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

February 23, 2017

Mr. Peter A. Gardner
Site Vice President
Monticello Nuclear Generating Plant
Northern States Power Company - Minnesota (NSPM)
2807 West County Road 75
Monticello, MN 55362-9637

SUBJECT: MONTICELLO NUCLEAR GENERATING PLANT - ISSUANCE OF AMENDMENT RE: EXTENDED FLOW WINDOW (CAC NO. MF5002)

Dear Mr. Gardner:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment No.191 to Renewed Facility Operating License No. DPR-22 for the Monticello Nuclear Generating Plant. The amendment consists of changes to the Technical Specifications in response to your application dated October 3, 2014.

The amendment revises the Technical Specifications (TSs) and Renewed Facility Operating License to allow operation in the extended flow window (EFW) domain.

The NRC staff has determined that its safety evaluation for the subject amendment contains proprietary information pursuant to Title 10 of the *Code of Federal Regulations*, Section 2.390. Accordingly, the NRC staff has prepared a redacted, publicly available, non-proprietary version of the safety evaluation. Both versions of the safety evaluation are enclosed.

NOTICE: Enclosure 2 transmitted herewith contains sensitive unclassified information. When separated from Enclosure 2, this document is DECONTROLLED.

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

Sincerely,

Robert F. Kuntz, Senior Project Manager
Plant Licensing Branch III
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-263

Enclosures:
1. Amendment No. 191 to DPR-22
2. Proprietary Safety Evaluation
3. Non-Proprietary Safety Evaluation

cc w/o enclosure 2: 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. 191
License No. DPR-22

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

   A. The application for amendment by Northern States Power Company (NSPM, the licensee), dated October 3, 2014, as supplemented by letters dated January 9, August 26, September 29, and December 8, 2015, and February 29, April 29, August 4, September 14, and September 28, 2016, 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 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. 191, are hereby incorporated in the license. NSPM shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented prior to start up from Monticello Nuclear Generating Plant Operating Cycle 29.

FOR THE NUCLEAR REGULATORY COMMISSION

David J. Wrona, Chief
Plant Licensing Branch III
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Renewed Operating
License No. DPR-22 and
Technical Specifications

Date of Issuance: February 23, 2017

ATTACHMENT TO LICENSE AMENDMENT NO. 191

MONTICELLO NUCLEAR GENERATING PLANT

RENEWED FACILITY OPERATING LICENSE NO. DPR-22

DOCKET NO. 50-263

Replace the following page of Renewed Facility Operating License DPR-22 with the attached revised page. The revised page is identified by amendment number and contains a marginal line indicating the area of change.

REMOVE	INSERT
3	3

Replace the following pages of Appendix A, Technical Specifications, with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

REMOVE	INSERT
2.0-1	2.0-1
-------	2.0-2
3.3.1.1-1	3.3.1.1-1
3.3.1.1-3	3.3.1.1-3
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
--------	5.6-4
-------	5.6-5

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. 191, 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

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

2.1.1.1 With the reactor steam dome pressure < 686 psig or core flow < 10% rated core flow (GEH methods):

or

With the reactor steam dome pressure < 586 psig or core flow < 10% rated core flow (AREVA methods):

THERMAL POWER shall be ≤ 25% RTP.

2.1.1.2 With the reactor steam dome pressure ≥ 686 psig and core flow ≥ 10% rated core flow (GEH methods) MCPR shall be ≥ 1.15 for two recirculation loop operation or ≥ 1.15 for single recirculation loop operation.

2.1.1.3 With the reactor steam dome pressure ≥ 586 psig, core flow ≥ 10% rated core flow (AREVA methods):

a. For operation not in the EFW domain, MCPR shall be ≥ 1.15 for two recirculation loop operation, or ≥ 1.20 for single recirculation loop operation,

or

b. For operation in the EFW domain and the ratio of power to core flow < 42 MWt/Mlb/hr, MCPR shall be ≥ 1.15,

or

c. For operation in the EFW domain and the ratio of power to core flow ≥ 42 MWt/Mlb/hr, MCPR shall be ≥ 1.19.

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

2.1.2 Reactor Coolant System Pressure SL

Reactor steam dome pressure shall be ≤ 1332 psig.

2.0 SAFETY LIMITS (SLs)

2.2 SL VIOLATIONS

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

2.2.1 Restore compliance with all SLs; and

2.2.2 Insert all insertable control rods.

3.3 INSTRUMENTATION

3.3.1.1 Reactor Protection System (RPS) Instrumentation

LCO 3.3.1.1 The RPS instrumentation for each Function in Table 3.3.1.1-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.1.1-1.

ACTIONS

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

1. Separate Condition entry is allowed for each channel.

2. When the Function 2.b and 2.c channels are not within the limit of SR 3.3.1.1.2 due to APRM indication not within limits, entry into associated Conditions and Required Actions may be delayed for up to 2 hours if the APRM is indicating a lower power value than the calculated power, and for up to 12 hours if the APRM is indicating a higher power value than the calculated power.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required channels inoperable.	A.1 Place channel in trip. OR A.2 ----------NOTE---------- Not applicable for Functions 2.a, 2.b, 2.c, 2.d, 2.f or 2.g. ---------- Place associated trip system in trip.	12 hours 12 hours
B. ----------NOTE---------- Not applicable for Functions 2.a, 2.b, 2.c, 2.d, 2.f or 2.g. ---------- One or more Functions with one or more required channels inoperable in both trip systems.	B.1 Place channel in one trip system in trip. OR B.2 Place one trip system in trip.	6 hours 6 hours

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 alternate method to detect and suppress thermal hydraulic instability oscillations.	12 hours
	AND		
	I.2	----------NOTE---------- LCO 3.0.4 is not applicable ----------	
		Restore required channels to OPERABLE.	120 days
J. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	J.1	Reduce THERMAL POWER below the MELLLA boundary defined in the COLR.	12 hours
K. Required Action and associated Completion Time of Condition I or J not met.	K.1	Reduce THERMAL POWER to < 20% RTP.	4 hours

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
SR 3.3.1.1.16	Verify the oscillation power range monitor (OPRM) function is not bypassed when APRM Simulated Thermal Power is ≥ 25% RTP and drive flow is ≤ 60% of rated drive flow.	24 months

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] 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.

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	≥ 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 SR 3.3.1.1.16	As specified in COLR
g. Extended Flow Window Stability – High	Within EFW boundary defined in COLR	3[c]	J	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	As specified in COLR
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

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

(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
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
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

(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
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
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 Extended Flow Window 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 Reactor Protection System Instrumentation Period Based Detection Algorithm OPRM Upscale trip setpoints associated with Table 3.3.1.1-1 Function 2.f, and the EFWS – High setpoints associated with Table 3.3.1.1-1 Function 2.g.

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"

2. NEDO-31960-A, "BWR Owners' Group Long-Term Stability Solutions Licensing Methodology", with Supplement 1, dated November 1995

3. NEDO-32465-A, "Reactor Stability Detect and Suppress Solutions Licensing Basis Methodology and Reload Applications," August 1996

4. Engineering Evaluation EC 25987, "Calculation Framework for the Extended Flow Window Stability (EFWS) Setpoints", as docketed in Xcel Energy letter to NRC L-MT-15-065, dated September 29, 2015

5. XN-NF-81-58(P)(A) Revision 2 and Supplements 1 and 2, "RODEX2 Fuel Rod Thermal-Mechanical Response Evaluation Model," March 1984

6. EMF-85-74(P) Revision 0 Supplement 1(P)(A) and Supplement 2(P)(A), "RODEX2A (BWR) Fuel Rod Thermal-Mechanical Evaluation Model," February 1998

7. ANF-89-98(P)(A) Revision 1 and Supplement 1, "Generic Mechanical Design Criteria for BWR Fuel Designs," May 1995

5.6 Reporting Requirements

5.6.3 CORE OPERATING LIMITS REPORT (COLR) (continued)

8. XN-NF-80-19(P)(A) Volume 1 and Supplements 1 and 2, "Exxon Nuclear Methodology for Boiling Water Reactors - Neutronic Methods for Design and Analysis," March 1983

9. XN-NF-80-19(P)(A) Volume 4 Revision 1, "Exxon Nuclear Methodology for Boiling Water Reactors: Application of the ENC Methodology to BWR Reloads," June 1986

10. EMF-2158(P)(A) Revision 0, "Siemens Power Corporation Methodology for Boiling Water Reactors: Evaluation and Validation of CASMO-4/MICROBURN-B2," October 1999

11. XN-NF-80-19(P)(A) Volume 3 Revision 2, "Exxon Nuclear Methodology for Boiling Water Reactors, THERMEX: Thermal Limits Methodology Summary Description," January 1987

12. XN-NF-84-105(P)(A) Volume 1 and Volume 1 Supplements 1 and 2, "XCOBRA-T: A Computer Code for BWR Transient Thermal-Hydraulic Core Analysis," February 1987

13. ANF-913(P)(A) Volume 1 Revision 1 and Volume 1 Supplements 2, 3, and 4, "COTRANSA2: A Computer Program for Boiling Water Reactor Transient Analyses," August 1990

14. EMF-2209(P)(A) Revision 3, "SPCB Critical Power Correlation," September 2009

15. EMF-2245(P)(A) Revision 0, "Application of Siemens Power Corporation's Critical Power Correlations to Co-Resident Fuel," August 2000

16. EMF-2361(P)(A) Revision 0, "EXEM BWR-2000 ECCS Evaluation Model," May 2001

17. EMF-2292(P)(A) Revision 0, "ATRIUM™-10: Appendix K Spray Heat Transfer Coefficients," September 2000

18. EMF-CC-074(P)(A) Volume 4 Revision 0, "BWR Stability Analysis: Assessment of STAIF with Input from MICROBURN-B2," August 2000

19. BAW-10247P-A Revision 0, "Realistic Thermal-Mechanical Fuel Rod Methodology for Boiling Water Reactors," February 2008

20. ANP-10298P-A Revision 1, "ACE/ATRIUM 10XM Critical Power Correlation," March 2014

5.6 Reporting Requirements

5.6.3 CORE OPERATING LIMITS REPORT (COLR) (continued)

21. ANP-10307P-A Revision 0, "AREVA MCPR Safety Limit Methodology for Boiling Water Reactors," June 2011

22. BAW-10255(P)(A) Revision 2, "Cycle-Specific DIVOM Methodology Using the RAMONA5-FA Code," AREVA NP Inc., May 2008

23. ANP-10262PA, Enhanced Option III Long Term Stability Solution, Revision 0, May 2008

24. BAW-10255(P)(A) Rev. 2, "Cycle-Specific DIVOM Methodology Using the RAMONA5-FA Code," AREVA NP Inc., May 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.

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 191

TO RENEWED FACILITY OPERATING LICENSE NO. DPR-22

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* (10 CFR) Section 2.390.~~
~~Proprietary information is identified by underlined text (in red font) enclosed within double brackets as shown here [[example proprietary text]].~~

TABLE OF CONTENTS

APPENDICES

1.0 INTRODUCTION

By letter dated October 3, 2014 (Reference 1), as supplemented by letters dated January 9 (Reference 2), August 26 (Reference 3), September 29 (Reference 4), and December 8, 2015 (Reference 5), and February 29 (Reference 6), April 29 (Reference 7), August 4 (Reference 8), September 14 (Reference 61), and September 28, 2016 (Reference 62), Northern States Power Company (NSPM) submitted a license amendment request (LAR) for Monticello Nuclear Generating Plant (MNGP). The proposed amendment would revise the Technical Specifications (TSs) and Renewed Facility Operating Licenses to allow operation in the extended flow window (EFW) domain.

The supplemental letters dated January 9, August 26, September 29, and December 8, 2015, and February 29, April 29, and August 4, 2016, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the U.S. Nuclear Regulatory Commission (NRC, the Commission) staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on July 7, 2015 (80 FR 38775).

1.1 Background

General Site Information

MNGP is a General Electric (GE)-designed boiling-water reactor (BWR), Type 3 (BWR/3). The core contains 484 fuel assemblies. This is a relatively small core compared to the remainder of the domestically licensed fleet of BWRs; most domestic BWRs have cores that contain greater than 700 fuel assemblies. In addition, the core power density and peak bundle power at MNGP remain below fleet averages.

Previous Relevant Licensing Activities

By Amendment No. 176, dated December 9, 2013 (Reference 9), the NRC approved implementation of an extended power uprate (EPU) which allows MNGP to operate at 2004 megawatts-thermal (MWt), which is the current licensed power level. Prior to the EPU, the licensed thermal power level was 1775 MWt. By Amendment No. 180 dated March 28, 2014, (Reference 10), the NRC approved implementation of the Maximum Extended Load Line Limit Analysis Plus (MELLLA+ or M+) operating-domain. Compared to MELLLA operating domain, MELLLA+ allows for plant operation at higher power-to-flow ratios which can produce a higher steam void content in the reactor coolant water within the core region. The NRC staff's evaluation and approval of the MELLLA+ LAR is limited to operation with a full core of Global Nuclear Fuels (GNF) GE14 fuel.

While MNGP has already been approved to operate in the MELLLA+ operating domain, which is parametrically the same as the EFW operating domain, the MELLLA+ licensed methods do not encompass AREVA fuel designs. To support its October 3, 2014, application to operate MNGP in the EFW expanded operating domain, the licensee submitted a collection of reports to support the LAR. Attachment 1 to the LAR discussed the reactor protection for EFW operation. ANP-3274NP (Reference 11) contains the analytical methods for anticipated transient without scram with instability (ATWSI) for MNGP. ANP-3284NP (Reference 12) contains the results of

analysis and benchmarking of methods for ATWSI for MNGP. ANP-3295NP (Reference 13) contains the MNGP licensing analysis for EFW (EPU/EFW). ANP-3135NP (Reference 14) contains the analysis of the applicability of AREVA BWR methods to EFW for MNGP.

By Amendment No. 188, dated June 5, 2015 (Reference 15), the NRC approved an amendment which allows a transition to the AREVA ATRIUM 10XM fuel design, and consists of changes to the operating license and TSs. The amendment revised the TSs to reflect the use of fuel and safety analysis methods appropriate for the AREVA ATRIUM 10XM fuel bundle design. Specifically, the changes affect TS 2.1, "Safety Limits," to revise the reactor steam dome pressure safety limit value; TS 4.2.1, "Fuel Assemblies," to more accurately reflect the fuel assembly design feature as a "water channel" as opposed to a "water rod"; and, TS 5.6.3, "Core Operating Limits Report (COLR)," to add AREVA safety analysis methods to the references list used in determining core operating limits in the COLR. This approval was only for MELLLA operation.

The licensing basis for EPU and M+ operation is based on NEDC-33322P (Reference 16), and NEDC-33435P (Reference 17). ANP-3295P (Reference 18) supports MNGP operation with AREVA ATRIUM 10XM fuel at EPU and EFW conditions.

In NEDC-33435P, 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 are applicable for EFW for all areas other than reactor systems. Reactor systems areas are evaluated in ANP-3295P, which is the safety analysis report (SAR) for EFW, and it is provided to supplement NEDC-33435P by addressing the effect of AREVA ATRIUM 10XM fuel on operation in the EFW domain. AREVA methods are evaluated in Appendix-A of this safety evaluation report (SER). The analyses in ANP-3295P support EFW operation of MNGP at the post-EPU, current licensed thermal power of 2004 MWt with core flow as low as 80 percent of rated.

The licensee proposed the following solutions in its LAR, as supplemented, to ensure an acceptable safety margin under EFW:

1. Operation in the EFW domain will continue to require having all safety relief valves (SRV) in 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.

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

3. Single-loop operation (SLO) is not permitted in the EFW domain.

4. The method for determining the linear heat generation rate (LHGR) letdown value and maximum average planar linear heat generation rate (MAPLHGR) values will not be changed.

5. The long term stability solution oscillation power range monitor (OPRM) amplitude setpoint is initially set to [[ ]]. If the licensee changes this amplitude setpoint then a

corresponding change will be made to the operating limit minimum critical power ratio (OLMCPR) margin based on the calculations documented in ANP-3295P.

6. There is no change in operator actions times. Operator actions to initiate reduction of reactor vessel water level have been assumed to occur within 90 seconds of the ATWS initiation.

