stores a large volume of water (suppression pool), the connecting vent pipe system between the drywell and the wetwell, isolation valves, the vacuum relief system, the containment cooling systems and other service equipment.

The NRC staff has previously reviewed the GEH generic licensing topical report NEDC-33006P, “General Electric Boiling Water Reactor Maximum Extended Load Line Limit Analysis Plus,” and provided its SER (Reference 5). The following plant specific safety evaluation of the MNGP MELLLA+ is based on the NRC staff’s evaluation documented in Reference 5.

The NRC staff based its review on the following sections of Reference 20:

Section 4.1 Containment System Performance
Section 4.2.5 ECCS Net Positive Suction Head
Section 4.4 Main Control Room Atmosphere Control System
Section 4.5 Standby Gas Treatment System
Section 4.7 Post-LOCA Combustible Gas Control System

3.2.1 Containment Pressure and Temperature Response Analysis

The licensee used LAMB computer code (Reference 24) for the short term mass and energy (M&E) release analysis and M3CPT computer code (Reference 25) for the short term containment pressure and temperature response analysis for the proposed EPU MELLLA+ operating domain which are the same as those used in the current analysis.

Short-Term LOCA Analysis for Drywell Pressure Response

In Section 4.1.1 of Reference 1, the licensee stated that the **[[**

]]. The licensee's calculated peak drywell pressures for EPU MELLLA and the EPU MELLLA+ operating domains are 44.1 psig [pounds per square inch gauge] and 44.0 psig, respectively. The licensee's calculated peak drywell-to-wetwell differential pressures for the EPU MELLLA and the EPU MELLLA+ operating domain are 24.8 psig and 24.7 psig, respectively. Since the licensee used the same methodology as in the current licensing basis analysis, the NRC staff finds the licensee's evaluation acceptable.

Short-Term LOCA Analysis for Drywell Gas Temperature Response

The licensee stated that the **[[**

]]. The licensee stated that the peak drywell temperatures for the EPU MELLLA and the MELLLA+ operating domain are 291°F [degrees Fahrenheit] and 290°F, respectively. The NRC staff identified that Reference 22, Table 2.6-1, provides a peak drywell temperature of 335°F under the licensed thermal power of 1775 MWt, and 338°F under the EPU thermal power of 2004 MWt both in MELLLA operating domain. In an NRC staff RAI, the licensee was requested to explain this inconsistency or justify the difference. The licensee's response (Reference 23, RAI 10b) stated that the peak drywell temperatures reported under EPU MELLLA conditions in Table 2.6-1 of Reference 22 were obtained from a long-term containment response calculation for a small steam line break accident (SBA) at 102-percent EPU thermal power and 100-percent core flow using the Super Hex (SHEX) code. The licensee further stated that this SBA calculation also bounds all EPU MELLLA+ power and flow conditions. The licensee stated that the drywell temperatures of 291°F and 290°F were obtained from the short-term DBA LOCA RSLB analyses performed with the M3CPT and LAMB codes at EPU MELLLA and EPU MELLLA+ conditions, respectively. These were intended to show the effect of EPU MELLLA+

versus EPU MELLLA on the predicted RSLB peak drywell temperature. Based on the above, the NRC staff finds the licensee's evaluation acceptable.

Long-Term LOCA Analysis for Suppression Pool Temperature Response

In Reference 1, Section 4.1.2, the licensee provided evaluation of the long term pressure suppression pool temperature under EPU conditions operating in MELLLA+ operating domain. The licensee stated **[[**

]]. Therefore, the NRC staff finds the licensee's evaluation acceptable.

3.2.2 Hydrodynamic Loads

The key parameters **[[**

]]. The licensee's evaluation in Reference 1, Section 4.1.3 stated that these loads in the EPU MELLLA domain bound the same loads in MELLLA+ domain. Since the current containment pressure and temperature response remains bounding, the NRC staff finds the licensee's evaluation acceptable.

3.2.3 Safety/Relief Valve Loads

The SRV loads are affected by the SRV setpoints, sensible and decay heat. In Reference 1, Section 4.1.4, the licensee provided evaluation of MNGP specific containment dynamic loads due to SRV discharge. The licensee stated that because the SRV setpoints and sensible and decay heat do not change in the MELLLA+ operating domain, the containment loads due to SRV discharge are unaffected. Therefore, the NRC staff finds the licensee's evaluation acceptable.

3.2.4 Subcompartment Analysis

The subcompartment pressurization loads are affected by the short term containment response. In Reference 1, Section 4.1.3, the licensee provided evaluation of subcompartment loads due to DBA LOCA under EPU MELLLA+ operating domain and stated that **[[**

]]. Since the **[[**

]], the NRC staff finds the licensee's evaluation acceptable.

3.2.5 Post-LOCA Combustible Gas Control System

In Reference 1, Section 4.7, the licensee provided evaluation of the post-LOCA combustible gas control system. The NRC revised 10 CFR 50.44, "Combustible gas control for nuclear power reactors," in September 2003. The revised rule eliminated the requirements for hydrogen recombiners and relaxed the requirements for hydrogen and oxygen monitoring in containment. The revised 10 CFR 50.44 no longer defines a design-basis LOCA hydrogen release, and

eliminates requirements for hydrogen control systems to mitigate such a release. The MNGP has nitrogen inerted containment. The NRC issued Amendment No. 138 (ADAMS Accession No. ML041180612) on May 21, 2004, which approved the removal of the requirements for hydrogen recombiners, allowing them to be abandoned in place. The licensee's stated that MELLLA+ operating domain does not affect the current combustible gas control system. Therefore, the NRC staff finds the licensee's evaluation acceptable.

3.2.6 Emergency Core Cooling System (ECCS) and Containment Heat Removal Pumps Net Positive Suction Head

The ECCS and containment heat removal pumps are the Residual Heat Removal (RHR) and Core Spray (CS) pumps. Reference 18, Section 6.6 provides NRC guidance for a licensee that credits Containment Accident Pressure (CAP) for calculating the Net Positive Suction Head (NPSH) available (NPSHa) at the suction inlet of the RHR and CS pumps. The licensee provided responses in References 20 and 21 to the NRC staff guidance considering operation under EPU MELLLA+ domain. In response to the guidance in Section 6.6.2 of Reference 18, the licensee stated that ATWS is the only EPU MELLLA+ event that affects the NPSHa and is limiting compared to the EPU MELLLA ATWS evaluation provided in Reference 22, Section 2.6.5. The remaining events evaluated for NPSHa, other than ATWS, are based on the EPU MELLLA results and bound operation under EPU MELLLA+ conditions. The NRC staff's technical evaluation is provided in Section 2.6.5 of the MNGP EPU safety evaluation (ADAMS Accession No. ML13343A006), which includes: (a) the licensee's NPSH analysis provided in Reference 22, Section 2.6.5, and (b) the licensee's compliance with the guidance in Section 6.6 of Reference 18. In the NRC staff's MNGP EPU safety evaluation, the staff finds that the RHR and CS pumps NPSH analysis for proposed operation under EPU conditions in MELLLA+ domain is acceptable.

3.2.7 Main Control Room Atmosphere Control System

The Main Control Room Atmosphere Control System is described in the MNGP USAR, Section 6.7, "Main Control Room, Emergency Filtration Train Building and Technical Support Center Habitability". In Reference 1, Section 4.4, the licensee provided evaluation of the Main Control Room Atmosphere Control System. The licensee stated that the MELLLA+ operating domain expansion does not result in a change in the source terms or the release rates. **[[**

]]. Therefore, the NRC staff finds the licensee's evaluation acceptable.

3.2.8 Standby Gas Treatment System

The Standby Gas Treatment System (SGTS) maintains the secondary containment at a negative pressure and filters the exhaust air by removing fission products present during abnormal conditions. In Reference 1, Section 4.5, the licensee provided evaluation of the SGTS. **[[**

]]. The licensee stated that **[[**

]]. Therefore, the NRC staff finds the licensee's evaluation acceptable.

3.2.9 Containment Isolation

The containment isolation system is affected by the containment pressure and temperature response under design basis accident conditions. The licensee stated **[[**

]]. According to Section 4.1.3 of Reference 5, an evaluation of containment isolation systems is not required. Therefore, the NRC staff finds the licensee's conclusion to be acceptable.