7. Operation in the EFW region increases the core-average void fraction, and insufficient operating experience exists in this region. For this reason a 0.03 operating margin is necessary consistent with the previous MNGP MELLLA+ LAR.

The licensee's analyses results are summarized in Table 5.1 of ANP-3295P, and indicate that the limiting anticipated operational occurrences (AOOs) result in larger delta critical power ratio (ΔCPR) when initiated at nominal conditions than inside the EFW domain; therefore, additional OLPMCPR margin is not required for operation in the EFW domain. This result is typical for most reactors and is primarily a result of a shift in the axial power shape which makes scram faster and more effective. Based on the its review as described in detail below, the NRC staff concludes that the use of AREVA ATRIUM 10XM fuel and AREVA methods by MNGP in the EFW operating domain with the solutions proposed in the MNGP LAR, NEDC-33435P, and ANP-3295P are technically acceptable to satisfy the regulatory criteria.

2.0 REGULATORY EVALUATION

2.1 Title 10 of *The Code of Federal Regulations* (10 CFR) Requirements

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

1. 10 CFR 50.36, "Technical specifications," in which the Commission establishes its regulatory requirements related to the contents of the TSs. 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 [TSs] in accordance with the requirements of this section." 10 CFR 50.36(c)(3) 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."

2. 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.

3. 10 CFR 50.46, "Acceptance criteria for emergency core cooling systems [ECCS] for light-water nuclear power reactors," which establishes standards for the calculation of ECCS performance and acceptance criteria for that calculated performance.

4. 10 CFR 50.55a(h), requires that the protection systems meet Institute of Electrical and Electronics Engineers (IEEE) 279. Section 4.2 of IEEE 279-1971 discusses the general functional requirement.

5. 10 CFR 50.62, “Requirements for reduction of risk from anticipated transients without scram (ATWS) events for light-water-cooled nuclear power plants,” which requires licensees provide the means to address an ATWS event, an AOO 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.

6. 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.

7. 10 CFR 50.67, “Accident source term”, insofar as it requires that the applicable dose acceptance criteria are 5 roentgen equivalent man (rem) total effective dose equivalent (TEDE) in the control room (CR), 25 rem TEDE at the exclusion area boundary (EAB), and 25 rem TEDE at the outer boundary of the low-population zone (LPZ).

8. 10 CFR Part 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 loss-of-coolant accident (LOCA).

General Design Criteria

As the licensee described in Section 4.1 of its application, MNGP was not licensed to Appendix A, “General Design Criteria [GDC] for Nuclear Power Plants,” which were 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), as applicable.

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

1. GDC 1, “Quality standards and records,” insofar as it establishes that structures, systems, and components (SSCs) important to safety be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed.

2. GDC 4, “Environmental and dynamic effects design bases,” insofar as it establishes that SSCs important to safety be protected against dynamic effects associated with flow instabilities and loads such as those resulting from water hammer.

3. GDC 5, “Sharing of structures, systems, and components,” insofar as it establishes 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.

4. GDC 10, “Reactor design,” insofar as it establishes that the Reactor Protection System (RPS) 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.

5. GDC 11, “Reactor inherent protection,” insofar it establishes that as 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.

6. GDC 12, “Suppression of reactor power oscillations,” insofar as it establishes that unstable oscillations with the potential of violating specified acceptable fuel design limits (SAFDLs) either be impossible or readily detected and suppressed.

7. GDC 13, “Instrumentation and control,” insofar as it establishes 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.

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

9. GDC 16, “Containment design,” insofar as it establishes 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.

10. GDC 19, “Control room,” insofar as it establishes 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.

11. GDC 20, “Protection system functions,” insofar as it establishes that the RPS 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.

12. GDC 21, “Protection system reliability and testability,” insofar as it establishes that the system be designed for high functional reliability and in service testability, with redundancy and independence sufficient to preclude loss of the protection function from a single failure and preservation of minimum redundancy despite removal from service of any component or channel.

13. GDC 22, “Protection system independence,” insofar as it establishes that the system be designed so that natural phenomena, operating, maintenance, testing and postulated accident conditions do not result in loss of the protection function.

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

15. GDC 24, “Separation of protection and control systems,” insofar as it establishes that interconnection of the protection and control systems be limited to assure safety in case of failure or removal from service of common components.

16. GDC 25, “Protection system requirements for reactivity control malfunctions,” insofar as it establishes that the RPS be designed to assure that SAFDLs are not exceeded for any single malfunction of the reactivity control systems.

17. GDC 26, “Reactivity control system redundancy and capability,” insofar as it establishes that two independent reactivity control systems be provided, with both systems capable of reliably controlling the rate of reactivity changes so that SAFDLs are not exceeded.

18. GDC 28, “Reactivity limits,” insofar as it establishes 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.

19. GDC 29, “Protection against anticipated operational occurrences,” insofar as it establishes that protection and reactivity control systems be designed to assure an extremely high probability of accomplishing their safety functions in the event of AOOs.

20. GDC 31, “Fracture prevention of reactor coolant pressure boundary,” insofar as it establishes 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.

21. GDC 33, “Reactor coolant makeup,” insofar as it establishes that a system to provide reactor coolant makeup for protection against small breaks in the RCPB be provided and the system safety function be to assure that SAFDLs are not exceeded.

22. GDC 35, “Emergency core cooling,” insofar as it establishes that an emergency system to provide abundant emergency core cooling be provided to transfer heat from the reactor core following any LOCA.

23. GDC 38, “Containment Heat Removal,” insofar as it establishes that a Containment Heat Removal System (CHRS) be provided and that its function shall be to reduce rapidly the containment pressure and temperature following LOCA and maintain them at acceptably low levels.

24. GDC 41, “Containment atmosphere cleanup,” insofar as it establishes systems be provided 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.

25. GDC 50, “Containment Design Basis,” insofar as it establishes 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.

26. GDC 54, “Piping systems penetrating containment,” insofar as it establishes 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.

An evaluation comparing the MNGP design basis to the AEC-proposed GDCs is presented in the MNGP USAR, Appendix E, “Plant Comparative Evaluation with the Proposed AEC 70 Design Criteria. The following provides a comparison of the AEC-proposed GDC numbers with the 10 CFR Part 50, Appendix A, numbers:

1. GDC 1 is comparable to AEC-proposed GDCs 1 and 5.

2. GDC 4 is comparable to AEC-proposed GDCs 40 and 42.

3. GDC 5 is comparable to AEC-proposed GDC 4.

4. GDC 10 is comparable to AEC-proposed GDC 6, as further described in USAR, Section 14.4.

5. GDC 11 is comparable to AEC-proposed GDC 8.

6. GDC 12 is comparable to AEC-proposed GDC 7, as further described in USAR, Section 14.6.

7. GDC 13 is comparable to AEC-proposed GDC 12, as further described in USAR, Section 14.7.4.

8. GDC 15 is comparable to AEC-proposed GDC 9, as further described in USAR, Section 14.4.

9. GDC 16 is comparable to AEC-proposed GDC 49.

10. 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.

11. GDC 20 is comparable to AEC-proposed GDCs 14 and 15, as further described in USAR, Section 14.4.
12. GDC 21 is comparable to AEC-proposed GDCs 19, 20, and 25.

13. GDC 22 is comparable to AEC-proposed GDC 23.

14. GDC 23 is comparable to AEC-proposed GDC 26.

15. GDC 24 is comparable to AEC-proposed GDC 22.

16. GDC 25 is comparable to AEC-proposed GDC 31, as further described in USAR, Section 14.4.

17. GDC 26 is comparable to AEC-proposed GDC 27, as further described in USAR, Section 14.4.

18. GDC 28 is comparable to AEC-proposed GDC-32.

19. GDC 29 is comparable to AEC-proposed GDC-19.

20. GDC 31 is comparable to AEC-proposed GDCs 33, 34, and 35.

21. GDC 33 is comparable to AEC-proposed GDC-37.

22. GDC 35 is comparable to AEC-proposed GDCs 37, 42, and 44.

23. GDC 38 is comparable to AEC-proposed GDCs 41 and 52.

24. The intent of GDC-41 is described in USAR, Section 5.3.4.1.

25. GDC 50 is comparable to AEC-proposed GDC 49.

26. GDC 54 is comparable to AEC-proposed GDC-57.

2.2 <u>NRC Guidance Documents</u>

1. Review Standard RS-001, “Review Standard for Extended Power Uprates,” (Reference 20)

2. Standard Review Plan (SRP), specifically:

    a. SRP, Section 4, in particular: 4.2 “Fuel System Design,” 4.3 “Nuclear Design,” and 4.4 “Thermal and Hydraulic Design.”

    b. SRP, Section 5, in particular: 5.2.2 “Overpressure Protection,” 5.4.6 Reactor Core isolation Cooling System (BWR),” 5.4.7 Residual Heat Removal (RHR) System,”

    c. SRP, Section 6, in particular: 6.2.1.1.C “Pressure-Suppression Type BWR Containments,” 6.2.1.2 “Subcompartment Analysis,” 6.2.2 “Containment Heat Removal Systems,” 6.2.5 “Combustible Gas Control in Containment,” and 6.3 “Emergency Core Cooling System.”

d. SRP, Section 7, in particular Branch Technical Position (BTP) 7-19, Revision 6, "Guidance for Evaluation of Diversity and Defense-In-Depth in Digital Computer-Based Instrumentation and Control Systems."

e. SRP Section 9, in particular: 9.3.5 "Standby Liquid Control System (BWR)"

f. SRP Section 15, in particular: 15.0.1 "Radiological Consequence Analyses Using Alternative Source Terms," BWR-related sections in 15.2, 15.3, 15.4, 15.5, 15.6, and 15.7, 15.8 "Anticipated Transients without Scram," and 15.9 "Boiling Water Reactor Stability."

3. Draft Guide 1107, "Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident."

3.0 TECHNICAL EVALUATION

3.1 Overview of the License Amendment Request (LAR)

The Monticello Licensing Analysis for EFW (EPU/MELLLA+) ANP-3295P contains information divided into the following sections:

- Section 1.0 – Introduction
- Section 2.0 – Disposition of Events
- Section 3.0 – Mechanical Design Analyses
- Section 4.0 – Thermal-Hydraulic Design Analysis
- Section 5.0 – Anticipated Operational Occurrences
- Section 6.0 – Postulated Accidents
- Section 7.0 – Special Analyses
- Section 8.0 – Operating Limits and COLR Input

Summary of ANP-3295P, Revision 2, Sections 1.0 and 2.0

The licensing analyses provided in ANP-3295P support the "representative" core design documented in ANP-3213(P) (Reference 21). Although the first reload of AREVA fuel has subsequently been delayed until Cycle 29, Cycle 28 remains the representative first transition cycle for MNGP fuel transition and EFW LAR. The representative core design consists of a total of 484 fuel assemblies, including [[ ]] fresh AREVA ATRIUM 10XM assemblies and [[ ]] irradiated GE14 assemblies. The analyses are prepared to be the best representation of the proposed MNGP configuration (i.e., EPU at EFW). The Cycle 28 core design was used in this process as a representative design. The use of a representative core design is adequate for the purposes of the LAR, which are to: (1) demonstrate the core design meets the applicability requirements of the new analysis methods, (2) demonstrate that the results can meet the proposed safety limits, and (3) demonstrate either that existing TS limits do not need to be revised for the fuel transition or that the needed revisions are identified. The representative core design for these analyses assures the actual core design meets all these objectives.

Ultimately, the reload process will confirm the applicability of all plant inputs (including plant design changes made in the interim period) for all the appropriate safety analyses and will also perform the final confirmation that safety limits are satisfied for the actual core design that will

be loaded. These licensing analyses were performed for potentially limiting events and analyses identified in Section 2.0 of ANP-3295P. Results of analyses are used to establish the TS/COLR limits and ensure design and licensing criteria are met. Design and safety analyses are based on both operational assumptions and plant parameters provided by the license. The results of the licensing analysis support operation for the power/flow map presented in Figure 1 of this SER and also support operation with the equipment out-of-service (EOOS) scenarios presented in Table 1.1 of ANP-3295P.

The licensee reviewed all fuel-related design criteria, events, and analyses identified in the licensing basis. When operating limits are established to ensure acceptable consequences of an AOO or accident, the fuel-related aspects of the system design criteria are met. All fuel-related events were reviewed and dispositioned into one of the following categories:

1. No further analysis required.

This classification may result from one of the following:

A. The consequences of the event have been previously shown to be bounded by consequences of a different event, and the introduction of a new fuel design and transition to EFW conditions does not change that conclusion.

B. The consequences of the event are benign, i.e., the event causes no significant change in margins to the operating limits.

C. The event is not affected by the introduction of a new fuel design or transition to EFW conditions, and/or the current analysis of record remains applicable.

2. Address event each following reload.

The consequences of the event are potentially limiting and need to be addressed for each reload.

3. Address event for initial licensing analysis.

This classification may result from one of the following:

A. The analysis is performed using conservative bounding assumptions and inputs such that the initial licensing analysis results for EFW will remain applicable for following reloads of the same fuel design (AREVA ATRIUM 10XM).

B. Results from the initial licensing analysis will be used to quantitatively demonstrate that the results remain applicable for following reloads of the same fuel design because the consequences are benign or bounded by those of another event.

C. A disposition of events summary is presented in Table 2.1 of ANP-3295P. The disposition summary presents a list of the events and analyses, the corresponding USAR section, the disposition status of each event for transitioning to EFW conditions under AREVA methodologies, and any applicable comments.

Table 2-3 of the ANP-3295P lists all the computer codes used for this evaluation. Figure 1-1 of the ANP-3295P (reproduced below as Figure 1) defines the MELLLA+ operating domain. [[

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Summary of ANP-3295P, Revision 3 Section 3.0.

ANP-3158P (Reference 22), presents representative fuel rod thermal-mechanical analyses using RODEX4 methodology for Cycle 28, the transition cycle.

Summary of ANP-3295P, Revision 3, Section 4.0.

Following the approved methodology described in Appendix A of this SER, Section 4 of ANP-3295P, evaluates MNGP on a plant-specific basis for the following topics.

- 4.1 “Thermal-Hydraulic Design and Compatibility”
- 4.2 “Safety Limit MCPR Analysis”
- 4.3 “Core Hydrodynamic Stability”
- 4.4 “Voiding in the Channel Bypass Region”

Summary of ANP-3295P, Revision 3, Section 5.0.

Following the approved methodology described in Appendix-A of this SER, Section 5 of ANP-3295P, evaluates MNGP on a plant-specific basis for the AOOs.

- 5.1.1 “Load Rejection No Bypass (LRNB)”

- 5.1.2 “Turbine Trip No Bypass”
- 5.1.3 “Pneumatic System Degradation”
- 5.1.4 “Feedwater Controller Failure (FWCF)”
- 5.1.5 “Inadvertent HPCI [high pressure coolant injection] Start-Up”
- 5.1.6 “Loss of Feedwater Heating”
- 5.1.7 “Control Rod Withdrawal Error”
- 5.1.8 “Fast Flow Runup Analysis”
- 5.2.0 “Slow Flow Runup Analysis”
- 5.3.1 “Single Loop Operation”
- 5.3.2 “Pressure Regulator Failure Downscale (PRFDS)”
- 5.4.0 “Licensing Power Shape"

Summary of ANP-3295P, Revision 3, Section 6.0.

Following the approved methodology described in Appendix-A of this SER, Section 6 of ANP-3295P evaluates MNGP on a plant specific basis for the postulated accidents.

- 6.1 “Loss-of-Coolant Accident (LOCA)”
- 6.2 “Pump Seizure Accident”
- 6.3 “Control Rod Drop Accident (CRDA)”
- 6.4 “Fuel and Equipment Handling Accident”
- 6.5 “Fuel Loading Error (Infrequent Event)”

Summary of ANP-3295P, Revision 3, Section 7.0

Following the approved methodology described in Appendix-A of this SER, Section 7 of ANP-3295P, evaluates MNGP on a plant-specific basis for the Special Events.

- 7.1 “ASME Overpressurization Analysis”
- 7.2 “ATWS Event Evaluation”
- 7.3 “Reactor Core Safety Limits”
- 7.4 “Appendix-R Fire Protection Analysis”
- 7.5 “Standby Liquid Control System (SLCS)”
- 7.6 “Fuel Criticality”

Summary of ANP-3295P, Revision 3, Section 8.0.

Following the approved methodology described in Appendix-A of this SER, Section 8 of ANP-3295P, evaluates MNGP on a plant-specific basis for the operating limits and the COLR analyses.

- 8.1 “MCPR [minimum critical power ratio] Limits”
- 8.2 “LHGR Limits”
- 8.3 “MAPLHGR Limits”

3.2 Reactor Systems

3.2.1 Introduction

The NRC staff used RS-001, "Review Standard for Extended Power Uprates," as a reference in conducting the EFW review. Because EFW is not a power uprate, the staff recognizes that there are sections in RS-001 that are unnessary for the EFW application review and RS-001 guidance is not wholly applicable. Nevertheless, RS-001 provides a good framework for review of this application because operation in an extended operating domain presents similar operating consideration as a power uprate. 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

Fuel storage is unaffected by the EFW amendment and therefore is beyond the scope of the staff's review and will not be addressed further in this SE. The NRC staff's evaluation for the remainder of the above reactor systems is discussed in Section 3.2.

3.2.2 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 the following four objectives as specified in SRP 4.2 are met: (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 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 requires standards for the calculation of ECCS performance and acceptance criteria for calculated performance; (2) GDC 10, insofar as it establishes that the reactor core be designed with appropriate margin to assure that specified acceptable fuel design limits (SAFDLs) are not exceeded during any condition of normal operation, including the effects of AOOs; and (3) AEC proposed GDCs 37, 41, and 44, insofar as they establish that a system to provide abundant emergency core cooling be provided to prevent fuel damage following a 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 has reviewed the impact on the fuel system of the proposed EFW operating system domain based on the licensee-provided analyses results. Staff evaluation of these analyses and results are documented in this SE, Section 3.2.4.

The fuel system to be introduced in to MNGP for Cycle 29 operation is the AREVA ATRIUM 10XM fuel design. The fuel design is comprised of a 10x10 array of fuel rods with a square internal water channel that displaces a 3x3 array of rods, with 79 full-length rods (FLR), and 12 part length rods (PLFRs). The active length of a PLFR is approximately one-half the length of a FLR. Use of the PLFRs is expected to improve fuel utilization in the high void upper region of the fuel bundle, to enhance the shutdown margin, to improve stability, and to improve pressure drop performance. The AREVA ATRIUM 10XM fuel assembly consists of a lower tie plate (LTP), 91 fuel rods, nine spacer grids, a central water channel with [[
]], and miscellaneous assembly hardware. The structural connection between the LTP and upper tie plate (UTP) is provided by the central water channel.

Mechanical design details of the AREVA ATRIUM 10XM fuel was evaluated by the NRC staff and is summarized in Amendment No. 188 for fuel transition. This included fuel rods, fuel assembly and its components, and fuel channel. The four objectives provided in SRP Section 4.2 which are listed in the regulatory evaluation of this section, as well as the mechanical compatibility with the co-resident fuel, assure the structural integrity of the fuel and compatibility with the existing reload fuel (co-resident fuel). The American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) was used as guidance in establishing acceptable stress, deformation, and load limits for standard fuel assembly components.

Stresses under AOO and accident conditions were evaluated using a finite element analysis code. Post-irradiation examinations of AREVA fuel designs have confirmed that rod bow has not reduced spacing between adjacent rods. Rod growth, assembly growth, and fuel channel growth are calculated using correlations derived from post-irradiation data. NRC staff review of fuel design performance and structural design of the assembly and the fuel channel meet all mechanical compatibility and strength requirements for operation under EFW at MNGP.

The NRC staff reviewed the licensee's application of approved codes and methodologies in the fuel rod thermal-mechanical analyses for the AREVA ATRIUM 10XM fuel design that will be loaded and used for EFW operation at MNGP. The review was focused on fuel rod design, internal hydriding, cladding collapse, overheating of fuel pellets, channel bow, cladding stress and strain limits, fuel densification and swelling, fatigue, oxidation, hydriding and crud buildup and rod internal pressure. The staff determined that the fuel design criteria, as supported by the applicable regulations and sections of NUREG-0800, have been satisfied and provide reasonable assurance for safe operation at MNGP.

Fuel System Design Conclusion

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 staff concludes that the licensee has adequately accounted for the effects of the proposed operating domain extension on the fuel system and demonstrated that:

(1) the fuel system is not likely to be damaged as a result of normal operation and AOOs; (2) the fuel system damage, should it happen, is not likely to 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) coolability is likely to be maintained. Based on these findings, the NRC staff concludes that the fuel system and associated analyses will continue to meet the requirements of 10 CFR 50.46, and meets the intent of GDC 10, and AEC proposed GDCs 37, 41, and 44, following implementation of the proposed operating domain extension. Therefore, the NRC staff finds the proposed operating domain extension acceptable with respect to the fuel system design.

3.2.3 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 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 establishes 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 establishes 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 establishes 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 establishes 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 establishes 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 establishes 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 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 establishes 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

Analysis for the 'representative' core design is documented in ANP-3124 for Cycle 28 which was delayed until Cycle 29. This representative core design consists of a total of 484 fuel assemblies, including [[ ]] fresh AREVA ATRIUM 10XM and [[ ]] irradiated GE14 assemblies. The core design analysis has been performed using approved AREVA neutronics methodology as described in the fuel transition amendment. The CASMO-4 lattice depletion code was used to generate nuclear data including cross sections and local power peaking factors. The MICROBURN-B2 three dimensional core simulator code, combined with the application of the Siemens Power Corporation B (SPCB) critical power correlation, EMF-2209PA (Reference 23) for GE14 fuel and the AREVA critical power evaluator (ACE) critical power correlation ANP-10298PA (Reference 24), for the AREVA ATRIUM 10XM fuel, was used to model the core. Control rod patterns from Cycle 29 and the key operating parameters including thermal margins are shown in Appendix A of ANP-3215P. The cycle design calculations demonstrate adequate hot excess reactivity and cold shutdown margin throughout the cycle. ANP-3224P (Reference 25) provides additional details on the neutronics calculations that were performed for the nuclear design.

Cross Section Representation:

CASMO-4 performs a multi-group spectrum calculation using a detailed heterogeneous description of the fuel lattice components. Fuel rods, absorber rods, water rods/channels, and structural components are modeled explicitly. Depletion calculations are performed using predictor-corrector algorithm in each fuel or absorber rod. The two-dimensional transport solution based on [[ ]] provides pin power and exposure distributions, homogeneous multi-group microscopic cross-sections as well as macroscopic cross-sections. Discontinuity factors are determined from the solution.

MICROBURN-B2 performs microscopic fuel depletion on a nodal basis. The neutron diffusion equation is solved with a full two energy group method. This nodal method uses flux discontinuity factors for different regions and a multilevel iteration technique for efficiency. The model uses burnup gradient and spectral history gradient methods for accurate representation of in-reactor configuration. A full three-dimensional pin power reconstruction method is utilized. Traversing in-core probe (TIP) (neutron and gamma) and LPRM [local power range monitor] response models are included to compare calculated and measured instrument responses. Modern steady state thermal hydraulics models define the flow distribution among the assemblies. Models for the calculation of CPR, LHGR, and MAPLHR are included in the model for direct comparisons to the operating limits.

Microscopic and macroscopic cross-sections representation are from three void depletion calculations using CASMO-4. At any exposure point, a quadratic fit of the three CASMO-4 data points is used to represent the continuous cross-section over instantaneous variation of void or water density. Cross-section changes due to spectral changes during depletion have been included. Also, cross-section changes due to self-shielding that occurs with isotopic concentration change have been accounted for using void history and exposure. Quadratic interpolation methods have been employed to generate curves representing the behavior of the cross sections as a function of the historical void fraction during plant operation. The processed cross-sections for all isotopes in MICROBURN-B2 were compared to the cross-sections from

CASMO-4 calculations with continuous operation at all possible void fractions. The LAR reports that results show very good agreement for the entire exposure range of plant operation.

CASMO-4 uses an upper void fraction range of 80 percent as opposed to the traditional 70 percent, which introduces a slightly larger interpolation error for intermediate void conditions. However, Figure A-14 of ANP-3224P shows good accuracy for the 0 percent, 40 percent, and 80 percent methodology for the majority of assemblies and is considered appropriate for the extended power/flow operating domain (EPFOD) conditions, where void fractions are expected to be higher. MICROBURN-B2 uses water density rather than void fraction in order to account for pressure changes as well as subcooled density changes. Also MICROBURN-B2 uses spectral history rather than void history in order to account for other spectral influences due to actual core conditions (fuel loading, control rod inventory, leakage, etc.). Doppler feedback is modeled by accumulating Doppler broadening microscopic cross-sections of each nuclide using branch calculations performed with CASMO-4 at various exposures and void fractions for each void history depletion.

Uncertainty Calculations:

TIPs directly measure local neutron flux from the surrounding four fuel assemblies. The gamma scan data provides a means to determine a correlation between the TIP measurement and neutronic feedback that influences the power in the nearby assemblies. If a bundle is higher in power, neutronic feedback increases the power in the nearby assemblies. EMF-2158(P)(A) (Reference 26) data was reevaluated with deviations between measured and calculated TIP response at each axial level. The standard deviation of the error indicate that there is no significant trend versus axial position, which indicates no significant trend versus void fraction. Evaluation of core parameters such as core thermal power, core average void fraction, and the ratio between core power and core flow indicates that that there is no significant trend in the data associated with these plant parameters.

Comparison of core physics models to gamma scan results is done by converting pin power distribution to a Ba-140 density distribution. The Quad Cities Nuclear Power Station (QCNPS) assembly gamma scan data was used to determine the correlation coefficient which accounts for the correspondence between the assembly powers of adjacent assemblies. Quantification of this correspondence is achieved by a conservative multiplier to the TIP uncertainty. The accuracy of the MICROBURN-B2 model is demonstrated by comparison between measured and calculated TIP as well as comparison of calculated and measured La-140 activation. The accuracy of the MICROBURN-B2 models was further validated with detailed axial pin by pin gamma scan measurements of 9X9-1 and AREVA ATRIUM 10 fuel assemblies in the reactor designated as KWU-S.

Pin-by-pin gamma scan data is used for verification of the local peaking factor uncertainty. QCNPS measurements presented in the topical report (TR) EMF-2158(P)(A) have been reevaluated to determine any axial dependency. In order to determine axial dependency, full axial scans were performed on 16 fuel rods. Comparisons to calculated data show acceptable agreement at all axial levels. CASMO-4 and Monte Carlo N-Particle (MCNP) calculations have been performed to compare the fission rate distribution statistics. The fission rate differences at various void fractions demonstrate that CASMO-4 calculations have very similar uncertainties relative to the MCNP results for all void fractions. The NRC staff reviewed all the figures and Tables in ANP-3224P, and determined that the methodology is capable of accurately predicting

reactor conditions for fuel designs operated under current operating strategies and core conditions. The staff has determined that the neutronic and thermal hydraulic conditions predicted for the EPFOD operation are bounded by the data provided in the TR EMF-2158(P)(A) so the isotopic validation continues to be applicable to EPFOD operation.

Fuel Cycle Comparisons:

Fuel loading and control rod patterns are constrained by the MCPR limit that limits assembly power and exit void fraction regardless of the core power level. The LAR provided an evaluation of the void distribution by using the actual core designs used for each cycle with slightly different power distributions and reactivity characteristics than any other cycle. For all future EPFOD cycles, cycle-specific reload licensing calculations are performed using NRC-approved methodologies. The analysis presented in Section A.3 of ANP-3224P, indicates that EPFOD operation in the standard power/flow map is within the range of the original methodology approval for assembly power and exit void fraction.

Nuclear Design Conclusion

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 has: (1) adequately accounted for the effects of the proposed operating domain extension on the nuclear design and (2) demonstrated that the fuel design limits will not be exceeded during normal operation or AOOs 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 (see section 3.2.2 of this SE), thermal and hydraulic design (see section 3.2.4 of this SE), and transient and accident analyses (see section 3.2.6 of this SE), the staff concludes that the nuclear design of the fuel assemblies, control systems, and reactor core will continue to meet the intent of GDCs 10 and 12 and AEC proposed GDCs 6, 8, 12, 13, 14, 15, 27, 28, 29, 30, 31, and 32. Therefore, the NRC staff finds the proposed operating domain extension acceptable with respect to the nuclear design.

3.2.4 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 design-basis accident (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 establishes 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 establishes 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

Section 4 of ANP-3295P documents MNGP thermal hydraulic (TH) design analyses for MNGP, including the determination of the SLMCPR, stability, bypass voiding, and transition mixed core issues.

SLMCPR:

Following the approved methodology in ANP-10307PA (Reference 27), the SLMCPR is calculated as the minimum CPR value that guarantees that <0.1 percent of the fuel rods experience boiling transition. To confirm this criterion, a conservative power shape needs to be used. The radial power uncertainty used in the analysis includes the effects of up to 1 TIP machine out-of-service (TIPOOS) or the equivalent number of TIP channels and a LPRM calibration interval of 1000 MWd/ST average core exposure. The requirements associated with LPRM surveillance permit the frequency to be extended up to 25 percent of the specified frequency. This is included in the calculations through increased uncertainties for assembly radial peaking and nodal power.

ANP-3295P provides SLMCPR calculations at statepoints K, L, and M, of Figure 1, which bracket the operating flow conditions at high power. Even though the analysis does not follow the GE-Hitach Nuclear Energy (GEH) SAR, it also imposed the SLO flow uncertainties to points M & L to account for possible errors in the flow measurement at the higher void fraction conditions expected in the EFW region.

Rod peaking factors and associated uncertainties are calculated by MICROBURN-B2 using the methodology given in ANP-10307P. Table 4.3 of ANP-3295P documents the uncertainty values used for the analysis. The largest uncertainty contribution is related to power peaking factors [[ ]]. The main difference between two loop operation (TLO) and SLO uncertainties is related to the total core flow rate uncertainty (~2.5 percent for TLO, ~6.0 percent for SLO).

Table 4.4 of ANP-3295P documents a summary of the SLMCPR calculations. The conditions evaluated are points K, L, and M of Figure 1 (Note, SLO flow uncertainty applied to points L & M). In addition, an SLO condition is evaluated outside the EFW domain at 66 percent power – 52.5 percent flow (slightly to the left of point N). For all conditions evaluated, the number of rods in boiling transition is lower than 0.1 percent, with the limiting case being at point K (100 percent power – 105 percent flow) which is outside the EFW region.

Operation in the EFW region increases the core-average void fraction, and MNGP lacks sufficient operating experience in this region. For this reason, a 0.03 operating margin consistent with the previous MNGP MELLLA+ LAR is necessary.

The result of the MNGP analysis are [[ ]]. These values are acceptable because the calculation procedure uses approved methods without deviations,

including the application of SLO flow uncertainties at the higher void conditions inside the EFW domain.

CPR Correlations:

For the steady state and transient analyses, the AREVA ATRIUM 10XM fuel is analyzed and monitored with the ACE critical power correlation. The GE14 fuel is analyzed and monitored with the Siemans Power Corporation B (SPCB) critical power correlation, which has been reviewed and approved in EMF-2209PA. The SPCB additive constants and additive constant uncertainty for the GE14 fuel were developed using the indirect approach. To generate the SPCB parameters for GE14 fuel, AREVA provided an extended database of operating conditions, which used the approved GE14 GEXL CPR correlation to generate experimental data, which was then processed to generate the SPCB parameters.

For both, steady state and transient analyses, the internal structure of the codes requires a check of the CPR correlation to ensure that it is used within the accepted range of applicability. Thus, the staff concludes that both CPR correlations (ACE and SPCB) are guaranteed to be used within the acceptable range.