3.2.10 Generic Letter 89-10

The response to Generic Letter (GL) 89-10, Supplement 3, "Consideration of the Results of NRC-Sponsored Tests of Motor-Operated Valves," would be affected by the containment pressure and temperature under design basis accident conditions. The licensee stated the MELLLA+ operating domain does not impact its response to GL 89-10 **[[**

]]. The licensee also confirmed that other parameters such as environment temperature during normal conditions and under high energy line break conditions, that could potentially affect the safety-related motor-operated valves (MOVs), are unchanged in the MELLLA+ operating domain. According to Section 4.1.4 of Reference 5, an evaluation of the response to GL 89-10 is not required. Therefore, the NRC staff finds the licensee's conclusion to be acceptable.

3.2.11 Generic Letter 89-16

The response to GL 89-16, "Installation of a Hardened Wetwell Vent", would be affected by the power level. The licensee stated that there is no impact on response to GL 89-16 because the power level does not change from EPU MELLLA to EPU MELLLA+ operating domain.
[[

]]. Therefore, the NRC staff finds the licensee's conclusion to be acceptable.

3.2.12 Generic Letter 95-07

The response to GL 95-07, "Pressure Locking and Thermal Binding of Safety-Related Power-Operated Gate Valves", would be affected by the containment pressure and temperature under design basis accident conditions. The licensee stated **[[**

]]. Therefore, according to Section 4.1.6 of Reference 5, evaluation in response to GL 95-07 is not required. Therefore, the NRC staff finds the licensee's conclusion to be acceptable.

3.2.13 Generic Letter 96-06

The response to GL 96-06, "Assurance of Equipment Operability and Containment Integrity during Design-Basis Accident Conditions", would be affected by the containment pressure and temperature under design basis accident conditions. The licensee stated **[[**

]]. As stated in Section 4.1.7 of Reference 5, an evaluation in response to GL 96-06 is not required. Therefore, the NRC staff finds the licensee's conclusion to be acceptable.

Conclusion

The NRC staff has reviewed the licensee's assessment of the aforementioned topics, and concludes that they are adequately addressed for MNGP in the EPU MELLLA+ operating domain. The NRC staff also concludes that MNGP will continue to meet the requirements of GDCs 4, 16, 19, 38, 41, and 50 following implementation of the proposed MELLLA+ operating domain under EPU conditions.

3.3 Instrumentation and Controls

The NRC staff performed a review of the MNGP Oscillation Power Range Monitor (OPRM) Detect and Suppress Solution - Confirmation Density (DSS-CD) safety function in accordance with the NUREG-0800 (Reference 3), Chapter 7, Branch Technical Position 7-19, Revision 6, "Guidance for Evaluation of Diversity and Defense-In-Depth in Digital Computer-Based Instrumentation and Control Systems."

The NRC Staff Requirements Memorandum (SRM) on SECY 93-087, dated July 21, 1993 (ADAMS Accession No. ML003708056) describes the NRC's position regarding Diversity and Defense-In-Depth (D3). The SRM states that applicants using digital or computer based technology shall assess the defense-in-depth and diversity of the proposed instrumentation and control system to demonstrate that vulnerabilities to common mode failures have been adequately addressed. The SRM also states that "in performing the assessment, the vendor or applicant shall analyze each postulated common-mode failure for each event that is evaluated in the accident analysis section of the safety analysis report (SAR) using best-estimate methods. The vendor or applicant shall demonstrate adequate diversity within the design for each of these events."

The OPRM DSS-CD safety function is credited in the MNGP USAR, Chapter 14, "Plant Safety Analysis," for mitigation of a plant instability event. While this event is analyzed in Section 14.6, "Plant Stability Analysis," of the MNGP USAR, a failure of the Nuclear Measurement Analysis and Control (NUMAC) OPRM or Average Power Range Monitor (APRM) could disable the automatic safety trip function performed by the DSS-CD algorithms. The MNGP NUMAC system includes a means of providing Automatic Backup Stability Protection (ABSP) in the event that the primary means of stability protection (DSS-CD) becomes inoperable; however, the NRC staff notes that use of common software for both primary (DSS-CD) and backup (ABSP) stability protection can lead to a condition where both of these automatic functions would become disabled due to a software common-cause failure (CCF).

If the OPRM system is inoperable, and the ABSP function performed by the APRM either cannot be implemented or is inoperable, the Manual Backup Stability Protection (BSP) becomes the licensed stability solution. The MNGP power-to-flow graph contains regions of operation that are defined by a BSP Boundary. With the BSP boundary being the credited stability solution, the reactor power is reduced below the BSP line so that two recirculation pump trips will not result in immediate operation inside the exclusion region. When plant conditions result in operations within the exclusion region, administrative actions require operator initiation of a manual reactor scram. This is described in Section 7.3 and the technical specification changes documented in the approved DSS-CD Licensing Topical Report (LTR) NEDC-33075P-A, Revision 6 (Reference 7).

Due to the potential for loss of both primary and backup automatic protection functions, the licensee performed a D3 analysis which considered the effects of a postulated software CCF of the NUMAC Power Range Neutron Monitoring (PRNM) (APRM/OPRM) system in conjunction with the plant instability events described in Chapter 14 of the MNGP USAR. Two plant transient scenarios were considered in this analysis: 1) dual-recirculation pump trip, and 2) a loss of feedwater heating. This analysis identified manual operator actions as a diverse means of maintaining plant safety if the automatic trip functions performed by the DSS-CD algorithms and the ABSP become unavailable due to a postulated common-mode failure of the NUMAC PRNM system.

The D3 analysis identified that the postulated CCF in the PRNM system results in the system providing valid indications of plant conditions until the stability transient occurs, at which time they become anomalous. In the case of power oscillations, the PRNM system power and flow indications would track consistently with other plant indicators as they change to a state point where the potential exists for high growth-rate power oscillations (i.e., the region of the power/flow map where thermal hydraulic instabilities become prevalent), but fail to provide any protection when large amplitude oscillations begin to occur. Because of this, plant operators will have the necessary indications to identify plant operation in the manual BSP regions and be able to initiate manual actions to assure plant safety.

For the dual-recirculation pump trip scenario, the credited diverse manual operator action is to implement the manual BSP regions and initiate a manual scram of the reactor if the BSP boundary is exceeded. The D3 analysis identified multiple diverse control room indications of a dual recirculation pump trip that are independent from the effects of the postulated PRNM system CCF.

When a two-recirculation pump trip condition is identified, the plant operators are procedurally required to insert a manual scram if the BSP boundary is exceeded. This immediate action is uncomplicated and can be completed by simultaneously depressing two reactor scram push buttons located at the C-05 control room panel. Confirmation that the manual scram is successful is unambiguous and provided by the control rod display on C-05 within a few seconds. The NRC staff confirmed that the systems being used for initiation of a manual scram and confirming that the scram was successful do not rely on digital or software based technologies. The NRC staff determined that these systems would therefore be unaffected by the postulated software CCF that rendered the automatic protection functions inoperable.

For the loss of feedwater heating scenario, the credited diverse manual operator actions are those operations that control start-up trajectories on the power-to-flow map. These include adjustment of recirculation flow and control rod position. The D3 analysis identified multiple diverse control room indications of a loss of feedwater heating that are independent from the effects of the postulated PRNM system CCF.

When a loss of feedwater heating condition is identified, the D3 analysis shows that it is unlikely that the region of instability would be entered; however, plant operators are procedurally required to identify if the plant has entered into a region of thermal hydraulic instability, and to insert a manual scram if the BSP limits are exceeded. The NRC staff confirmed that the systems used for controlling core flow, reactor power, and inserting a manual scram do not rely on digital or software based technologies. The NRC staff further determined that these systems would not be affected by the postulated software CCF of the PRNMS that rendered the automatic protection functions inoperable.

The D3 analysis also identified the MNGP ATWS system as an additional means of providing backup protection in the event that the manual scram function is either not initiated by the operator or fails to insert control rods into the core. The MNGP analyses states that manual operator actions used to mitigate an ATWS event are sufficient to prevent excessive clad temperatures with significant margin. The NRC staff acknowledges that core damage can be avoided during an ATWS event, and that this could provide an additional means of meeting the acceptance criteria of BTP 7-19; however, the conclusions of this evaluation are based upon the manual BSP protection methodology and not the ATWS mitigation system.