The NRC staff has reviewed the performance of AREVA ATRIUM 10XM fuel under off-nominal flow conditions. Some fuel designs with advanced spacers suffer a degradation of CPR performance when the flow is reduced (e.g., following a two-recirculation pump trip (2RPT) event). In the response to RAI 23 provided in ANP-3434(P) (Reference 28), the licensee provided the CPR performance of GE14 and ATRIUM 10XM following a 2RPT. Even though AREVA ATRIUM 10XM does not recover as much CPR margin as GE 14 (AREVA ATRIUM 10XM CPR flow-induced recovery is lower by ~15 percent than GE14 at natural circulation), the differences are minimal indicating that the AREVA ATRIUM 10XM CPR performance is not degraded significantly at off-nominal flows.

Void Fraction Correlations:

A key issue for operation in EFW is the increased power-to-flow ratio, which results in higher void fractions during steady state operation as well as during transients.

Appendix D of ANP-3224P contains an evaluation of void correlations using experimental void data from the KATHY (Karlstein hydraulic loop) facility, which uses a full-size AREVA ATRIUM 10XM electrically heated bundle to measure the in-channel void fraction using gamma densitometry. All the modern-fuel features of AREVA ATRIUM 10XM are represented in the KATHY facility, including part-length rods and expansion/contractions caused by water rods.

The analysis methods use two correlations:

- [[ ]], which forms the basis for the drift flux formalism used in nuclear design, frequency domain stability, nuclear AOO transient and accident analysis.

- Ohkawa-Lahey, which is used in TH design, system AOO transient and accident analysis, and loss of coolant accidents.

For the AREVA ATRIUM 10XM experiments at the KATHY facility, void fraction measurements were performed at three axial locations, including in the part-length rod region as well as near the bundle exit where voids are largest. Figure 2 (Fig 2-1 of ANP-3224P) shows graphically the KATHY operating conditions where void (triangles) and pressure drop (squares) measurements were performed compared to the expected MNGP operating conditions in the EFW region and experience in the operating fleet. The KATHY experimental data bounds the expected conditions in MNGP under EFW. Table D-1 of ANP-3224P presents the complete AREVA void fraction validation database, which includes FRIGG-2, FRIGG-3, and KATHY tests with two different axial power profiles for a total of ~400 experimental conditions. Void fractions as high as [[ ]] are included in the validation database.

Figure 3 and Figure 4 (Figures D-2 and D-4 of ANP-3224P) presents the validation of the [[ ]] and Ohkawa-Lahey correlations against the available experimental data. The predicted versus measured comparison does not show significant trends, indicating that the correlations accurately predict the behavior in the ATRIUM 10XM bundle. The absolute error bands are lower than 5 percent.

Bounding simulation analyses were conducted where the void correlation was biased beyond the experimental spread (see Fig D-7 of ANP-3224P) and used that biased correlation to calculate SLMCPR and OLMCPR. The results (documented in in Table D-2 of ANP-3224P) show very little sensitivity to this bounding void correlation biases. The SLMCPR varies by ±0.002 and the OLMCPR varies by ±0.005, which is insignificant.

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In summary, the NRC staff has reviewed validation data for the proposed void fraction correlation methods. A large number of experimental data points are available that represent the exact geometry of the AREVA ATRIUM 10XM fuel and cover all the expected range of operation in MNGP in EFW. The proposed methods benchmark well against the experimental data. In addition, the possible impact of bounding errors in the void correlation methods were evaluated and concluded that at most the impact is ±0.002 in SLMCPR and ±0.005 in OLMCPR,

which is insignificant. Based on this evaluation, the staff concludes that the use of the proposed void fraction methods is acceptable for MNGP EFW application.

Long Term Stability Solution EO-III Implementation:

MNGP has implemented, as part of the long term stability solution, enhanced option III (EO-III) to support EFW operation. Reload validation has been performed in accordance with ANP-10262(P)(A). Important features of EO-III include:

1. the channel instability exclusion region,

2. the required OLMCPR margin as function of OPRM setpoints, and

3. the backup stability exclusion region.

As part of the MNGP EFW LAR, MNGP has performed analysis, documented in ANP-3295P, to evaluate the impact of transitioning to AREVA fuels and methods while continuing operation in the EFW, which is parametrically the same as MELLLA+. The conclusion of these analyses is that the change required to MNGP TSs is related to the implementation of long term stability solution EO-III, which would replace the currently approved detect and suppress solution – confirmation density (DSS-CD). The other proposed change to the TSs is the addition of AREVA methods AISHA and SINANO for calculation of the ATWSI event, which is a required calculation upon fuel transition.

The implementation of the EO-III solution is essentially identical to the standard Solution III with the following modifications:

- The introduction of a calculated exclusion region on the power-flow map to preclude single channel instabilities. The exclusion region is enforced automatically.

- The introduction of a Hot Channel Oscillation Magnitude (HCOM) penalty to account for the impact of higher growth ratios expected in expanded operating domains such as EFW as compared to the original Solution III application.

The HCOM penalty has two implementation options:

1. A generic 0.005 reduction of the amplitude setpoint calculated from pre-EFW HCOM analysis is applied, or

2. A detailed, cycle-specific calculation HCOM analysis is performed to account for the increased probability of higher oscillation growth ratios.

The licensing basis for EO-III remains unchanged from the approved Solution III. When the period based detection algorithm (PBDA) detects an oscillation in the OPRM of amplitude greater than the setpoint, the reactor scrams. The exclusion region enforces the requirement that the individual channel thermal-hydraulics be stable, thus guaranteeing the well-behaved structure of the Delta Initial MCPR Versus Oscillation Magnitude (DIVOM) correlation.

The generic 0.005 penalty on the setpoint is a small correction that was imposed on the EO-III SER to account for the possibility of faster growth ratios in extended operating regions (e.g., EFW) than was originally accounted for in the approval of Solution III licensing analyses. The generic value of 0.005 was determined from a series of RAMONA5-FA and XCT-MODES calculations at representative conditions. The EO-III solution allows for a detailed cycle-specific HCOM calculation that explicitly accounts for the possibility of faster growth ratios when the initiating condition is inside the EFW domain.

The PBDA amplitude setpoint calculation in solution EO-III follows the same procedure as the standard Solution III. This methodology is defined in NEDO-31960-A, Supplement 1 (Reference 29), and NEDO-32465-A (Reference 30), and remains unchanged by this solution.

EO-III Implementation in MNGP:

MNGP takes no deviations from the approved EO-III methodology described in ANP-10262(P)(A) (Reference 31), which replaces the currently licensed DSS-CD method described in NEDC-33075P (Reference 32).

To implement EO-III and disable DSS-CD, MNGP will install a jumper in the Power Range Monitor (PRM) system to disable the confirmation density algorithm (CDA) trip, so that the PBDA will become the licensing basis for stability protection. In addition, the flow-biased scram function setpoints will be modified to implement the channel instability exclusion region (CIER). EO-III is an approved solution for use in the EFW domain, which is equivalent to the MELLLA+ domain. Therefore, the use of EO-III is acceptable.

EFWS (Channel Instability) Exclusion Region in MNGP:

The EFW stability (EFWS) region is a change of nomenclature in the MNGP EO-III implementation. The EFWS region is referred to as the stability protection trip (SPT) region in the approved EO-III licensing topical report (LTR). The primary reason for the change in nomenclature was to avoid possible confusions with the existing simulated thermal trip (STP) scram. The EFWS Trip is ensured by hardware/firmware that is part of the Nuclear Measurement Analysis and Control (NUMAC) power range neutron monitoring system (PRNMS) installed in MNGP in 2009 as part of the EPU.

The EFWS trip serves two functions:

1. During normal operation in the EFW region, the EFWS trip ensures that an automated scram will occur before any individual TH channel can become unstable, thus guaranteeing that the EO-III DIVOM curve is well-behaved and conservative.

2. In case the PBDA algorithm scram is declared inoperable, the EFWS serves as a backup stability solution, which defines an exclusion region with an automated scram if entered.

MNGP has provided sample calculations for the CIER and EFWS regions. Because of the low power density and relatively large inlet orifice pressure drop, MNGP is a relatively stable plant and the CIER lies completely outside the operating map, as shown in Figure 5. Note that this

calculation is performed on a cycle-specific basis and this region may change in future cycles, but it is not likely to vary significantly.

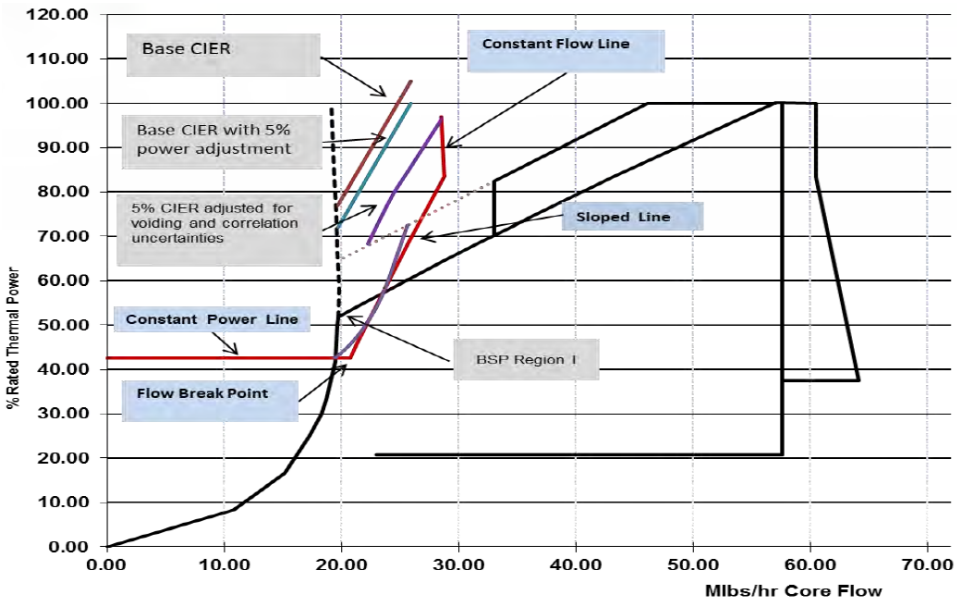

Figure 5 – MNGP CIER and EFWS regions

Backup Stability Solution:

The EO-III SER for ANP-10262(P)(A) did not specifically endorse a backup stability solution; instead, it required a plant specific review of the proposed implementation. MNGP has proposed to use a single exclusion region (the EFWS) to enforce both the CIER when the PBDA algorithm is enabled, and the backup stability protection (BSP) regions. For this reason, only one EFWS region is defined, and it corresponds to the red line in Figure 5. It is defined with three straight lines: constant power, constant drive flow lines and a straight line connecting them. The red line is defined to encompass the entire calculated BSP region.

The BSP region is calculated using the Boiling Water Reactor Owners Group (BWROG) procedures (Reference 29), to ensure that out-of-phase (or regional) core-wide and channel instabilities are extremely unlikely outside this region. In addition, the BWROG procedures specify that the BSP cannot be smaller than the original interim corrective actions (ICAs), which are defined based on the 80 percent rod line intercept with the natural circulation line. In the case of MNGP the calculated BSP is smaller than the ICA exclusion region, so the BSP is increased to include the 80 percent rod line intercept.

The EO-III SER imposes a condition that the EFWS region be armed (i.e., that the trip be active) before the EFW region is entered. MNGP proposes to implement this condition by including in the Technical Requirements Manual (TRM) that the EFWS region needs to be armed when operating above 70 percent power. The 70 percent power threshold for arming the EFWS is intended mostly for startup procedures.

TLCO 3.3.6.1 was added to the Technical Requirements Manual to define the EFWS High Trip power setpoints used in the AREVA EO-III long-term solution (LTS). The EO-III methodology

relies on EWFS protection in combination with the OPRMs to provide long term stability protection.

TLCO 3.3.6.1 establishes power levels at which the EFWS will be enabled depending on the operability of OPRM channels along with operability requirements. When no more than one OPRM channel is inoperable, the power level for enabling EFWS is set to 70 percent rated thermal power (RTP). This limit was set to ensure adequate CIER protection in the event of an AOO. When two or more OPRM channels are inoperable, the power level for enabling EFWS is set to the RTP value at which Region I intersect the natural circulation line (NCL), historically ~43 percent RTP. This limit was set to conservatively preclude operation in Region I in the event of core flow reduction from the EFW domain.

In addition, the EO-III SER imposes a condition that the EFWS region must protect the natural circulation line in case of a 2RPT when the PBDA OPRM scram has been declared inoperable. The intent of the EO-III SER condition is that an automated scram should occur if a 2RPT occurs from an initial condition above the rod line where the CIER region intercepts the natural circulation line. For conditions below this rod line, the natural circulation line does not need protection because single-channel instabilities are extremely unlikely. Given the high relative stability of MNGP, operation above 70 percent power always guarantees that, should a 2RPT occur, the CIER region will not be entered.

The TRM provision that the EFWS region be armed above 70 percent power (with plant procedures requiring it armed above 50 percent) satisfies both EO-III SER conditions. With this provision, the EFWS region will be armed before entering the EFW region, and it will protect against instabilities in the natural circulation line with an automated scram should a 2RPT occur. In the response to RAI 31 (Reference 28), the licensee stated that the BSP regions calculated using the EO-III methodology are equal to or more conservative than if it had been calculated with the standard Option III. The basis of this position is that the BSP is the line bounding the region where the decay ratio (DR) > (1.0 - uncertainties) and the methodology does not change the position of that line. Thus, the BSP regions calculated using the EO-III methodology may be used for Option III implementation before the EFWS (channel instability region) region is armed outside the EFW region.

Based on the NRC staff's evaluation of the response, the staff concludes that the backup stability solution proposed by MNGP is acceptable because it satisfies all the requirements from the EO-III SER.

Generic 0.005 HCOM Penalty:

The response to RAI 15 (Reference 28) and report ANP-10262Q1P (Reference 33), address the impact of the normally occurring reactor noise in the response time of EO-III. The response and report explain that because [[

]]

Following a 2RPT from the EFW domain, the potential exists for fairly large ultimate DR values; however, ANP-10262Q1P states that the [[

]] because the [[
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Figure 4.1b of ANP-10262Q1P (shown here as Figure 6) shows an instability calculation with noise. Figure 7 shows a simulation of the PBDA algorithm and the scram time, which occurs with DR only slightly larger than 1.0. ANP-10262Q1P also shows RAMONA calculations of instability in EFW. Figs 1.1 through 1.10 of ANP-10262Q1P (not reproduced here) show transient calculations with noise included. The staff has reviewed these figures and [[
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During the review of EO-III, the staff imposed a 0.005 penalty on HCOM to account for the possibility of fairly high DRs at the moment of scram. During ACRS review of option EO-III, the members recommended in the associated ACRS letter that the 0.005 penalty be reviewed for adequacy. Upon further review of these data, the discussion provided by ANP-10262Q1P including the supporting calculations with noise included support the conclusion that the EO-III OPRM scram is likely to occur when [[

]] Thus, it is concluded that the 0.005 HCOM penalty was a conservative limitation imposed by the staff in the absence of detailed calculation at the time, and it is no longer required.
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Thermal and Hydraulic Design Conclusion

The NRC staff has reached the following conclusions based on its review of MNGP's implementation of EO-III for operation in the EFW region:

1. MNGP's EO-III implementation takes no deviations from the approved SER and is, thus, an acceptable long term stability solution for operation in the EFW domain.

2. Implementing a single EFWS exclusion region satisfies both the CIER and BSP functions of EO-III. It is an acceptable backup stability implementation.

3. Arming the EFWS exclusion region above 70 percent power is an acceptable implementation of the "protect the natural circulation line" SER requirement because, should a 2RPT event happen below 70 percent power, the operating conditions will be lower than the CIER intercept with the natural circulation line where single-channel instabilities are very unlikely.

4. Simulations with noise included demonstrate that the EO-III OPRM scram is very likely to happen [[ ]] Therefore, the 0.005 HCOM penalty is no longer required.

Based on this evaluation, the staff concludes that MNGP's EO-III implementation takes no deviations from the approved SER and is, thus, an acceptable long term stability solution for operation in the EFW domain.

Bypass Voiding:

Bypass voiding is a concern because it has the potential of de-calibrating the LPRM detectors. A limit of 5 percent bypass voiding is typically enforced to avoid de-calibration. The proposed methods (all steady state, transient, and stability codes) explicitly include a check for all calculations of the bypass and the bypass voiding. The maximum bypass voiding calculated at the D-level LPRM detector in MNGP is [[ ]] and it occurs at point M. Since this number is less than the criterion of 5 percent bypass voiding is not at issue for MNGP.

Compatibility of Fuels and Mixed-Core Issues:

MNGP is planning to transition from GE14 to AREVA ATRIUM10XM fuel. Mixed cores are always a concern, especially if the two fuel types are not TH compatible and one of the fuel types has significantly lower flow resistance, which would result in flow starvation of the other fuel type (i.e., most flow would be diverted to the lower resistance bundle). The NRC staff has performed a series of confirmatory calculations to verify independently that AREVA ATRIUM 10XM and GE14 are TH compatible.

The results of the NRC staff calculations indicate that GE14 and AREVA ATRIUM 10XM are TH compatible. The average flow deviation between the two fuel types for a wide variety of power and flow conditions is 1.6 percent. The highest deviation observed was 2.5 percent. AREVA ATRIUM 10XM has a slightly higher (~1.6 percent) flow than GE14 at the same conditions.

ANP-3295P presents an evaluation of the impact of the transition mixed cores. The results are summarized in Table 4.2 of that report. Table 1 of this SE provides a summary. The last column presents the minimum CPR during steady state operation along the cycle depletion for the transition cores. The NRC staff observes that:

1. Once a full load of AREVA ATRIUM 10XM is achieved, the MCPR increases, providing additional margin to limits.

2. The worst transition cycle is expected to be the second transition. An MCPR reduction of 1.2 percent is expected on twice burned GE14 fuel. This reduction is insignificant and does not pose a safety concern.

Table 1 – CPR calculations for transition cores

		CPR
Full GE14 core	GE14	1.470
First Transition Core	GE14	1.461
	AREVA ATRIUM 10XM	1.659
Second Transition Core	GE14	1.451
	AREVA ATRIUM 10XM	1.650
Full AREVA ATRIUM 10XM Core	AREVA ATRIUM 10XM	1.645

Based on this evaluation, the staff concludes that GE14 and AREVA ATRIUM 10XM fuels are sufficiently TH compatible that mixed core issues are minimized. Therefore, the fuel transition is acceptable from the CPR impact point of view.

Conclusions:

The NRC staff has reached the following conclusions based on this evaluation:

1. The proposed SLMCPR values for MNGP Cycle 29 ([[ ]]) are acceptable because the calculation procedure uses approved methods without deviations, including the application of SLO flow uncertainties at the higher void conditions inside the EFW domain.

2. The use of the ACE CPR correlation for AREVA ATRIUM10XM and SPCB for GE14 is acceptable because these two correlations have been approved in the past and no deviations have been noted.

3. The range of validity is checked for each calculation; therefore, both CPR correlations (ACE and SPCB) are guaranteed to be used within the acceptable range.

4. MNGP's EO-III implementation takes no deviations from the approved SER and is, thus, an acceptable long term stability solution for operation in the EFW domain.

5. Bypass voiding has been calculated for MNGP EFW conditions and is lower than the 5 percent criterion, which is acceptable.

6. GE14 and AREVA ATRIUM 10XM fuels are sufficiently TH compatible that mixed core issues are minimized.

3.2.5 Emergency Systems

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

1. Control rod drive system

2. Overpressure protection for the RCPB during power operation

3. Reactor core isolation cooling

4. Reactor heat removal system

5. Standby liquid control system

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 effect 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 establishes 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 establishes 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 establishes 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 establishes 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 alternative rod injection system have redundant scram air header exhaust valves. Specific review criteria are contained in SRP, Section 4.6.

Control Rod Drive System Evaluation and Conclusion:

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

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 establishes 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 establishes 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.

Overpressure Protection for the RCPB during Power Operation Evaluation and Conclusion:

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.

Section 7 of ANP-3295P documents analyses performed to demonstrate that the MNGP acceptance criteria (dome pressure < 1332 psig and vessel pressure < 1375 psig) are satisfied for pressurization events (main steam isolation valve (MSIV) closure (MSIVC), TSV closure, and TCV closure). AREVA's plant simulator COTRANSA2 was used for these analyses.

MSIV closure is the most limiting pressurization event in MNGP and was analyzed at 102 percent power and both 80 percent and 105 percent flow. A maximum pressure of 1361 psig results in the vessel bottom, which is below the acceptable limits. Even though MNGP operates with no more than a single safety relief valve out-of-service (SRVOOS) in EFW because of ATWS over-pressure limitations, the non-ATWS analyses assumed 3 SRVOOS, a 5 percent drift over the TS SRV opening setpoint, and a fast 2.2 MSIV closure time. The maximum overpressure occurs at 105 percent flow conditions, indicating that EFW operation has no significant impact.

Reactor Core Isolation Cooling (RCIC)

Regulatory Evaluation:

The 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 EFW 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 establishes that SSCs important to safety be protected against dynamic effects; (2) GDC 5, insofar as it establishes 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 establishes 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 establishes 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.

RCIC Evaluation and Conclusion

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 10 CFR 50.63 and the intent of GDCs 4, 5, 33, and 54 continue to be satisfied.

Residual Heat Removal

Regulatory Evaluation:

The 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 EFW 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 establishes that SSCs important to safety be protected against dynamic effects; and (2) GDC 5, insofar as it establishes 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.

RHR Evaluation and Conclusion

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

Standby Liquid Control (SLC)

Regulatory Evaluation:

The SLC system provides backup capability for reactivity control independent of the control rod system. The SLC system functions by injecting a boron solution into the reactor to affect shutdown. The NRC staff's review covered the effect of the proposed EFW 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 establishes 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 SLC system 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.

Evaluation:

The hot shutdown boron weight (HSBW) is calculated on a generic bases for each fuel line (e.g., ATRIUM 10XM in the case of MNGP). The HSBW is confirmed effective on plant- and cycle-specific basis. Section 7.5 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 EFW initial conditions. Therefore, no modification to the SLC system design is required for EFW.

The SLC system design has not been modified relative to the baseline, the reactor pressure has not been modified and the SLC system boron inventory shutdown margin has been evaluated for the initial core in the SAR. Therefore, the NRC staff finds that sufficient information has been provided to review the SLC system, and the requirements of 10 CFR 50.62(c)(4), as well as the intent of GDC 26, continue to be satisfied.

SLC Conclusion:

The NRC staff has reviewed the licensee's analyses related to the effects of the proposed operating domain extension on the SLC system and concludes that the design has not been modified relative to the baseline, the reactor pressure has not been modified and the SLC system boron inventory shutdown margin has been evaluated for the initial core; 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 staff concludes that the SLC system will continue to meet the requirements of 10 CFR 50.62(c)(4) and the intent of GDC 26 following implementation of the proposed operating domain extension. Therefore, the NRC staff finds the proposed operating domain extension acceptable with respect to the SLC system.

3.2.6 Accident and Transient Analyses

Anticipated Operational Occurrences (AOOs)

Section 5 of ANP-3295P documents MNGP analysis of AOOs using AREVA methods under EFW conditions. The results of these analyses are used to determine the power- and flow-dependent MCPR operating limits and power- and flow-dependent LHGR multipliers.

AREVA AOO Methods:

COTRANSA2, ANF-913PA (Reference 34), XCOBRA, XN-NF-80-19PA (Reference 35), XCOBRA-T, XN-NF-84-105PA (Reference 36) and CASMO-4/MICROBURN-B2, EMF-2158PA are the major codes used in the thermal limits analyses as described in the AREVA THERMEX methodology report, XN-NF-80-19PA and neutronics methodology report. Fuel pellet-to-cladding gap conductance values are based on RODEX2, XN-NF-81-58PA (Reference 37), calculations for the MNGP Cycle 28 representative core.

COTRANSA2 is AREVA's system transient simulation code, which includes an axial one-dimensional neutronics model that captures the effects of axial power shifts associated with the system transients. XCOBRA-T is a transient thermal-hydraulics code used in the analysis of

thermal margins for the limiting fuel assembly. XCOBRA is used in steady-state analyses. The previous codes use RODEX2 calculations for gap conductance as function of exposure. These methods have been approved for use up to the MELLLA boundary at the original licensed thermal power, and they were reviewed by the NRC staff for applicability to EPU conditions. The applicability of these methods to EFW conditions has been evaluated in Appendix-A of this report.

Analyzed AOO Events:

The AOO assumptions are described in detail in ANP-3295P. The analyzed events are:

1. Load Rejection No Bypass (LRNB)
2. Turbine Trip No Bypass (TTNB)
3. Pneumatic System Degradation - Turbine Trip With Bypass and Degraded Scram (TTWB)
4. Feedwater Controller Failure (FW CF)
5. Inadvertent HPCI Start-Up (HPCI)
6. Loss of Feedwater Heating
7. Control Rod Withdrawal Error
8. Fast Flow Runup Analysis

In addition to the above system transients, the slow flow runup analysis was performed to determine flow-dependent MCPR limits and LHGR multipliers to ensure that the MCPR limit is not violated during an uncontrolled flow increase.

The calculated AOO ΔCPR values are shown in Table 2 for the equilibrium AREVA ATRIUM 10XM fuel. As expected, the limiting AOO event occurs at the maximum flow (105 percent) and is, therefore, not directly affected by EFW operation. These results are similar to the current MNGP results with GE14 fuel in EFW.

Table 2 – AOO ΔCPR FOR AREVA ATRIUM 10XM

	ΔCPR	
Event	80 percent Flow	105 percent Flow
LRNB	0.30	0.36
TTNB	0.33	0.41
TTWBP	0.32	0.38
FWCF	0.35	0.43
HPCI	0.38	0.47
LOFWH	0.16	0.17

Even though SLO is not permitted under EFW conditions, AOOs were evaluated under SLO outside the EFW region. The analyses concluded that the limiting TLO AOOs are not more severe when initiated from SLO conditions and the same ΔCPR and LHGR multipliers apply.

The pressure regulator out of service (PROOS) scenario results in a more severe accident if the backup regulator is not available to control the pressure when the remaining regulator fails, which would then result in a scram either because of high pressure or high power. The results of this analysis shows that this failure is similar to the Turbine Trip No Bypass AOO, and both result in a ΔCPR of 0.41, which is not the limiting AOO (limiting AOO is inadvertent HPCI startup, which results in ΔCPR = 0.47).

In addition to SLO and PROOS, ANP-3295P analysis support MNGP operation with three SRVs out-of-service, up to 1 TIPOOS (or the equivalent number of TIP channels) and a LPRM calibration interval of 1000 MWd/ST average core exposure. The requirements associated with LPRM surveillance permit the frequency to be extended up to 25 percent of the specified frequency.

The NRC staff has reviewed the AOO analyses for MNGP at EFW conditions using AREVA methods. For all AOOs analyzed, the limiting condition occurs at the 105 percent flow condition, which is outside the EFW domain. Thus, the staff concludes that EFW operation with AREVA ATRIUM 10XM fuel does not affect directly the transient ΔCPR of LHGR limits. Therefore, the staff finds the AOO analyses for MNGP at EFW conditions using AREVA methods acceptable.

Operating Limits

Section 8.0 of ANP-3295P documents an implementation of AREVA's methodology to MNGP to determine the operating limits. MCPR and LHGR are calculated using standard approved methods. Table values are provided for all the conditions required for input to the COLR.

Currently, MNGP has three sets of penalties to operate in the MELLLA+ region with GEH methods:

1. 0.01 OLMCPR penalty due to lack of experimental data supporting the [[ ]] void-quality correlation to demonstrate its accuracy, especially at high void fractions.

2. 0.03 SLMCPR penalty for power densities greater than 42MWt/Mlbm/hr because of lack of experimental data supporting the accuracy of power distributions, especially at high void fractions.

3. A requirement to use the SLO uncertainties when calculating the SLMCPR inside the MELLLA+ domain.

MNGP has presented additional information in the EFW submittal and requests that these penalties be removed.

OLMCPR 0.01 Penalty:

The 0.01 OLMCPR penalty was imposed by the NRC staff to the application of GEH methods due to lack of experimental data supporting the [[ ]] void-quality correlation to demonstrate its accuracy, especially at high void fractions.

The LAR presented void fraction measurements that were performed to specifically assess the impact of the AREVA ATRIUM 10XM fuel design attributes on void fraction predictions by AREVA methods. These were performed at the KATHY test facility using two prototypical BWR test assemblies (including AREVA ATRIUM 10XM) with part-length rods and mixing vane spacer grids. In addition, AREVA has used reference void fraction data from FRIGG-2 and FRIGG-3. These data are summarized in ANP-3224.

AREVA shows an excellent agreement between the experimental data and the steady-state core simulator void fraction predictions, which uses the [[ ]] correlation. The agreement for the transient methods, which use the Ohkawa-Lahey correlation, is not as accurate as the [[ ]] results, and it shows a small negative bias (~-0.05) for void fractions between 40 percent and 80 percent, with unbiased agreement for $\alpha<40$ percent and $\alpha>80$ percent. See Figure 3 and Figure 4.

In addition to the new experimental data, Appendix D of ANP-3224 provides a sensitivity study where the void-quality correlation is biased up and down (±0.05 void) based on the scatter observed between calculation and measurements. The sensitivity results indicate that biasing the void-quality correlations results in a very small increase in ΔCPR, a very small decrease in SLMCPR, and a very small increase in OLMCPR for this study; therefore, the impact of the correlation bias is small (see Table D-2 of ANP-3224).

The NRC staff has reviewed the above additional information and concludes that the void fraction benchmarks against prototypical BWR assemblies operated at high void fractions along with sensitivity analysis provide sufficient basis for the removal of the existing 0.01 OLMCPR limit in MNGP.

SLMCPR 0.03 Penalty:

Gamma Scan data for modern fuel designs was presented in the 1999 CASMO-4/MICROBURN-B2 topical report, EMF-2158, Section 8. In addition, in the response to RAI 10, the licensee presents the results of more recent gamma scan measurements in 48 assemblies, including AREVA ATRIUM 10XM, with good agreement for the high-power fuel rods.

In the response to RAI 10 (Reference 28), the licensee provided Figure 8 which shows the TIP statistical uncertainty for the EPU plants in their fleet. The red points are TIP uncertainties pre-EPU, and the green points represent the uncertainty post-EPU. This figure shows no discernible trend pre- and post-EPU. The figure also confirms that a few TIP measurements were collected up to a power-to-flow ratio value of 52 MWth/Mlb/hr, which exceeds the operating conditions in MNGP (<50 MWth/Mlb/hr) and no discernible trend can be observed pre- and post-EPU on the power distribution uncertainty.

The NRC staff notes that an EPU power uprate does not increase the maximum allowed power-to-flow density because it simply follows the MELLLA line. However, operation pre-EPU occurs

at 100 percent original licensed thermal power (OLTP) with variable flow, while post EPU, the operation occurs at the maximum power-to-flow density because the flow is fixed at ~100 percent. Thus, operation at EPU conditions reflects a higher average void fraction than pre-EPU for most of the cycle and, therefore, it provides information about the accuracy of AREVA methods at higher void fractions.
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This information, along with additional TIP uncertainty provided in the response to RAI 1, has been used by the NRC staff in its evaluation of the applicability of the 0.03 SLMCP penalty. Gamma scan data has been provided for the specific AREVA ATRIUM 10XM fuel, and limited TIP data for power-to-flow ratios greater than 50 MWth-hr/Mlb has been evaluated and show no significant error trend with increased power-to-flow ratio. While the 0.03 SLMCPR penalty may not be needed for MNGP using AREVA methods in the EFW region, no data has been collected at MNGP at the lower corner of the EFW region. Therefore, the staff concludes that it is prudent to maintain the 0.03 operating margin while operating experience and plant data is collected for first time operation in the complete EFW region.

The NRC staff concludes that the information provided by MNGP does not sufficiently justify the removal of the 0.03 SLMCPR penalty and, therefore, an additional 0.03 operating margin is necessary.