Conclusion

In its evaluation of BSP protection (Reference 7), the NRC staff concluded that the proposed BSP methodology is an acceptable solution because it provides sufficient protection against plant Safety Limit Minimum Critical Power Ratio (SLMCPR) violations commensurate with the probability of an instability event in the short period of time that they are active. Furthermore, the staff's evaluation concludes that manual control measures needed to support BSP protection are sufficiently diverse from the digital PRNMS NUMAC systems. Therefore, the NRC staff concludes that this provides an acceptable means of diverse protection for the DSS-CD safety function.

3.4 Technical Specifications

The licensee submitted changes to the MNGP technical specifications (TS) to support its MELLLA+ license amendment request. The proposed TS changes are primarily associated with implementation of the DSS-CD long term stability solution, and described in Reference 7, Section 8.0, "Effect of Technical Specifications." The NRC staff's review of the proposed changes is discussed below.

TS 3.3.1.1, Reactor Protection System (RPS) Instrumentation, Conditions I, J, and K

The licensee proposed to revise the TS 3.3.1.1 action statements for Conditions I and J, and to insert a new Condition K, to support implementation of the Backup Stability Protection (BSP)

requirements in the event that the DSS-CD is inoperable. The proposed changes are consistent with the requirements specified in Section 8.0 of Reference 7, with the following exceptions:

In its February 24, 2014, supplemental letter (Reference 26), the licensee proposed renumbering Required Actions I.2.1 and I.2.2, to I.2 and I.3, respectively, to maintain consistency with the numbering and logical connector convention described in NUREG-1433, "Standard Technical Specifications - General Electric Plants (BWR/4), and Section 1.0, "Use and Application," of the Monticello Technical Specifications.

In Reference 26, the licensee also proposed to revise the Completion Time for Action I.3 from "90 days" to "Immediately." This change incorporates the correct usage and structure of the technical specifications as described in NUREG-1433 for TS 3.3.3.1, "Post Accident Monitoring (PAM) Instrumentation," Conditions B and F, and TS 5.6.5, "Post Accident Monitoring Report." Furthermore, this change is consistent with Monticello TS 3.3.3.1, "Post Accident Monitoring (PAM) Instrumentation," Conditions B and F, and TS 5.6.4, "Post Accident Monitoring Report."

Based on the above, the NRC staff finds the changes acceptable.

TS Section 3.3.1.1, RPS Instrumentation, Surveillance Requirement (SR) 3.3.1.1.6

The licensee proposed to delete SR 3.3.1.1.6. The surveillance is no longer required and eliminates unnecessary actions. The proposed changes are consistent with the requirements specified in Section 8.0 of Reference 7. Therefore, the NRC staff finds the change acceptable.

TS Section 3.3.1.1, RPS Instrumentation, Table 3.3.1.1-1, APRM Function 2.b

The licensee proposed to revise the allowable value for Function 2.b, "Simulated Thermal Power – High" (for two-loop operation) from 0.55(W – DeltaW) + 61.5% RTP to 0.61(W – DeltaW) + 67.2% RTP. The basis for the allowable value setpoint is discussed in Reference 1, Section 5.3.1, "APRM Flow-Biased Scram." The proposed change is also consistent with the requirements specified in Section 8.0 of Reference 7. Based on the above, the NRC staff finds the change acceptable.

TS Section 3.3.1.1, RPS Instrumentation, Table 3.3.1.1-1, Note (h)

The licensee proposed to add new Note (h) to require resetting the allowable value setpoint for the APRM Simulated Thermal Power (STP) - High when the OPRM is inoperable. New Note (h) will read as follows:

> (h) With OPRM Upscale (function 2.f) 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 I of this Specification.

The proposed change is consistent with the requirements specified in Section 8.0 of Reference 7. Therefore, the NRC staff finds the change acceptable.

TS Section 3.3.1.1, RPS Instrumentation, Table 3.3.1.1-1, Function 2.f

The licensee proposed to revise Note (e) which describes the initial DSS-CD arming requirements. Note (e) will read as follows:

> 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.

In addition, the licensee proposed to delete the reference to SR 3.3.1.1.6. The surveillance is no longer required, as described above.

The proposed changes are consistent with the requirements specified in Section 8.0 of Reference 7. Based on the above, the NRC staff finds the changes acceptable.

TS Section 3.4.1, Recirculation Loops Operating

The licensee proposed to modify Limiting Condition for Operation (LCO) 3.4.1 for one recirculation loop operation. The proposed change further defines requirements while in single-loop operation, and restricts single-loop operation in the MELLLA+ operating domain. Operation in the MELLLA+ domain is not analyzed for single-loop operation.
LCO 3.4.1 will be revised (change highlighted in **BOLD**) to read as follows:

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

OR

One recirculation loop may be in operation provided the plant is not operating in the MELLLA+ domain defined in the COLR and provided the following limits are 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 Monitor Simulated Thermal Power - High), Allowable Value of Table 3.3.1.1-1 is reset for single loop operation.

As discussed in Section 3.6.3 of Reference 1, single-loop operation is not allowed in the MELLLA+ operating domain. The proposed modification to LCO 3.4.1 recognizes that one recirculation loop may be in operation provided the plant is not operating in the MELLLA+ operating domain as defined in the COLR. The NRC staff has reviewed the proposed change and finds it acceptable

TS Section 5.6.3, Core Operating Limits Report (COLR)

The licensee proposed to change TS Section 5.6.3 by changing Item 5.6.3.a.6 and replacing Items 5.6.3.b.4 and 5.6.3.b.5.

Item 5.6.3.a.6 will read as follows:

6. 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.f), Automated BSP Scram Region, and the BSP Boundary for Specification 3.3.1.1.

Items 5.6.3.b.4 and 5.6.3.b.5 will be deleted, and Item 5.6.3.b.4 will read as follows:

4. NEDO-33075-A, Revision 6, "General Electric Boiling Water Reactor Detect and Suppress Solution – Confirmation Density," January 2008.

The proposed changes are consistent with the requirements specified in Section 8.0 of Reference 7. The changes will ensure that applicable thermal limits continue to be met and reflect NRC-approved analytical methodologies. Therefore, the NRC staff finds the changes acceptable.

TS Section 5.6.6, OPRM Report

In its application (Reference 1), as revised by its February 24, 2014, supplemental letter (Reference 26), the licensee proposed to add a new item to Section 5.6, "Reporting Requirements," to read as follows:

5.6.6 <u>OPRM Report</u>

When a report is required by Condition I of LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation," a report shall be submitted within the following 90 days. The report shall outline the preplanned means to provide backup stability protection, the cause of the inoperability, and the plans and schedule for restoring the required instrumentation channels to OPERABLE status.

The change is consistent with the requirements specified in Section 8.0 of Reference 7, and conforms to the content and structure of structure of the technical specifications as described in NUREG-1433. Therefore, the NRC staff finds the changes acceptable.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Minnesota State official was notified of the proposed issuance of the amendment. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to the use of facility components located within the restricted area as defined in 10 CFR Part 20 and changes surveillance requirements. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration and there has been no public comment on such finding (75 FR 57527, September 21, 2010). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

6.0 CONCLUSION

The NRC staff has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) there is reasonable assurance that 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.

7.0 REFERENCES

1. GE-Hitachi, NEDC-33435P, Revision 1, "Safety Analysis Report for Monticello Maximum Extended Load Line Limit Analysis Plus," dated December 2009 (ADAMS Accession No. ML100280566, Attachment 3 (proprietary); ADAMS Accession No. ML100280557, Attachment 5 (non-proprietary)).

2. RS-001, "Review Standard for Extended Power Uprates," Revision 0, December 2003 (ADAMS Accession No. ML033640024).

3. NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," U.S. Nuclear Regulatory Commission.

4. http://www.nrc.gov/reading-rm/doc-collections/nuregs/staff/sr0800/

5. GE-Hitachi, NEDC-33006P-A, Revision 3, "General Electric Boiling Water Reactor Maximum Extended Load Line Limit Analysis Plus," dated June 2009 (ADAMS Accession No. ML091800530).

6. GE-Hitachi, Final SE for NEDC-33173P, “Applicability of GE Methods to Expanded Operating Domains,” dated July 21, 2009 (ADAMS Accession No. ML083520464).

7. GE Nuclear Energy, NEDC-33075P-A, Revision 6, “Detect and Suppress Solution - Confirmation Density Licensing Topical Report,” dated January 2008 (ADAMS Accession Nos. ML080310384 (package) and ML080310396; ML080310402 (proprietary)).