Use of SLO Uncertainties:

The MELLLA+ SER in NEDC-33006PA (Reference 38) states “The higher CF [core flow] and FWF uncertainties must be applied to the non-rated conditions, and should be high enough to compensate for the difficulties associated with benchmarking the reduced CF conditions. Therefore, the NRC staff concludes that for the 55 percent CF statepoint and along the MELLLA+ upper boundary up to the minimum CF statepoint, the highest reduced CF

uncertainty will be applied." This conclusion imposes the requirement to use the higher SLO uncertainties for SLMCPR calculation in the MELLLA+ domain. The justification for this penalty was that the uncertainty in evaluating the core flow during SLO conditions should bound the uncertainty associated with the large void fraction operation, for which there was no experimental data available for benchmarking.

In MNGP, the Gardel uncertainties are used for SLMCPR calculation. The uncertainty for core flow at TLO is 2.5 percent, while the uncertainty at SLO is 6 percent. Operation in SLO also increases the assembly radial peaking uncertainty from [[ ]] in TLO to [[ ]] in SLO. The application of these increased SLO uncertainties to EFW operating points results in an increase of SLMCPR of approximately 0.01 in MNGP when using the approved SAFLIM3D methodology.

For operation in the new proposed EFW domain, this uncertainty remains in place until such time as MNGP provides experimental data collected inside the EFW domain to quantify the uncertainty on core flow.

OLMCPR/SLMCPR Penalties Conclusion:

The NRC staff has reviewed new information provided by the licensee, which includes:

1. Experimental void fraction data from prototypical AREVA ATRIUM 10XM assemblies operating up to exit void fractions close to 100 percent,

2. a sensitivity analysis showing that a ±0.05 bias in the void-quality correlation has a very small effect on SLMCPR, AOO ΔCPR, and OLMCPR,

3. Gamma scan data, including AREVA ATRIUM 10XM scans, which show good agreement between measurements and AREVA methods,

4. Comparison of TIP measurements versus AREVA methods predictions for plants loaded with AREVA fuels, which show no discernible trend with power-to-flow ratio, and no significant difference between pre- and post-EPU operation, and

5. TIP measurements for power-to-flow ratios as high as 52 MWth/Mlb/hr, which benchmark well against AREVA methods, however there is limited data in this region.

Based on the staff's review of this additional information, as described above, the staff concludes that

1. The 0.01 OLMCPR penalty to account for void-quality correlation uncertainty at higher voids may be removed because the data provided shows that the uncertainty does not increase at higher void fractions.

2. The 0.03 SLMCPR penalty to account for an increase in power distribution uncertainties at higher power-to-flow ratios is necessary to maintain a 0.03 operating margin and may not be removed.

3. The requirement to use the higher SLO core flow uncertainties for calculation of the SLMCPR inside the EFW region remains in place because the calculation of core flow (based on the drive flow measurement) in these off-normal conditions may be subject to larger uncertainties than at full flow and data has not been provided to justify a reduction in uncertainty. The staff concludes that the SLO uncertainty level should bound any increased uncertainty in these off-normal conditions. Only the SLO core flow uncertainty is applied to this calculation. SLO operation also affects the uncertainty of the power distribution, but this effect is caused primarily by the power asymmetry induced by the SLO, which would not be expected in EFW conditions.

Postulated Accidents

LOCAs:

Section 6.1 of ANP-3295P summarizes the results from the LOCA break spectrum analysis (ANP-3211P, Reference 39), and MAPLHGR limit analysis (ANP-3212P, Reference 40), reports.

Break spectrum calculations were performed for the maximum core flow (105 percent F) and minimum core flow (80 percent F) that would be allowed during operation at rated EPU power. In support of operation within the EFW, additional LOCA break spectrum analyses were performed at the low core flow boundary of the extended power/flow map (point “M”).

Mid-peaked and top-peaked axial power shapes were considered along with the following single failures: single-failure of battery (DC) power, single failure of diesel generator, single-failure of the HPCI system, single-failure of an LPCI injection valve, and single failure of one automatic depressurization system valve.

The LOCA break spectrum calculations were performed with the EXEM BWR-2000 Evaluation Model (Reference 41), which is the approved AREVA’s LOCA evaluation methodology. No deviations were taken from the approved methodology. The calculations were performed in conformance with 10 CFR 50, Appendix K, requirements and satisfy the event acceptance criteria identified in 10 CFR 50.46 (peak cladding temperature (PCT) < 2200 ^{o}F, local oxidation < 17 percent, <1 percent hydrogen generation, and long-term core coolability). The power level was increased by 2 percent at all analyzed points to reflect its maximum measurement uncertainty.

The limiting break location for the AREVA ATRIUM 10XM equilibrium core is the recirculation suction pipe (split break 3.3ft^2) with an assumed LPCI injection failure and mid-peaked power shape. The limiting break occurred at the 105 percent flow condition, and it is more limiting than both the 80 percent and 57.4 percent flow EFW region corners.

SLO LOCA analyses are described in ANP-3211P. The SLO analyses are performed with an initial power (before scram) of 102 percent, which maximizes the decay heat and is conservative for SLO operation. A 0.70 MAPLHGR multiplier is applied for SLO. The limiting SLO LOCA is the 3.4 ft^2 split pump suction line break with SF-LPCI and a mid-peaked axial power shape. With the 0.70 MAPLHGR multiplier, the TLO LOCA is more limiting than the SLO LOCA.

The MAPLHGR limit for AREVA ATRIUM 10XM fuel is calculated in ANP-3212P. The impact of thermal conductivity degradation (TCD) was analyzed with sensitivity studies using AREVA's newer code RODEX4 to increase the stored energy calculated by RODEX2, which is used to perform the analyses. The sensitivity studies show that the maximum PCT occurs for fresh fuel (at 0 GWd/MTU), which is not affected by TCD. The proposed MAPLHGR limit for AREVA ATRIUM 10XM fuel is 13.1 kW/ft, which is acceptable because the calculations with a smaller value (12.5 kW/ft) show margin to acceptable limits.

LOCA break spectrum analyses were performed for a full load of AREVA ATRIUM 10XM fuel. The licensee's claim is that full-core results are similar to mixed-core LOCA results because the [[

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For all conditions analyzed, the limiting LOCA scenario occurs at TLO and at the 105 percent flow condition, which is outside the EFW. Thus, the NRC staff concludes that the EFW domain extension has no detrimental impact on LOCA performance and the analyses presented are acceptable because the 10 CFR 50.46 acceptance criteria are satisfied when using the appropriate MAPLHGR limit.

Pump Seizure Accident:

Pump seizure is a mild accident and it is defined as an instantaneous stoppage of one recirculation pump shaft while the reactor is operating at TLO full power. The licensee concludes, and the NRC staff concurs that the TLO pump seizure is bounded by the consequences of LOCA events.

In addition to the TLO single-pump seizure, the licensee analyzes the SLO single-pump seizure, which is more limiting than at TLO. Using AREVA methods, the licensee analyzes this event on a cycle-specific basis and ensures it satisfies AOO criteria. Since SLO operation is not allowed in EFW, the analysis is performed only for EPU/MELLLA conditions using approved methods, and it is not bounding.

Thus, the NRC staff concludes that the pump seizure event is not limiting for MNGP and is properly analyzed using approved AREVA methods on a cycle-specific basis.

Control rod drop accident (CRDA):

The licensee has performed CRDA analyses for both the A and B sequence startups allowed by the banked withdrawal sequences using approved methodology in XN-NF-80-19PA. The maximum deposited fuel rod enthalpy is 227.7 calories per gram (cal/g), which is below the acceptance criterion of 280 cal/g. The 736 fuel rods exceed the 170 cal/g fuel damage threshold which is bounded by the number of failed rods assumed in the USAR.

Effects (like core loading and control rod patterns) are taken into account by the cycle-specific analyses. Thus, the NRC staff concludes that the CRDA event is properly analyzed using approved AREVA methods on a cycle-specific basis.

Fuel and Equipment Handling Accident:

The licensee states that the fuel handling accident radiological analysis of record for the alternate source term (AST) was dispositioned with consideration of AREVA ATRIUM 10XM core in L-MT-12-076 (Reference 42).

Fuel handling accidents (FHAs) are not affected by EFW operation. Thus, the NRC staff concludes that this event has been properly dispositioned.

Fuel Loading Error (Infrequent Event):

The licensee evaluates both mislocated and misoriented fuel bundle events, which are categorized as infrequent events, where the acceptance criteria are based on the offsite dose consequences. Using AREVA methods, the licensee has evaluated the consequences and the analysis concludes that a mislocated bundle may result in up to ΔCPR of 0.13. A misoriented bundle may result in up to ΔCPR of 0.25. Both ΔCPR values are well below the AOO ΔCPR of 0.41, and the proposed MCPR, which is greater than 1.55. Thus the consequences of mislocated and misoriented fuel bundle are minimal and should not result in any fuel failures or offsite dose consequences.

The NRC staff concludes that the fuel loading error event has been properly analyzed for MNGP using AREVA methods and all applicable criteria are satisfied.

Operating Limits and COLR Input

Section 8.0 of ANP-3295P documents the determination of operating limits for MNGP. MCPR and LHGR are calculated using standard approved methods. Table values are provided for all the conditions required for input to the COLR.

ATWS Evaluation

Overpressurization Analysis:

Overpressurization analyses with failure to scram have been performed for MNGP operating at both 80 percent and 105 percent flow with 1 SRVOOS and a 5 percent drift over the TS SRV opening setpoint for the remaining seven SRVs. Even though the analysis in support of this LAR assumes 1 SRVOOS the licensee has conservatively maintained no SRVOOS in EFW. Consistent with ATWS best-estimate assumptions, nominal conditions were assumed for dome pressure and feedwater temperature, and a nominal MSIV closure time of 4.0 seconds was assumed. Based on the analysis from ANP-3224P, a 20 psi adder was included to account for void-quality correlations, Doppler void effects and thermal conductivity degradation.

The pressure regulator failure open (PRFO) was determined to be the limiting event. The maximum calculated pressure is 1452 psig at the lower vessel, which is less than the 1500 psig criterion.

The NRC staff has reviewed Appendices D & E of ANP-3224P, where the impact of void-quality correlation errors and thermal conductivity degradation is evaluated. The staff concurs with the evaluation that a 20 psi adder is sufficient to account for these uncertainties.

Long-Term Evaluation:

Section 7 of ANP-3295P states that the MNGP analysis of record for isolation ATWS remains applicable after the transition to AREVA ATRIUM 10XM fuel because the void coefficient for GE14 and AREVA ATRIUM 10XM is very similar, GE14 has a slightly more negative void coefficient, but the difference is smaller than the variation as function of exposure during the cycle, Boron worth for both fuels is very similar. Therefore, the licensee concludes that the core power generation (and, thus, the heat load to containment) is not changed significantly by the introduction of the new fuel.

The NRC staff notes that the ODYN analysis of record in Table 9.3 of NEDC-33435P shows a significant margin to suppression pool limits (84 ºF), and the TRACG best-estimate ATWS with depressurization analysis in Table 9-4 of NEDC-33435P also shows significant margin (80 ºF). Therefore, the staff concludes that sufficient margin exists for long term cooling for an isolation ATWS event; minor changes in core response due to the introduction of a new fuel are not likely to erode this margin. Therefore, the staff concludes that the MNGP anaysis of record for isolation ATWS under EFW conditions is acceptable for EFW operation with AREVA ATRIUM 10XM fuel.

ATWSI:

The MELLLA+ SER in NEDC-33006PA imposes the following limitation 12.23.6 “For MELLLA+ applications involving GE fuel types beyond GE14 or other vendor fuels, bounding ATWS Instability analysis will be provided to the staff. Note: this limitation does not apply to special test assemblies.” To satisfy this limitation with the introduction of AREVA ATRIUM 10XM fuel, the licensee has submitted plant-specific ATWSI calculations, including both the transition and equilibrium cores. In the LAR, an unmitigated (i.e. no operator actions) case was analyzed. In the response to request for additional information (RAI) 8 and RAI 32 (Reference 28), the licensee provided calculations for the ATWSI event mitigated by the appropriate operator actions (i.e., modelling the prescribed operator actions), and a 2RPT ATWSI event. These two mitigated events have been documented in ANP-3435P. Only the mitigated case have been analyzed with the MICROBURN-B2 issues corrected (see discussion in Section 1.1 of ANP-3435P). The licensee presented these results to satisfy the MELLLA+ SER requirement to perform plant-specific ATWSI calculations following significant configuration changes, which is the case now with the introduction of AREVA ATRIUM 10XM fuel and the use of AREVA methods.

Based on the staff's evaluation of the licensee's ATWSI analyses, the staff concludes that:

1. With the prescribed operator actions, including 90 seconds to initiate water level reduction, MNGP is not susceptible to unstable oscillations during a turbine trip ATWSI, and, thus, satisfies the ATWS criteria. This confirms the existing licensing basis for MNGP GEH methods.

2. Since the mitigated ATWSI turbine trip event does not result in oscillations, the limiting ATWSI event becomes the 2RPT, where oscillations develop after the recirculating pump trip (RPT) and the OPRM scram fails to insert rods. The event has been analyzed and meets the ATWS acceptance criteria.

Mitigated ATWSI:

The licensee has submitted analysis of ATWSI events when operator actions are taken into account (i.e., mitigated ATWSI) for MNGP in ANP-3435P. When operator actions are credited at 90 seconds to reduce water level, no oscillations are predicted for the turbine trip with bypass (TTWBP) ATWS event. A sensitivity analysis was performed and the results indicate that significant unstable oscillations would develop if the operator delays water level reduction to [[ ]] Even with this delay, ATWS acceptance criteria would be satisfied.

Sensitivity analyses were also performed by destabilizing the reactor by increasing the fuel gap conductance or reducing the inlet friction. The changes affect [[ ]] Similar results were obtained by [[ ]] which resulted is insignificant ultimate PCT value changes.

Since the nominal (TTWBP) ATWS event does not result in oscillations (i.e., operator actions are credited at 90 seconds to reduce water level), the staff requested that the licensee perform a 2RPT ATWS, which becomes the limiting ATWSI event. The results were provided in ANP-3435P. The calculations show that during a 2RPT ATWS the PCT increases [[ ]] All ATWS criteria are satisfied for the 2RPT ATWS event.

ATWSI Conclusion

The staff has reviewed the ATWSI information provided by the licensee. The mitigated ATWSI analysis (i.e., operator actions are credited at 90 seconds to reduce water level) confirms the current licensing basis and unstable oscillations do not develop for the turbine trip event, making the 2RPT ATWSI the limiting case. The licensee has demonstrated that significant unstable oscillations develop if the operator delays water level reduction [[ ]] The licensee's calculations of the 2RPT ATWSI show that all the ATWS acceptance criteria are satisfied.

For all licensee ATWSI calculations, an artificially high hot rod peaking factor of 1.52 was used such that the initial hot node was either at or above the LHGR limit. An NRC staff review of the AREVA ATRIUM 10XM lattice calculations shows that the pin peaking factor is maximum for

fresh fuel and is a function of void, but is always lower than 1.35, therefore, the 1.52 factor used is conservative.

Thus, the staff concludes that limitation 12.23.6 of the MELLLA+ SER (NEDC-33006PA) is satisfied by the above calculations, and LAR satisfies all ATWSI acceptance criteria for MNGP.

Pressure Regulator Failed Open (PRFO)

The PRFO event is a depressurization event. The licensee has analyzed this event for a large number of power-flow operating conditions and determined that the minimum pressure with power >25 percent occurs for initiating conditions at 60 percent power and 4 percent flow. For all operating conditions, this transient is of little or no consequence to thermal limits, but it is a TS violation to operate with pressure <800 psia and power >25 percent. The analysis shows that following a PRFO event, TSs would be violated and a minimum pressure of 665 psia can be reached without violating thermal limits. The licensee has also documented the applicability of the CPR correlations for GE14 and AREVA ATRIUM 10XM fuel for pressures greater than 600 psia.

The NRC staff has reviewed the PRFO event evaluation in Section 7 of ANP-3295P and concurs with the evaluation that this event does not challenge thermal limits and operation in EFW does not modify the result because the limiting initial condition is outside the EFW domain.

Accident and Transient Analysis Conclusion

Overpressure analyses indicate that acceptable criteria are maintained with and without scram available as long as SRVOOS is limited to a single valve. The isolation ATWS analysis of record is still applicable to EFW operation with AREVA ATRIUM 10XM fuel. Consequences of depressurization events like PRFO are not impacted by EFW operation and do not challenge thermal limits. The NRC staff concludes that MNGP will continue to meet the intent of GDCs 4, 16, and 19, following implementation of the proposed EFW operating domain with the AREVA ATRIUM 10XM fuel.

3.3 Containment and Ventilation

3.3.1 Introduction

MNGP is a BWR-3 with a Mark I pressure suppression type primary containment. As described in Section 5.1 of the MNGP USAR, Revision 33, the primary containment encloses the reactor vessel (RV), the reactor coolant recirculation loops, and other branch connections of the 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.

3.3.2 Containment Integrity (Pressure and Temperature Response) Analysis

Containment integrity analysis, which consist of drywell pressure and temperature response and the suppression pool temperature response analyses during a design basis LOCA, depend on

the Mass and Energy (M&E) release in the containment and the conservatively biased assumed initial conditions inside the reactor and containment.

In response to an NRC staff RAI (Reference 3), the licensee was requested, while considering the differences in the proposed AREVA ATRIUM 10XM fuel and the current GE14 fuel (e.g., mass, material properties, core flow, decay heat, heat transfer coefficients and any other variations), to justify that the initial core stored energy, energy transferred to the reactor coolant, and the decay heat during the transient is bounded by the same in the current licensing basis analysis. The licensee was requested to provide a comparison of all parameters used in the analysis with qualitative reasons justifying that the value of each parameter in the current analysis leads to a transient that will bound the same transient in the analysis for the proposed fuel. In response to the RAI, the licensee stated that parameters that may affect the M&E release are: (a) decay heat, (b) heat transfer between the fuel and the reactor coolant, (c) void and Doppler reactivity coefficient for the GE14 and AREVA ATRIUM 10XM fuel, and (d) fuel and assembly components mass.

Decay heat is principally a function of the reactor power level, the irradiation time, and the time after shutdown. In response to the above RAI, the licensee stated that the following standards that were used for the decay heat calculation in the analysis of record (EPU/ MELLLA+ operating domain) are also applicable for AREVA ATRIUM 10XM decay heat calculation in the EPU EFW operating domain: ANSI/ANS-5.1-1971, “American Nuclear Society Decay Energy Release Rates Following Shutdown of Uranium-Fueled Thermal Reactors”, October 1971, ANSI/ANS-5.1-1979, “American Nuclear Society Decay Heat Power in Light Water Reactors”, August 1979, and ANSI/ANS-5.1-1994, “American Nuclear Society Decay Heat Power in Light Water Reactors”, August 1994.

The licensee stated that since these standards are also applicable for AREVA ATRIUM 10XM fuel, therefore, [[

]]

In response to SCVB RAI-2 the licensee provided the following comparison of the [[

]]

Review of the above decay heat comparison of the GE14 and AREVA ATRIUM 10XM fuel shows that the decay heat for both fuels is essentially the same.

The M&E release for containment response is also sensitive to the heat transfer between the fuel and the coolant which depends on the heat transfer coefficient and the core heat transfer area. In response to the above RAI, the licensee provided a qualitative comparison of the core average heat transfer capabilities for GE14 and AREVA ATRIUM 10XM fuels and concluded that they are very similar, and any differences would be bounded by the uncertainties in the analysis methods.

The licensee stated that the short term (during first few seconds) M&E release is influenced by the Doppler and void reactivity coefficients of the fuel. The licensee performed their comparison for GE14 and AREVA ATRIUM 10XM fuel and determined that changes in void and Doppler reactivity are small compared to the scram reactivity when the control blades were fully inserted.

The licensee stated that the fuel related input parameters (UO_2 (Uranium Oxide) and component mass) for M&E for containment response are very similar for the GE14 and AREVA ATRIUM 10XM fuel; their values used in the current licensing basis were increased by 10 percent to accommodate future change in fuel designs. Therefore, the current licensing basis stored energy in the fuel and the assembly components, which is proportional to their mass, is applicable for AREVA ATRIUM 10XM fuel.

The initial conditions inside the reactor and containment do not change from a GE14 EPU MELLLA+ to AREVA ATRIUM 10XM EPU EFW operating domain.

The NRC staff agrees with the licensee statement that transition to AREVA fuel operation is in the same parametric envelope of EPU/MELLLA+ and that it has no effect on the M&E release.

Short Term LOCA Analysis for Drywell Pressure Response

The licensee used LAMB computer code for the short term M&E release analysis and M3CPT computer code for the short term containment pressure and temperature response analysis for the current licensing bases. The licensee stated that no new analyses are needed for the short term M&E release or the short term containment pressure and temperature response because the M&E release and the reactor initial conditions which determine the pressure response are not changed.

In ANP-3376P (Reference 43), the licensee stated that the design basis limiting event for short term containment pressure is the recirculation suction line break. The licensee stated that the M&E release during a design basis LOCA, which depends on the decay heat, and the stored energy in the reactor does not change with the proposed fuel transition. Therefore, the short term drywell pressure response does not change from the current analysis of record (EPU/MELLLA+ analysis), which remains bounding. [[ ]]. The NRC staff finds that the licensee's assertion that neither M&E release nor the short term containment pressure and temperature response is affected by the proposed fuel transition is appropriate and, therefore, finds the licensee's evaluation acceptable.

Short Term LOCA Analysis for Drywell Gas Temperature Response

The licensee stated that the peak drywell temperatures for the EPU MELLLA and the MELLLA+ operating domain are [[ ]]. The drywell temperatures were obtained from the short-term design-basis accident (DBA) LOCA recirculation steam line break analyses performed with the M3CPT and LAMB codes. The evaluations were intended to show the effect of EPU MELLLA+ and do not need to be revised with new methods for AREVA ATRIUM 10XM fuel for EFW because it depends on the M&E release and the initial conditions, which do not change with the transition to the AREVA EFW domain. Therefore, the NRC staff finds this approach is acceptable.

Long Term LOCA Analysis for Suppression Pool Temperature Response

In ANP-3295P, the licensee provided evaluation of the long term pressure suppression pool temperature under EPU conditions operating in the AREVA EFW operating domain. The licensee stated that the current analyses assumes conservative M&E release and biased initial conditions for the suppression pool response, and the proposed fuel design does not impact the analysis because the same ANS standard is used for decay heat input. Therefore, the proposed transition to AREVA EFW domain has no effect on the M&E into the suppression pool.

The licensee also stated that [[

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

3.3.3 Hydrodynamic Loads

The key parameters [[

]]. The licensee evaluated the hydrodynamic loads in the MELLLA+ application. Since the operating domain for MELLLA+ and EFW is the same and the current containment pressure and temperature response remains bounding, the NRC staff finds the licensee's evaluation acceptable.

3.3.4 SRV Loads

The SRV loads are affected by the setpoints, sensible, and decay heat. The licensee provided a comparison of the MNGP specific decay heat for GE14 and AREVA ATRIUM 10XM which was the same to three decimals. The licensee stated that because the SRV setpoints and sensible and decay heat do not change in the EFW operating domain, the containment loads due to SRV discharge are unaffected. Therefore, the NRC staff finds the licensee's evaluation acceptable.

3.3.5 Subcompartment Analysis

The subcompartment pressurization loads are affected by the short term containment response. The licensee stated that the [[

]]. Since the [[

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

3.3.6 Post-LOCA Combustible Gas Control System

MNGP has eliminated the requirements for hydrogen recombiners and committed to maintain hydrogen and oxygen monitoring systems. 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 on May 21, 2004 (Reference 44), which approved the removal of the requirements for hydrogen recombiners, allowing them to be abandoned in place. Operation in the EFW operating domain does not affect the current combustible gas control system. Therefore, the NRC staff finds the licensee's current method acceptable.

3.3.7 Emergency Core Cooling System and Containment Heat Removal Pumps Net Positive Suction Head

The ECCS and CHRS pumps are the residual heat removal (RHR) and core spray (CS) pumps. SECY-11-0014 (Reference 45), Section 6.6, provides NRC guidance for a licensee that credits containment accident pressure for calculating the net positive suction head (NPSH) available (NPSHa) at the suction inlet of the RHR and CS pumps. In its December 2014, supplement to their LAR, the licensee states that its calculations are conservatively biased to minimize the NPSH and not constrained by fuel design. The licensee's analysis of record was calculated within the same operating window as EFW. The licensee used a conservative approach to ECCS CHRS NPSH. Therefore, the NRC staff finds this acceptable.

3.3.8 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." The licensee states that the operating domain does not adversely affect the main control room atmosphere control system because the source terms and release rates are not changed, as described in the licensing basis for MNGP MELLLA+. The operating domain for MELLLA+ and EFW are identical. [[

]] The NRC staff finds that this is an acceptable approach since the operating domains or MELLLA+ and EFW are identical and the licensing basis fuel is bounding.

3.3.9 Standby Gas Treatment System (SBGTS)

The SBGTS maintains the secondary containment at a negative pressure and filters the exhaust air by removing fission products present during abnormal conditions. Transition to EFW does not impact the current analyses and it therefore remains valid. Accordingly, the NRC staff finds this acceptable.

3.3.10 Containment Isolation

The containment isolation system is affected by the containment pressure and temperature response under DBA conditions. In the December 2014, supplement, the licensee stated that operating within the EFW window with AREVA fuel will have no effect on containment isolation functions because there is no change to the limiting containment conditions. The NRC staff agrees with the licensee's conclusion and, therefore, finds it acceptable.

3.3.11 Generic Letter (GL) 89-10

The response to 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 DBA conditions. The licensee states that, "the process temperatures and heat load from motors and cables are bounded by the EPU process temperatures and heat loads and as such are within the design of the heating, ventilation, and air conditioning (HVAC) equipment chosen for worst conditions." The licensee concluded that operating in the EFW

domain would not result in more severe conditions and GL 89-10 is not impacted. The NRC staff agrees with the licensee's conclusion and, therefore, finds it acceptable.

3.3.12 GL 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 EFW operating domain. The NRC staff agrees with the licensee's conclusion and, therefore, finds it acceptable.

3.3.13 GL 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 DBA conditions. The licensee concluded that the transition to EFW will not result in a more severe containment pressure and temperature response and the current evaluation is valid. The NRC staff agrees with the licensee's conclusion and, therefore, finds it acceptable.

3.3.14 GL 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 DBA conditions. Since there is no impact to the containment pressure and temperature response analyses, an evaluation in response to GL 96-06 is not required. Therefore, the NRC staff finds the licensee's conclusion acceptable.

3.3.15 Containment and Ventilation Conclusion

As explained above, the NRC staff has reviewed the licensee's assessment of the aforementioned topics, and concludes that they are adequately addressed for MNGP for the AREVA ATRIUM 10XM fuel in the EFW operating domain. The NRC staff also concludes that MNGP will continue to meet the intent of GDCs 4, 16, 19, 38, 41, and 50, following implementation of the proposed EFW operating domain with the AREVA ATRIUM 10XM fuel.

3.4 Instrument and Control

3.4.1 Introduction

The NRC Staff Requirements Memorandum (SRM) on SECY-93-087 (Reference 46) describes the position of NRC regarding diversity and defense-in-depth (D3). This 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; "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 SAR using best estimate methods. The vendor or applicant shall demonstrate adequate diversity within the design for each of these events."

3.4.2 Evaluation

The OPRM upscale trip 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 314.6 “Plant Stability Analysis” of the MNGP USAR; a failure of the NUMAC OPRM or average power range monitor (APRM) could disable the automatic safety trip function performed by the OPRM Upscale Trip algorithms. The MNGP NUMAC system includes a means of providing an automatic EFWS trip function. This EFWS trip function is required to be enabled when the plant is within the EFW region of operation. The EFWS trip serves as a backup to the OPRM Upscale Trip function; however, the NRC staff notes that use of common software for both primary (OPRM upscale trip) and backup (EFWS) stability protection can lead to a condition where both of these automatic functions would become disabled due to a common-cause software error.

If both the OPRM upscale trip function becomes inoperable and the EFWS function is inactive, cannot be implemented, or is inoperable, then manual operator actions become the only available means of providing core stability protection. The MNGP power – recirculation flow graph contains two regions which require manual actions to prevent core instabilities during operations. When plant conditions lead to operation within region II, manual BSP solution involves immediate actions to exit the region. When plant conditions result in operation within Region I, administrative actions require initiation of a manual reactor scram.

Because of 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 common-cause failure (CCF) of the NUMAC power range neutron monitoring (PRNM) (APRM/OPRM) system in conjunction with the plant instability events described in the MNGP UFSAR, Chapter 14. This analysis identified Manual Operator Actions as diverse means of maintaining plant safety when the automatic trip functions performed by the OPRM algorithms become unavailable due to a postulated common-mode failure of the NUMAC PRNMS.

The D3 analysis identified that a postulated CCF in the PRNMS 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, PRNMS indications of power and flow would track consistently with other plant indicators as they change to a statepoint 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 protection when large amplitude oscillations begin to occur. Because of this, operators will have necessary indications to identify plant operation in the BSP in Regions I and II of the power to core flow map and will be able to initiate manual actions to assure plant safety.

The D3 analysis identified multiple diverse control room indications [[
]] that are independent from the effects of the postulated PRNMS CCF. When operation in region I is identified and the OPRM is inoperable, MNGP operators are procedurally required to insert a manual scram. This immediate action is uncomplicated and is completed by simultaneously depressing two reactor scram push buttons that are located at the control room panel. Confirmation that the manual scram is successful is unambiguous and provided by the control rod display within a few seconds. The NRC staff confirmed that the systems used for initiation of the manual scram and for confirmation that the scram was successful do not rely on digital or software based technologies. The NRC staff determined these systems would

therefore not be affected by the postulated software CCF that renders the automatic protection functions inoperable.

Credited diverse manual operator actions are taken to control start-up trajectories on the power to core flow map. These include adjustment of recirculation flow and control rod positions. [[

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MNGP operators are procedurally required to first identify if the plant has entered into a region of thermal hydraulic instability and then to insert a manual scram if operation within region I is identified. The NRC staff confirmed the systems used for controlling core flow, reactor power and manual scram do not rely on NUMAC based technology. The NRC staff determined these systems would, therefore, not be affected by the postulated software CCF of the PRNMS that renders 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 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 to mitigate an ATWS event are sufficient to prevent excessive clad temperatures with sufficient margin. The NRC staff acknowledges that core damage can be avoided during an ATWS event and ATWS could provide an additional means of meeting the acceptance criteria of BTP 7-19.

Setpoint Review

The setpoint methodology used for determination of the EFWS setpoints is described in engineering evaluation EC 25987 (Reference 47). This method includes three calculations to determine EFWS scram analytical limits, and operating limits as well as nominal trip setpoints and allowable values for scram and rod block functions. The NRC staff reviewed this methodology and determined it provided an acceptable means of establishing the EFWS setpoints to ensure adequate margin exists between the instrument setpoints and the analytical limits established to ensure plant safety.

The first calculation uses plant empirical data to predict core flow using reactor power and recirculation drive flow. A second calculation is then performed to determine EFWS setpoint analytical limits and to determine the separation between the EFWS trip and rod block allowable values. Finally, a third calculation is performed to determine the process instrumentation setpoints and operating limits. EFWS setpoints are based on the CIER of operation. The CIER is determined during the core flow mapping activity and includes a 5 percent bias offset to account for operational variances in bundle conditions. Calculations of EFWS setpoints also account for uncertainty terms. [[

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The licensee provided a proprietary EFWS setpoint calculation as Enclosure 2 to the August 26, 2015, letter. The NRC staff evaluated this calculation to determine how core flow measurement uncertainties were addressed and found that multiple uncertainty factors were included in the instrument uncertainty determinations and that acceptable methods for combining these factors were used to determine instrument settings. The NRC staff concludes that adequate margins are established between instrument settings and analytical safety limits

such that reactor safety parameters will be maintained within acceptable limits during plant operation.

3.4.3 Instrument and Control Conclusion

The NRC staff concludes that the proposed EFW methodology is an acceptable solution, because it provides sufficient protection commensurate with the probability of an instability event in the period of time they are active. This evaluation also concludes manual control measures needed to support EFW protection are sufficiently diverse from the digital PRNMS NUMAC systems and, therefore, provide an acceptable means of diverse protection for the OPRM upscale trip function.