8. GE Nuclear Energy, NEDE-33147P-A, Revision 2, “DSS-CD TRACG Application,” dated November 2007 (ADAMS Accession No. ML073120638).

9. GE-Hitachi, GE-MNGP-AEP-1913 Revision 1, “GEH Responses to Reactor Systems RAIs – Proprietary,” dated August 27, 2010 (ADAMS Accession No. ML1027903742).

10. Xcel Energy, Inc. Letter L-MT-10-049, “Response to Request for Additional Information (RAI) for the MNGP Proposed Maximum Extended Load Line Limit Analysis Plus (MELLLA+) Amendment (TAC ME3145),” dated September 28, 2010 (ADAMS Accession No. ML1027903756).

11. GE-Hitachi, “DSS-CD Methodology Implementation for Plant-Specific Application to Monticello,” collection of slides, dated May 21, 2010 (ADAMS Accession No. ML102790391).

12. Escrivá, A., J.L. Muñoz-Cobo, J. Melara San Roman, M. Albendea Darriba, and J. March-Leuba, “LAPUR 6.0 R.0 User’s Manual,” NUREG/CR-6958, ORNL/TM-2007/233, Oak Ridge National Laboratory, December 2007.

13. Otaduy-Bengoa, P. J., “Modeling of the Dynamic Behavior of Large Boiling Water Reactor Cores,” Ph.D. dissertation, University of Florida, 1979.

14. Global Nuclear Fuel, 0000-0092-5740-SRLR, Revision 3, “Supplemental Reload Licensing Report for Monticello, Reload 25, Cycle 26, Maximum Extended Load Line Limit Plus (MELLLA+),” dated January 2012.

15. GE-Hitachi, NEDE-32176P, Revision 4, “TRACG Model Description,” dated January 2008 (ADAMS Accession No. ML080370276).

16. Xcel Energy, Inc. Letter L-MT-12-108, “Maximum Extended Load Line Limit Analysis Plus License Amendment Request - Request for Additional Information Responses for TRACE/TRACG Differences (TAC ME3145),” Monticello Nuclear Generating Plant, dated December 21, 2012.

17. Xcel Energy, Inc. Letter L-MT-13-096, “Maximum Extended Load Line Limit Analysis Plus License Amendment Request - Request for Additional Information Responses (TAC No. ME3145),” dated October 4, 2013 (ADAMS Accession No. ML13282A122).

18. SECY-11-0014, “The Use of Containment Accident Pressure in Reactor Safety Analysis,” dated January 31, 2011 (ADAMS Accession No. ML102110167).

19. ACRS Letter to the Honorable Allison M. Macfarlane "Monticello Nuclear Generating Plant Extended Power Uprate"

20. Xcel Energy, Inc. Letter L-MT-12-082, "Monticello Extended Power Uprate and Maximum Extended Load Line Limit Plus License Amendment Requests: Supplement to Address SECY 11-0014, Use of Containment Accident Pressure (TAC Nos. MD9990 and ME3145)," dated September 28, 2012 (ADAMS Accession No. ML12276A057).

21. Xcel Energy, Inc. Letter L-MT-12-107, "Monticello Extended Power Uprate and Maximum Extended Load Line Limit Plus License Amendment Requests: Supplement to Address SECY 11-0014, Use of Containment Accident Pressure (TAC Nos. MD9990 and ME3145)," Sections 6.6.4 and 6.6.7, dated September 28, 2012 (ADAMS Accession No. ML12276A057).

22. GE-Hitachi Nuclear Energy, "Safety Analysis Report for Monticello Constant Pressure Power Uprate," NEDC-33322P, Revision 3, Enclosure 5 to NSPM letter to U.S. NRC dated November 5, 2008, "License Amendment Request: Extended Power Uprate (TAC No. MD9990) (ADAMS Accession No. ML083230125).

23. Enclosure 2 of NSPM Letter to NRC, "Monticello Extended Power Uprate: SECY 11-0014 Use of Containment Accident Pressure – Responses to Requests for Additional Information (TAC No. MD9990)," dated March 21, 2013 (ADAMS Accession No. ML13085A034).

24. GE Nuclear Energy, NEDE-20566P-A, "General Electric Model for LOCA Analysis in accordance with 10 CFR 50 Appendix K," dated September 1986.

25. General Electric Company, MC3PT, "The General Electric Mark III Pressure Suppression Containment Analytical Model," NEDO-20533, dated June 1974, and Supplement 1, dated September 1975.

26. Xcel Energy, Inc. Letter L-MT-14-023, "Maximum Extended Load Line Limit Analysis Plus: Technical Specification Clarifications (TAC ME3145)," dated February 24, 2014 (ADAMS Accession No. ML14057A526).

Principal Contributors: Tai Huang, NRR
Ahsan Sallman, NRR
Richard Stattel, NRR

Date of issuance:

Appendix A

NRC Staff Evaluation

of

Requests for Additional Information Responses

Appendix A provides a summary of the NRC staff's evaluation of the licensee's responses to requests for additional information (RAI) 1 through 27, documented in References 9 and 10. A second round of RAIs resulting from a staff audit of the TRACG quench-front model are documented in Reference 16 and labeled here as RAI 10/12-1 through 5.

RAI-01

> *Section 2.1.1 of NEDC-33435P/Rev1 states that "no additional fuel and core design evaluation is required" because Monticello will use a full load of GE14 for the first MELLLA+ core reload. What fuel evaluations will be required for future core reloads if fuel other than GE14 is used?*

The RAI response provided in Reference 9 describes the process to implement new fuel designs from Global Nuclear Fuel, LLC (GNF) into a GEH BWR. It involves two steps: first, the US NRC approves the fuel generically via the GESTAR II Amendment 22 fuel compliance process. Second, plant-specific analyses are performed to justify use of the new fuel design in the plant reload. Cycle-dependent analyses (e.g., cold shut-down reactivity margin through the cycle, core stability performance, margin to the vessel over-pressure safety limit) are necessary for each reload regardless of fuel design.

The cycle dependent analyses have been performed for the MELLLA+ MNGP core, which is loaded with GE14 fuel that has been approved generically. Therefore, no additional fuel and core design evaluations are required. The NRC staff considers this RAI closed.

RAI-02

> *Section 2.1.1 of NEDC-33435P/Rev 1 states that "Because there is no increase in the average bundle power or in the maximum allowable peak bundle power there is no change required to the fuel thermal monitoring threshold." Figures 2-7 through 2-17 provide 2D bundle distributions for relevant parameters for the representative MELLLA+ core. Provide similar figures for the last non-MELLLA+ Monticello core.*

Figures for the MNGP Cycle 25 Extended Power Uprate (EPU) reload licensing core were provided. These figures are "representative" of the last non-MELLLA+ MNGP core. A comparison of the power distributions pre- and post-MELLLA+ indicate that there are no significant changes in power distribution or performance. The NRC staff considers this RAI closed.

Table A-2 - Comparison of Key Operating Parameters for Representative EPU and MELLLA+ Cores

	Peak Bundle Power (relative)		**Peak LHGR (kW/ft)**		**Bundle Operating MCPR**	
	EPU	**MELLLA+**	**EPU**	**MELLLA+**	**EPU**	**MELLLA+**
BOC	1.38	1.40	11.29	11.21	1.71	1.73
MOC	1.46	1.40	9.50	10.67	1.75	1.72
EOC	1.47	1.49	8.62	8.56	1.77	1.74

<u>RAI-03</u>

> *Section 2.1.1 of NEDC-33435P/Rev1 stats that "For Monticello, the predicted bypass void fraction at the D-Level Local Power Range Monitor (LPRM) is less than the [[ ]] design requirement." Identify the methodology used to perform the reported bypass void analysis (i.e., ISCOR hot channel, ISCOR average channel, TRACG …). Provide the results of the bypass void analysis and identify the limiting operating conditions assumed.*

The results were provided. The bypass void calculated by the hot-channel ISCOR methodology is **[[ ]]** at the lower MELLLA+ region elbow. The value is less than the **[[ ]]** required. The NRC staff considers this RAI closed.

<u>RAI-04</u>

> *Section 2.2 of NEDC-33435P/Rev1 states that "the bundle R-factors used during the reload analysis are consistent with lattice axial void conditions expected for the hot channel operating state. The nodal void reactivity biases applied in TRACG are applicable to the lattices representative of fuel loaded in the core." Provide additional information about the R-factors used for the Monticello analyses and their applicability. Provide a description or a reference to justify the TRACG reactivity biases used.*

The RAI response provides data for the controlling bundles (i.e., bundles with lowest MCPR margin). For all those bundles, the channel-average cross section is less than **[[ ]].** Most bundles have average voids between **[[ ]],** which is consistent with lattice axial void conditions used for the analyses.

The TRACG AOO and stability analyses use the "PIRT 18" option to sample a response surface for the void reactivity coefficient. The TRACG ATWS analyses do not use the PIRT18 option

and use a best-estimate void coefficient in accordance to the limitation imposed on the MELLLA+ staff SER (Reference 5). The NRC staff considers this RAI closed.

RAI-05

> *Section 2.3.3 of NEDC-33435P/Rev1 states that the SLCS shutdown margin is evaluated to ensure it remains within Tech Specs. Provide the Hot Shutdown Boron Weight (HSBW) and its injection time for MELLLA+ and the last non-MELLLA+ core in Monticello.*

The response to RAI-05 states that the cold shutdown boron weight (CSBW) is calculated generically based on a desired concentration of 660 ppm boron at 68°F. The response also specifies the conditions assumed for the generic calculation of the hot shutdown boron weight (HSBW). The calculated HSBW in MNGP is 460 gallons, which requires an injection time of 19.2 minutes. The RAI response indicates that the HSBW and injection times are the same for MELLLA+ and EPU conditions. The NRC staff considers this RAI closed.

RAI-06

> *Section 2.4.1 of NEDC-33435P/Rev1 and tables 2-2 through 2-4 provide the conclusions to a series of TRACG analyses to demonstrate satisfactory DSS-CD application when the amplitude discrimination setpoint is **[[** **]]**. Provide additional detail about the procedure used to perform these calculations and the methodology to ensure that the **[[** **]]**.*

GEH provided a detailed presentation of the DSS-CD methodology application for MNGP. The slides used for the presentation provide an excellent record (see Reference 11).

[[

]].

Figure A-1: **[[**

]].

Figure A-1: [[ ]]

This approach requires that MNGP **[[**

]] and is, therefore, acceptable. The NRC staff considers this RAI closed.

RAI-07

Provide the power and CPR time traces for the TRACG04 plant-specific demonstrations in Table 2-4 of NEDC-33435P/Rev1.

The time traces were provided. For all cases studied, the DSS-CD scram was effective and the final MCPR value is greater the SLMCPR. The NRC staff considers this RAI closed.

RAI-08

> *Provide the actual values of final MCPR for the matrix cases in Table 2-3 of NEDC-33435P/Rev1. Provide the power and MCPR time traces for **[[ ]]** and **[[ ]]**.*

The licensee provided the requested information. The NRC staff considers this RAI closed.

RAI-09

> *Section 2.4.3 of NEDC-33435P/Rev1 mentions two BSP options. Specify which BSP option will be implemented in Monticello. Provide a copy of the relevant sections in the Monticello Technical Specifications. Specifically, what is the maximum period of time that Monticello will be allowed under BSP conditions without the primary DSS-CD option operable?*

The response to RAI-09 states that MNGP will implement, as a first option, the Automated Backup Stability Protection (ABSP) option. If the ABSP function is inoperable, the manual BSP option will be implemented. Technical Specifications require restoration of the primary DSS-CD instrumentation within 120 days if both OPRM and ABSP functions are inoperable. Also, within 90 days, a special report should be provided with a plan for restoration of the primary stability licensing option. The NRC staff considers this RAI closed.

RAI-10

> *Section 2.4.1 and some tables and figures in NEDC-33435P/Rev1 reference TRACG002 results and criteria. Others contain references to TRACG004. Provide a summary of the code versions used for the Monticello FSAR analyses. Provide a short discussion of the licensing applicability of each code version, and specifically discuss the use of TRACG002 versus TRACG004. Section 2.6.1 states that the most recent versions of TGBLA/PANAC were used for the analyses. Specify which versions were used and discuss any interface issues with older codes like TRACG002.*

The discussion about the use of TRACG02 versus TRACG04 was provided in a presentation shared with NRC staff reviewers on May 4, 2010. There are no interface issues with older codes like TRACG02 because the MNGP equilibrium core described in Section 2.6.1 was not used to run any TRACG02 analysis. The NRC staff considers this RAI closed.

RAI-11

> *Table 9-1 of NEDC-33435P/Rev1 shows the AOO results in terms of peak power, flux, pressure and delta-CPR. The turbine trip with bypass (TTWBP)*

AOO appears to be the limiting delta-CPR event. The peak power during over-pressure events is typically very sensitive to the steam separator inertia (L/A) values used. Justify the steam separator L/A values used for these analyses.

The steam separator L/A is a function of separator inlet quality and the relationship between separator inlet quality and L/A determined in qualification testing was input into the ODYN computer code. The licensee provided a copy of the ODYN qualification report and the separator L/A correlations used. The MNGP calculations are consistent with the ODYN qualification bases. The NRC staff considers this RAI closed.

RAI-12

Table 9-1 of NEDC-33435P/Rev1 indicates that the turbine trip with bypass results in a higher peak power and lower CPR margin that generator load rejection or turbine trip without bypass. Provide an explanation why the bypass-failed transients result in a smaller power peak than the bypass available condition.

The Turbine Trip with Bypass (TTWBP) event is more conservative in the safety analyses because it also assumes a reduced air volume in the scram discharge header. The NRC staff considers this RAI closed.

RAI-13

Section 9.1.1 of NEDC-33435P/Rev1 states that "Results for all AOO pressurization transient events analyzed, including equipment out-of-service, showed at least 10% margin to the fuel centerline melt and the 1% cladding circumferential plastic strain acceptance criteria." Provide a table with the actual margins.

The most limiting flow condition for the pressurization transient events analyzed with respect to these criteria is increased core flow; therefore, MELLLA+ conditions (at lower flow) are bounded by the EPU conditions because the limiting condition is maximum power and maximum flow. The minimum calculated margin to the fuel centerline melt criterion was reported as 26 percent. The minimum calculated margin to the cladding strain criterion was reported as 35 percent. These values may change, and the actual values for the MELLLA+ reload will be included in Appendix I of the SRLR. The NRC staff considers this RAI closed.

RAI-14

Provide the results of the slow recirculation flow increase mentioned in Section 9.1.2 of NEDC-33435P/Rev1 and compare them with the MCPR flow factor.

The response to RAI-14 provides a table with the results of the MCPR response to a slow flow increase. The table shows that the MCPR increases are bounded by the reference IMCPR values that are enforced by the OLMCPR limit. The NRC staff considers this RAI closed.

RAI-15

For the licensing ODYN ATWS analysis and the best estimate TRACG analysis described in Section 9.3 of NEDC-33435P/Rev1, provide time traces and tabulated values for reactor power, pressure, peak PCT, and suppression pool temperature. Provide the HCTL as function of reactor pressure and the HSBW injection time.

The data was provided. The limiting event is the Pressure Regulator Failure Open (PRFO) at either BOC (peak power, peak pressure, PCT) or EOC (suppression pool temperature). The HCTL is a function of vessel pressure and suppression pool water level. A conservative value of 175°F is used for most analyses, which corresponds to the SRV lifting pressure and nominal pool level. The HSBW injection time for MNGP is 19.2 minutes. The NRC staff considers this RAI closed.

RAI-16

The Heat Capacity Temperature Limit (HCTL) is set to provide sufficient temperature margin in the suppression pool so that a conservative blow-out of the vessel with that initial suppression pool temperature will not result in a final temperature that compromises containment limits. Therefore, the HCTL limit is a function of operating reactor pressure. How is the HCTL limit determined in Monticello? Why are two arbitrary HCTL values of 150°F and 175°F used in the ATWS analyses of Section 9.3.1.2 of NEDC-33435P/Rev1? How do these values compare with the actual HCTL limit?

The HCTL limit is a function of vessel pressure and suppression pool water level. The response to RAI-15 provided the actual figure used in MNGP Emergency Operating procedures. The reference ATWS TRACG analysis uses nominal pool water level and a vessel pressure equal to the SRV lifting pressure. The NRC staff considers this RAI closed.

RAI-17

Section 9.3.1.2 of NEDC-33435P/Rev1 presents a sensitivity analysis for different water level control strategies during ATWS. Have the Emergency Operating Procedures (EOPs) in Monticello been updated to reflect the lessons learned from these simulations?

In the response to RAI-17, MGNP states that a number of studies were performed to simulate different EOP strategies during ATWS. The results of the studies suggest that no EOP changes

are required because the study demonstrates successful event mitigation with the existing EOP procedures. This study included evaluation of the plant response with and without reactor depressurization. The need for reactor depressurization is covered by the existing EOPs and may occur depending on initial conditions and event severity. The NRC staff considers this RAI closed.

RAI-18

> *What are the net positive suction head (NPSH) requirements for critical equipment in Monticello? Provide an evaluation of NPSH requirements versus the predicted ATWS conditions.*

The NPSH requirements have been resolved by the application of SECY-11-0014 (Reference 18). The NRC staff considers this RAI closed.

RAI-19

> *What is the status of the Monticello plant simulator with respect to MELLLA+. Provide a schedule of upgrades and validations that ensures proper operator training prior to operation in the MELLLA+ domain.*

The licensee stated that the simulator was expected to be ready by the end of spring 2011 with sufficient time to instruct the operators. The simulator was not ready in 2011; however, the simulator is now ready and being used to instruct the operators. The NRC staff considers this RAI closed.

RAI-20

> *Since the MELLLA+ SER was issued, a number of Part 21 notifications have been issued and evaluated. These issues are not part of the accepted SER, but have safety relevance to Monticello operation in the MELLLA+ domain. Provide a list of the applicable Part 21 issues that have been issued since the approval of the MELLLA+ SER and may affect MELLLA+ operation and a short description of their disposition.*

10 CFR Part 21 notifications that could be applicable to MELLLA+ were reviewed. Part 21 notifications for hardware components were covered by utility actions. Generic methodology for Part 21 notifications of relevance include 2007-20-00 and 2007-20-01, which involve the non-conservatism in the GESTR-M Thermal-Mechanical Methodology. The new requirement in accordance with these two Part 21 notifications is applied to MELLLA+. The NRC staff considers this RAI closed.

RAI-21

The NRC staff intends to perform confirmatory calculations of Monticello stability with the LAPUR code. Provide the following Monticello design data to support these calculations. Refer to Fig 1-1 of NEDC-33435P, Rev. 1. Point A is defined in the figure and point A' is at the intersection of the natural circulation line and the MELLLA+ rod line.

1. *Provide the inlet loss coefficients for the Monticello channels. Provide the "ODYSY" combined loss coefficient, not the "TRACG" separate coefficients, along with the reference flow area for the K-values provided.*

2. *Point A for the last non-MELLLA+ Monticello core using equilibrium FW temperature, provide:*

   a. *Thermal power*
   b. *Fraction of power deposited in the fuel*
   c. *Total core flow*
   d. *Bypass flow*
   e. *3D steady-state power distribution in digital form (i.e., axial node power for each bundle)*
   f. *Core-average void reactivity coefficient (special PANACEA edit)*
   g. *First harmonic mode sub-criticality*

3. *Point A' for a representative Monticello MELLLA+ core using equilibrium FW temperature, provides the same information as the above point [Point A].*

4. *Provide the same information for Points A and A' above, but setting FW temperature a near vessel-pressure saturation conditions. This condition will simulate lowering the water level below the FW sparger and pre-heating the FW with vessel steam as required by EOPs. This condition will bound the stability during an ATWS event because it will over-estimate the power and flow by keeping the water level high.*

The data was provided, and the NRC staff performed LAPUR confirmatory calculations (see Appendix B of this Safety Evaluation). The NRC staff considers this RAI closed.

RAI-22

Please explain why the PCT for the top peaked axial power distribution produces a lower PCT than the mid-peak distribution in the Table for section 4.3.2 corresponding to the 100 / 80 condition under the Appendix K column. Also, from this Table, please explain why the mid-peak from the 100 / 80 condition

produces a lower PCT than that for the top-peak PCT at the 100 / 100 condition under the Appendix K column. Lastly, please explain why the top-peak 100 / 100 condition is higher than the top-peak at 100 / 80 condition under the Appendix K column. The lower flow rate would be expected to reduce the subcooled level in the core, increase the boiling length, decreasing the two-phase level and increasing PCT.

The response was provided, and the NRC staff's evaluation found the response to be acceptable. The NRC staff considers this RAI closed.

RAI-23

In section 4.3.8, please quantify what "small change" in PCT means. Also, it is stated that because the drive flow mismatch is small compared to MELLLA+, the PCT change due to the drive flow mismatch is expected to be smaller than the MELLLA+ sensitivity. What is this sensitivity? Was the PCT change demonstrated to be smaller through analyses? If so, what are the results? Please explain.

The response was provided, and the NRC staff's evaluation confirmed that the change in PCT is small and considered acceptable. The NRC staff considers this RAI closed.

RAI-24

Please provide the analysis results for the Appendix K results in the section 4.3.2 table for the top and mid peaked axial power distributions.

The response was provided, and the NRC staff's evaluation found it acceptable. The NRC staff considers this RAI closed.

RAI-25

NRC Generic Letter (GL) 88-16 provides guidance for technical specification changes for cycle-specific parameter limits in the Core Operating Limits Report (COLR), which requires that: (1) to identify which particular cycle-specific core operating limits listed in TS 5.6.3.a will be supported by the referenced approved methodologies listed in TS 5.6.3.b for calculating the cycle-specific operating limit listed in TS 5.6.3.a; and (2) to identify the supported approved methodologies by report number, title, revision, date, and any supplements. TS 5.6.3.b states that the COLR will contain the complete identification for each of the Technical Specification referenced topical reports used to prepare the COLR (i.e., report number, title, revision, date, and any supplements). The proposed change to TS 5.6.3.b.4 follows the statement in TS 5.6.3.b and the GL 88-16 guidance. Please provide: (1) justification for not identifying the date, revision or supplement for TS

5.6.3.b.1 through TS 5.6.3.b.3; (2) identification of which applicable cycle-specific parameters listed in TS 5.6.3.a are supported by TS 5.6.3.b.1 through TS 5.6.3.b.3; and (3) a COLR report reflected in the attachment 1 of L-MT-10-003 Markups to the Technical Specifications for MELLLA Plus.

The response was provided, and the NRC staff's evaluation found it acceptable subject to further clarification. The NRC staff considers this RAI closed.

RAI-26

Please provide the following information relating to the Monticello MELLLA+ operation:

1. *Details to obtain a final core loading pattern including procedure, guidance, criteria, and approved methodologies used for this analysis.*

2. *When the final or reference core loading pattern will be available for analyzing the cycle-specific operating limits listed in the Table of Section 2.2.3. When the final reload analysis report will be available for parameters listed in Sections 2.3, 2.4, and 2.5 in the reload analysis report.*

The response was provided and it is evaluated and found acceptable. The NRC staff considers this RAI closed.

RAI-27

Please provide clarification for the relationship between footnote (b) in Table 3.3.1.1-1 and Function 2.b in term of the RTP and footnote (h) in Attachment 1 of L-MT-10-003.

The response was provided and the NRC staff's evaluation found it acceptable. The NRC staff considers this RAI closed.

RAI 10/12-1

For a typical quench front calculated by TRACG04 for representative ATWSI [anticipated transients without scram with instability] conditions, provide the heat rate to the liquid and vapor state and the quench component of the heat rate. Compare with the measured heat rate values published by Thompson in NED 1974, "On the Process of Rewetting a Hot Surface by a Falling Liquid."

The requested data was provided based on a TRACG calculation for Halden experiment #4. For this particular case, **[[**

]]. The calculation only shows that the quench heat front is a high-heat-loss zone, as reported by Thompson in NED 1974.

The NRC staff considers this RAI closed.

RAI 10/12-2

> *Provide a detailed description of the TRACG implementation of the quench front model. Provide a numerical comparison of the heat transfer coefficients used by TRACG downstream of the front and "normal" coefficients in nucleate boiling.*

A detailed description of the TRACG quench front models was provided, and it will be included in a future update of the TRACG Model Description Report.

The heat transfer coefficients for the Halden experiment #4 case were provided. **[[**

]]. We note, however, that this quench-front heat transfer coefficient is not used directly to calculate a heat transfer rate; instead it is only an intermediate step in the correlation for quench-front heat transfer, which reproduces the experimental data.

The NRC staff considers this RAI closed.

RAI 10/12-3

> *Generate TRACG quench model inputs for a number of Halden dryout experiments and provide a comparison of the results to validate the quench front velocity model at high power and pressures.*

Halden experiments 3, 4, 11 c, and 12 were modeled with TRACG and the results are presented in the RAI response. The experiments were selected to cover high pressure and high power conditions with Zr rods. For all these experiments, the rods quench from temperatures significantly higher than T_{min}. The TRACG code reproduces the Halden quenching data for these conditions.

A number of sensitivity analyses were performed to determine the impact of nodalization, T_{min} correlation, and quench front model. The results show **[[**

]]. If the TRACG quench front model is turned off, the calculated results do not agree with the data, indicating that the quench model is necessary to reproduce the physics of the experiments.

The NRC staff considers this RAI closed.

RAI 10/12-4

Provide the TRACG input decks for the Halden experiments including the digitized data from the experiments used to compare the results. In addition, please provide the TRACG output file and CEDAR file (in ascii if possible).

The licensee provided a data CD with input and output files. The NRC staff considers this RAI closed.

RAI 10/12-5

Reproduce the ATWSI calculations for MNGP with and without applying the void and the Zr credit in the Shumway Tmin correlation using the latest version of the TRACG code.

Provide a comparison of results.

Provide a plot that shows the hot rod clad temperature on the same plot as the calculated Tmin as function of time.

Provide a comparison for the variables shown in Figs 9-12 through 9-14 of NEDC-33435P, Revision 1.

The requested calculations were performed and the data was provided. As expected, the choice of T_{min} correlation does not affect significantly the power oscillations for core average or hot channel. However, removing the Zr beta-term credit in the Shumway correlation reduces the value of T_{min} significantly, which in turn increases the calculated PCT because the rod stays blanketed by the steam film a larger portion of the oscillation. Even when T_{min} is reached during the oscillations, sufficient steam flow is present to maintain a degree of cooling that keeps the rod temperatures at acceptable levels (< 2200°F).

The sensitivity calculations confirm that MNGP satisfies the ATWS acceptance criteria for two T_{min} correlations with, and without, the Zr beta-term credit in the Shumway correlation, which should bound the expected value of T_{min}.

The NRC staff considers this RAI closed.

RAI 9/11/13-1

The failure of the reactor to shut down during certain transient can lead to unacceptable reactor coolant system pressure, fuel conditions, and/or containment conditions. Provide a training schedule and tracking method to train operators on the importance of taking action within 90 seconds to mitigate an ATWS event.

MNGP explains that the 90-second action is from now on a Time Critical Operator Action (TCOA) used during training. Operators are not only trained to achieve it, they are tested based on meeting this criteria. MNGP provided the following commitment:

> NSPM commits to train and test licensed reactor operators to initiate Monticello Nuclear Generating Plant feedwater flow reduction in a time frame required to support the MELLLA+ Anticipated Transient Without Scram Instability (ATWSI) analysis.

The NRC staff considers the RAI response and commitment to be acceptable. The NRC staff considers this RAI closed.

RAI 9/11/13-2

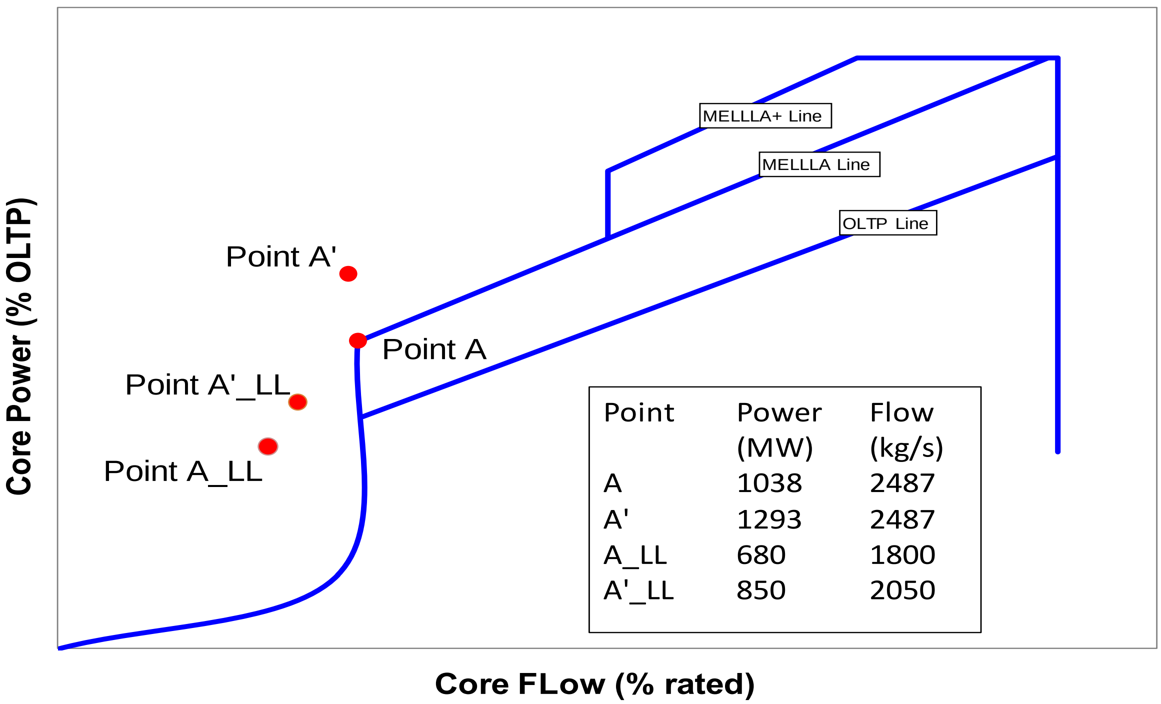
> *Providing an ATWSI analysis is a MELLLA+ SER requirement per NEDC-33006P-A, Rev. 3, but the ATWSI analysis of record in the Safety Analysis Report (SAR) is based on the incorrect TRACG04 quench model. Provide documentation of the updated ATWSI analysis that includes assumptions, sequence of events, plots of relevant variables, and margin to acceptance criteria. Provide a justification for the assumptions used, with special emphasis on operator action timing (including actual timing values from simulator trials) and the type of initiating transient selection (i.e., Recirculation Pump Trip (RPT) versus Turbine Trip with Bypass (TTWBP)). A SAR revision or update would be adequate.*

Documentation of the updated ATWSI analyses was provided for both the TTWBP and the new limiting event, the 2RPT. Sequence of events, figures and description of the transient were provided as “changed pages” for the SAR. The NRC staff considers this RAI closed.

Appendix B

LAPUR Confirmatory Calculations

The NRC's LAPUR6 code was used to perform a series of calculations to evaluate the impact of MELLLA+ implementation on the stability response of MNGP. To this end, four separate operating conditions were prepared and modeled with LAPUR6. The operating points are drawn graphically on the power-flow map provided below.



Point	Power (MW)	Flow (kg/s)
A	1038	2487
A'	1293	2487
A_LL	680	1800
A'_LL	850	2050

- Point A corresponds to the corner of the operating map pre-MELLLA+. This point is achieved when the recirculation pumps are tripped and feedwater (FW) temperature is allowed to reach equilibrium.

- Point A' is the equivalent to Point A for MELLLA+. If the reactor is operating at the upper end of the MELLLA+ domain and the pumps are tripped, Point A' is reached when FW temperature achieves equilibrium.

- Point A'_LL is a simulation of ATWS from the MELLLA+ domain, where the downcomer water level has been reduced 2 feet below the FW spargers, and FW flow is pre-heated by the upper plenum steam. For this condition, the core inlet temperature is at saturation, and the downcomer water level is approximately 1.5 meters below normal (2 feet below spargers).

- Point A_LL is similar to A'_LL, but with pre-MELLLA+ conditions. It represents ATWS with the downcomer water level reduced 2 feet below the FW spargers, and FW flow is

pre-heated by the upper plenum steam. For this condition, the core inlet temperature is at saturation and the downcomer water level is approximately 1.5 meters below normal (2 feet below spargers).

In RAI-21, the NRC staff requested that the licensee provide the operating conditions for these points. The licensee provided the power, flow, and 3D power distribution for these points. In addition, the licensee provided geometrical data (e.g., loss coefficients) and neutronic data (e.g. void reactivity coefficient) sufficient to perform the LAPUR6 confirmatory calculations. The data was generated from PANACEA steady state calculations.

A full-core LAPUR6 model was generated for MNGP at the above four operating conditions. Each core bundle is modeled by LAPUR6 and each bundle has its own axial and radial power distribution. GE14 channel geometry was used for all bundles. In the response to RAI-21, the licensee provided the effective core-inlet contraction coefficients **[[ ]]**. LAPUR6 **[[ ]]** models a single contraction coefficient at the core entrance and does not differentiate between the active-core and bypass flows; therefore the LAPUR6 inlet contraction coefficient is a function of the bypass flow. Bypass in LAPUR6 was modeled as a single channel, with the license provided flow and direct heating component.

To determine the operating conditions of Point A'_LL, a number of iterations were performed with LAPUR6. To determine the flow, a series of LAPUR6 calculations were performed as function of flow and the core delta-pressure was calculated. A reduction of downcomer water level to 2 feet below the spargers is equivalent to a 1.5 meter level reduction (at MNGP, the spargers are located at a reactor water level of -33 inches). At the reactor operating pressure, a reduction of 1.5 meters is equivalent to a pressure drop of approximately 0.1 bar. Thus, LAPUR6 runs were iterated until the core pressure drop calculated at Point A' (0.47 bar) was reduced to a value of 0.37 bar. To determine the operating power, the core-average void fraction was matched by iterating on the power level. Point A' has a core-average void fraction of 64.3 percent. The power of Point A'_LL was changed until the same void fraction was achieved, indicating that the core would be critical (i.e., $k_{eff} = 1$) at this power level. Note that LAPUR6 computes the power-square average of the void, which is a better approximation in determining $k_{eff} = 1$ than the non-weighted average.

Point A_LL represents the low-level state for the pre-MELLLA+ core, where the downcomer water level has been reduced to 2 feet below the FW spargers and the core-inlet temperature is at saturation.

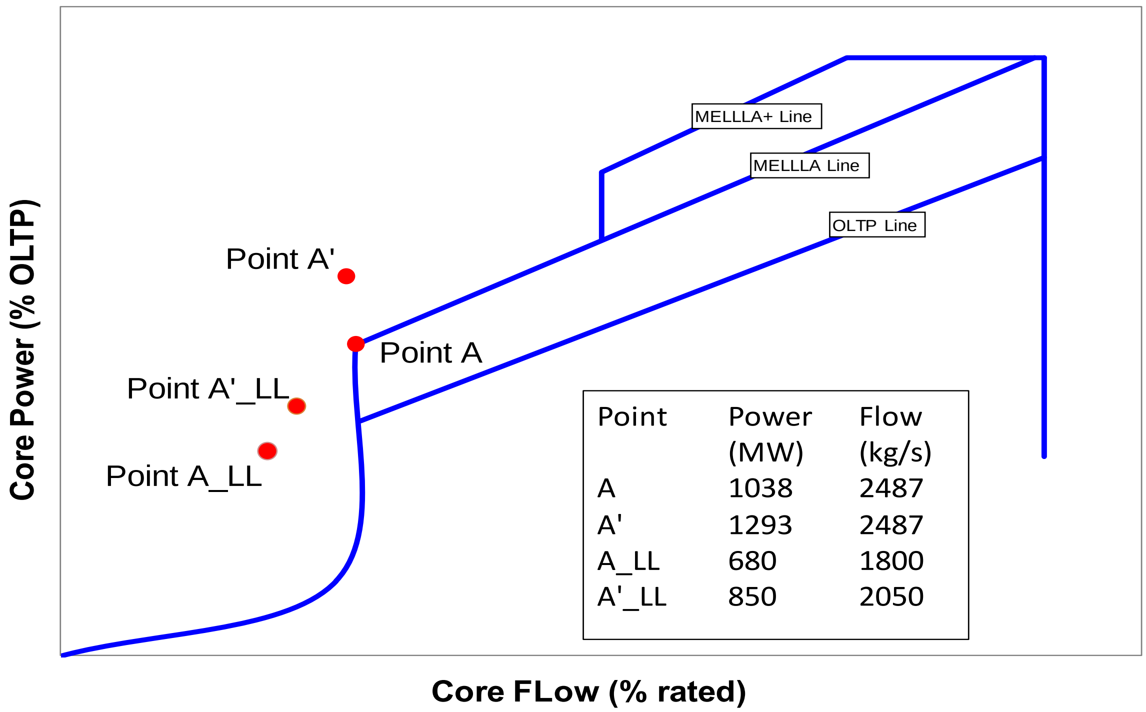


Figure B-2: Operating Points Analyzed

The result of the iterations shown in Figure B-2 and Table B-3 corresponds to a power of 850 MW and a flow of 2040 kg/s for point A'_LL, and 680 MW and 1800 kg/s for point A_LL.

Table B-3: Results of LAPUR6 Confirmatory Calculations

Point	Power (MW)	Flow (kg/s)	Decay Ratio (DR)		Frequency (Hz)	
			Core Wide	OOP	Core Wide	OOP
A	1038	2487	0.94	0.19	0.44	0.38
A'	1293	2487	1.53	0.43	0.48	0.45
A_LL	680	1750	1.43	0.37	0.41	0.36
A'_LL	850	2050	1.57	0.46	0.42	0.41

As indicated in Table B-3, the pre-MELLLA+ Point A is barely stable (DR = 0.94). When applying uncertainties (typically, ± 0.2), this point should be considered unstable. The post-MELLLA+ Point A' is clearly unstable, and unstable power oscillations of large amplitude should be expected at this point. Therefore, the NRC staff concludes that MELLLA+ has a significant impact on the reactor stability.

The potential for instability is one of the reasons why Point A' is not in the allowed operating domain. The MELLLA+ domain is clipped at approximately 55 percent flow. Point A' can only be reached under failure conditions, such as an unexpected dual recirculation pump trip event.

Furthermore, only stability Long Term Solutions (LTSs) approved for MELLLA+ can be used. The MELLLA+ LTSs have been reviewed and approved to ensure that Specified Acceptable Fuel Design Limits (SAFDLs) are not violated even if Point A' is reached during an event.

The calculation of Point A'_LL simulates ATWS conditions. Under many ATWS conditions, extraction steam for the FW heaters is not available (e.g., turbine trip, or containment isolation). Thus, under ATWS conditions, the FW temperature continues to drop, and it eventually reached condenser well temperature. If allowed to progress, this FW temperature decrease results in a large increase in operating power, which in turn destabilizes the reactor and extremely large power oscillations are predicted. To prevent this situation, the EOPs direct operators to reduce downcomer water level to at least two feet below the FW spargers. When the spargers are uncovered, FW flow mixes with the vessel steam and preheats to essentially saturated conditions. This mixing has also the beneficial effect of condensing steam, which does not contribute to the containment heat load.

The LAPUR6 confirmatory calculations confirm that during ATWS conditions, after the water level has been reduced, the growth in decay ratio and the associated oscillation amplitude is stabilized. Independent of how low a value the FW temperature reaches, the decay ratio will reach an asymptotic value of approximately 1.57, and the oscillations will not continue to grow. The ultimate DR value for the upgraded MELLLA+ condition is slightly larger than the one calculated for pre-MELLLA+ (DR = 1.43 for Point A_LL), but not significantly different. Therefore, the NRC staff concludes that the MELLLA+ upgrade does not significantly change the ATWS/Instability event at MNGP. The consequences of an ATWS/Instability event are managed by the EOP procedures, and water level reduction is as effective under MELLLA+ conditions as in EPU.

The NRC staff notes that all points analyzed for MNGP are predicted to be unstable; therefore, should an ATWS event occur, unstable power oscillation are highly likely. Furthermore, water level reduction by itself will not terminate the event. Boron injection will be required to stabilize the oscillations and shutdown the reactor.

Enclosure 3

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 180 TO

RENEWED FACILITY OPERATING LICENSE NO. DPR-22

NORTHERN STATES POWER COMPANY

MONTICELLO NUCLEAR GENERATING PLANT

DOCKET NO. 50-263

(Proprietary Version)

ADAMS Accession Number: ML13317A866

A Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,

/RA/

Terry A. Beltz, Senior Project Manager
Plant Licensing Branch III-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-263

Enclosures:
1. Amendment No. 180 to License No. DPR-22
2. Safety Evaluation (non-proprietary): ML14087A013
3. Safety Evaluation (proprietary): ML13317A866

cc w/o Enclosure 3: Distribution via Listserv

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Enclosure 3 – SE (proprietary): ML13317A866
*** Concurrence via memorandum** **** Concurrence via e-mail**

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