The NRC staff determined the proposed license amendment to revise MNGP Renewed Operating License including the TSs allowing the plant operation in the EFW domain with manual BSP provides reasonable assurance of adequate protection of public health, safety, and security based on its evaluation which applies current applicable regulatory evaluation criteria identified in the regulatory analysis in Section 2.0 of this SE.

3.5 Radiological Consequences

The original AST analyses for MNGP was submitted for NRC staff approval in an LAR dated September 15, 2005 (Reference 48). The submittal contained the radiological consequence analyses based upon the AST methodology for the following four DBAs that result in CR and offsite exposure.

- LOCA
- FHA
- CRDA
- Main Steam Line Break (MSLB)

The submittal also included changes to the MNGP TSs and associated Bases to reflect implementation of AST assumptions in accordance with 10 CFR 50.67. The licensee evaluated the impact of EFW operation on the DBAs in Section 6.0 of ANP-3295NP. A disposition of events summary is also provided in Table 2.1 of ANP-3295NP to support this proposed LAR. The disposition summary presents a list of the events and analyses, the corresponding USAR section, and the disposition status of each event for transitioning to EFW conditions under AREVA methodologies. Based on the EFW domain representing the same region as Maximum Extended Load Line Limit Analysis Plus (MELLLA+), there is no change in consequences of postulated accidents when operating in the EFW operating domain compared to the operating domain previously evaluated. The results of accident evaluations remain within the NRC approved acceptance limits. The dose acceptance criteria for the FHA and CRDA are a TEDE of 6.3 rem at the EAB and LPZ for an analysis duration of 2 and 24 hours, respectively, and 5 rem in the CR for the duration of the accident.

3.5.1 LOCA

To support the approved amendment 188 to transition to the AREVA ATRIUM 10XM fuel at EPU/MELLLA+ conditions, the licensee revised the LOCA radiological consequence analysis using the approved AST methodology that is described in MNGP USAR, Section 14.7.2. The

NRC staff found that the LOCA radiological dose consequence analysis was consistent with the guidance provided in Regulatory Guide (RG) 1.183 (Reference 49). The NRC staff compared the doses estimated by the licensee and concluded that the radiological consequences at the EAB, LPZ, and in the CR are within the dose criteria specified in 10 CFR 50.67.

3.5.2 FHA

The number of fuel rods assumed to fail during a fuel handling accident for an ATRIUM 10XM assembly dropping over the core was analyzed in support of the fuel transition (Amendment 188). This accident is independent of operation with AREVA ATRIUM 10XM at EFW conditions. The FHA is described in MNGP USAR, Section 14.7.6. To support the approved Amendment 188 to transition to the AREVA ATRIUM 10XM fuel, the licensee reviewed the quantity of fuel rod damage following the postulated drop of an AREVA ATRIUM 10XM fuel assembly and calculated the radiological source term from AREVA ATRIUM 10XM fuel rods. The licensee's evaluation also showed that the overall accident dose from a FHA would be lower for the AREVA ATRIUM 10XM fuel than the previously licensed GE14 fuel. Based on the above, the NRC staff finds that the current MNGP FHA analysis to be bounding. Therefore, the MNGP FHA regulatory dose limits are unaffected and still meet the regulatory requirement in 10 CFR 50.67 and the accident specific dose criteria described in SRP 15.0.1.

3.5.3 CRDA

To support the approved Amendment No. 188 to transition to the ATRIUM 10XM fuel, the licensee revised the CRDA radiological consequence analysis using the previously approved AST methodology that is described in MNGP USAR, Section 14.7.1. MNGP will continue to evaluate the CRDA on a cycle-specific basis when using AREVA methods. The licensee performed an evaluation of the CRDA for a representative transition cycle. The evaluation showed the number of rods calculated to fail in this event remains below the value of 850 assumed in the MNGP USAR radiological evaluation of this event. The revised CRDA calculation uses the source terms for the proposed AREVA ATRIUM 10XM fuel design. The NRC staff found that the CRDA radiological dose consequence analysis was consistent with the guidance provided in RG 1.183. The NRC staff reviewed the doses estimated by the licensee and concluded that the radiological consequences at the EAB, LPZ and in the CR are within the dose criteria specified in 10 CFR 50.67 and accident specific dose criteria described in SRP 15.0.1.

3.5.5 MSLB

The MSLB accident is described in MNGP USAR, Section 14.7.3. As stated in the MNGP USAR, no fuel failures are expected to occur as a result of this accident. The radionuclide inventory released from the primary coolant system is present in the coolant prior to the event. Therefore, MSLB accident analysis is not affected by a change in fuel design. Based upon this information, the NRC staff finds that the proposed change does not alter the radiological consequences of a MSLB accident. The MNGP MSLB regulatory dose limits are unaffected and continues to meet the regulatory requirement in 10 CFR 50.67 and accident specific dose criteria described in SRP, 15.0.1.

3.5.6 Radiological Consequence Conclusion

The NRC staff reviewed the analyses used by the licensee to assess the radiological impacts of the transition to the EFW domain. The staff found that MNGP continues to comply with the regulatory and design basis criteria established for the AST. The results of accident evaluations remain within the NRC approved acceptance limits. The staff also finds, with reasonable assurance that the licensee's estimates of the EAB, LPZ, and CR doses will continue to comply with these criteria. Therefore, the proposed change is acceptable with regard to the radiological consequences of postulated DBAs.

3.6 Limitations and Conditions

3.6.1 EO-III Long Term Stability Solution

ANP-10262(P)(A) describes the EO-III long term stability solution. ANP-10262Q1P and ANP-10262Q2P (Reference 50) are a series of RAI responses that were submitted during the review of the EO-III application and AREVA's DIVOM methodology.

The limitations and conditions imposed on the ANP-10262(P)(A), SER apply to all applications of EO-III and they also apply to MNGP. The limitations and conditions and the staff's evaluation of their applicability for EFW is described below.

1. The NRC staff has not reviewed the hardware and software implementation of EO-III Long Term Stability Solution because it will be plant specific. AREVA has stated that implementation is not part of the generic EO-III Long Term Stability Solution, even though the EO-III Long Term Stability Solution implements an additional scram function (the channel-stability exclusion region) not present in the original Option III platforms. Plant implementations, including those using any original Option III platform, will require plant-specific reviews.

The Monticello EFW application uses the approved GEH Solution 3 NUMAC hardware, which is acceptable since it is the same hardware necessary to implement EO-III.

2. The original Option III is already approved for plant operation up to 20 percent EPU. The EO-III Long Term Stability Solution is an extension of Option III, where the DIVOM correlation is guaranteed to be well-behaved by the channel-stability exclusion region. Thus, the EO-III Long Term Stability Solution is, in essence, an Option III implementation with the added channel stability exclusion region scram. Therefore, the NRC staff finds that EO-III is a technically acceptable methodology for any reactor operating up to 20 percent EPU conditions.

The Monticello EFW LAR only request operation up to 120 percent OLTP (20 percent EPU); therefore, use of EO-III is acceptable for EFW.

3. The confirmation analyses documented in Section 5 of TR ANP-10262(P), Revision 0, and the response to the NRC staff RAI, indicate that the EO-III Long Term Stability Solution methodology provides significant protection against MCPR criteria violations during anticipated instability events even under high-power-density conditions, including

EPU and MELLLA+. Under all analyzed conditions, the loss of MCPR margin induced by the instability event is compensated by the gain in MCPR margin induced by the reduction in flow, so that the net MCPR margin is positive. Based on this analysis, the NRC staff finds that the EO-III Long Term Stability Solution is a technically acceptable methodology for any reactor operating up to MELLLA+ conditions. Extension of operating domains beyond MELLLA+ have not been considered by the NRC staff and will require a re-evaluation of the EO-III Long Term Stability Solution scram effectiveness by the NRC staff.

The Monticello EFW LAR only requests operation inside the approved MELLLA+ domain. Therefore, the staff finds that the use of EO-III is acceptable.

4. Operation with feedwater heaters out of service (FWHOOS) is not anticipated in EFW like MELLLA+; therefore TR ANP-10262(P), Revision 0, specifies the use of equilibrium feedwater conditions. If a plant-specific application of the EO-III Long Term Stability Solution allows for a FWHOOS condition, two SPT regions will have to be calculated, with and without FWHOOS. TSs must enforce the change of SPT region settings when the FWHOOS condition is declared. Alternatively, a plant-specific application may choose to implement the more conservative of the two SPT regions.

The Monticello EFW application does not request operation with FWHOOS. Therefore, the staff finds that the use of EO-III is acceptable

5. The EO-III Long Term Stability Solution does not provide an integrated backup stability solution if the primary stability protection system is declared inoperable. Instead, it provides an example of TSs and rationale for their applicability in the responses to the NRC staff RAI. Therefore, the NRC staff review of the stability related TS requirements and/or a different backup stability solution must be performed on a plant-specific basis.

The Monticello EFW application proposes the channel instability exclusion region (SPT region) as an acceptable backup stability solution. As discussed in Section 3.2.4 of this SE, the staff finds the the use of the SPT region as the backup stability solution acceptable because it provides protection for the most likely instability scenario, which is a fast flow reduction following a recirculation pump trip.

6. Plant-specific applications will include the specification of the backup stability protection. One possible EO-III Long Term Stability Solution backup stability protection is the SPT, which provides an automated scram upon entry on the channel-stability exclusion region. The SPT is an acceptable backup stability protection solution for up to 120 days (typical TS range) if the primary stability protection system is declared inoperable. However, the SPT scram region must include the natural circulation line. To be an acceptable backup stability solution, the SPT must include the following scram conditions: (1) recirculation pumps are tripped, or (2) the power flow inside the channel stability exclusion region. Either of the two conditions should result in scram.

   In addition to the SPT, administrative interim corrective actions must be enforced with cycle-specific regions. The NRC staff finds that a SPT implemented with the above conditions would provide an acceptable backup stability implementation for up to 120

days under MELLLA+ conditions because it provides protection for the most likely scenarios where large amplitude oscillations could occur.

The Monticello EFW application specifies the SPT region as the backup stability solution, which as discussed in section 3.2.4 of this SE, is acceptable.

7. Plant specific applications will include an evaluation of the uncertainty induced by the presence of bypass voids on the OPRM and APRM readings. OPRM uncertainties will result in a set-down of the OPRM PBDA setpoint APRM uncertainties will be applied to the SPT exclusion region.

The licensee included in ANP-3295 the impact of bypass voiding on the SPT region in the SPT definition for Monticello. Therefore, the staff finds the plant specific evaluation of uncertainty caused by bypass voiding acceptable.

The limitations and conditions imposed on the BAW-10255(P)(A) SER apply to all applications of EO-III and they also apply to MNGP. They are:

1. If a reduced scope of parameter variations is used to define the cycle-specific DIVOM slope as described in Section 7 of the TR, the scope must be justified and documented for NRC staff review.

2. The NRC staff imposes a condition to perform a full code review of RAMONA5-FA, including constitutive relations, numerics, neutronic methods, and benchmarks before RAMONA5-FA can be used to calculate DIVOM curves in EFW operating domains without the 10 percent penalty on DIVOM slopes, as noted in Limitation and Condition No.3 below.

3. The NRC staff imposes an interim 10 percent penalty on DIVOM slopes calculated using the RAMONA5-FA methodology under EFW conditions. This is an interim restriction that will be revised when the full RAMONA5-FA Code review is completed.

For item 1, a reduced scope of parameters was not used for MNGP so the limitation is not applicable. For item 2, the NRC performed the RAMONA5-FA evaluation and issued a letter March 8, 2010. Therefore this condition has been satisfied. For item 3, the 10 percent penalty does not apply because the RAMONA5-FA review has already been completed by the NRC staff.

EO-III Long Term Stability Solution Conclusion

Based on the considerations discussed above, the staff finds that the limitations and conditions related to EO-III Long Term Stability Solution have been adequately addressed for operation in the EFW domain.

3.6.2 Limitations from NEDC-33173P, “Applicability of GE Methods to Expanded Operating Domains”

The purpose of NEDC-33173P was to approve the use of GE’s methods for the MELLLA+ operating domain. Its associated SE contained several limitations. Since the licensee is

requesting approval for use of the AREVA methods in the EFW operating domain (parametrically the same operating domain as MELLLA+), it is necessary to ensure that any limitation of NEDC-33173P that could be applicable to the AREVA methods used in this LAR are addressed.

The following is the NRC staff's disposition of the limitations described in NEDC-33173P (Reference 52).

Limitation 9.1, TGBLA/PANAC Version

> The neutronic methods used to simulate the reactor core response and that feed into the downstream safety analyses supporting operation at EPU/MELLLA+ will apply TGBLA06/PANAC11 or later NRC-approved version of neutronic method.

The licensee states that this limitation is not applicable because they use CASMO4/MICROBURN-B2, which is an approved method. The NRC staff has reviewed CASMO4/MICROBURN-B2 and determined that it is applicable to MNGP. In Appendix-A of this SE, "Methods Extension Review, Code Evaluation for EFW Applicability, CASMO4/MICROBURN-B2," the staff concludes that the use of CASMO4/MICROBURN-B2 in the Monticello EFW domain is an acceptable extension of the existing approval. Therefore, the staff agrees that this limitation is not applicable to MNGP.

Limitation 9.2, 3D Monicore

> For EPU/MELLLA+ applications, relying on TGBLA04/PANAC10 methods, the bundle RMS [root mean square] difference uncertainty will be established from plant-specific core-tracking data, based on TGBLA04/PANAC10. The use of plant-specific trendline based on the neutronic method employed will capture the actual bundle power uncertainty of the core monitoring system.

The licensee states that this limitation is applicable, and that Xcel Energy has determined uncertainties applicable to the Gardel core monitoring system for MNGP, with these uncertainties being used in the statistical analyses for SLMCPR and LHGR. The staff has reviewed the applicable uncertainties in the LAR as supplemented. Based on its review, the staff concurs that the uncertainties are properly accounted for in the LAR and finds this disposition of the limitiation is acceptable.

Limitation 9.3, Power-to-Flow Ratio

> Plant-specific EPU and expanded operating domain applications will confirm that the core thermal power to core flow ratio will not exceed 50 MWt/Mlbm/hr at any statepoint in the allowed operating domain. For plants that exceed the power-to-flow value of 50 MWt/Mlbm/hr, the application will provide power distribution assessment to establish that neutronic methods axial and nodal power distribution uncertainties have not increased.

The power-to-flow ratio in Monticello is 49.94 MWt/Mlbm/hr, which is lower than the criteria to trigger an evaluation; therefore, the staff finds the disposition of this limitation acceptable.

Limitation 9.4, SLMCPR 1

> For EPU operation, a 0.02 value shall be added to the cycle-specific SLMCPR value. This adder is applicable to SLO, which is derived from the dual loop SLMCPR value.

The licensee states that the limitation is not applicable, as the NRC did not impose an adder during the review and approval of the transition to AREVA fuel and methods for Monticello for EPU, consistent with previous reviews of AREVA methods for EPU in Brunswick, and particularly in Susquehanna which uses neutron TIPs as does Monticello. In response to SRXB RAI 2, AREVA stated that "There are no proposed penalties for Monticello OLMCPR or SLMCPR with AREVA methods." The staff has reviewed the LAR and RAI response and agrees with the disposition of this limitation.

Limitation 9.5, SLMCPR 2

> For operation at MELLLA+, including operation at the EPU power levels at the achievable core flow statepoint, a 0.03 value shall be added to the cycle-specific SLMCPR value.

The NRC staff did not impose an adder during the review and approval of the transition to AREVA fuel and methods for MNGP for EPU, consistent with previous reviews of AREVA methods for EPU in Brunswick, and, particularly, in Susquehanna which uses neutron TIPs as does MNGP. However, operation in the EFW region increases the core-average void fraction, and MNGP lacks operating experience in this region. The staff concludes that the information provided by Xcel does not sufficiently justify the removal of the 0.03 SLMCPR penalty and, therefore, the 0.03 operating margin is necessary above 42MWt/Mlbm/hr. By letter dated September 14, 2016 (Reference 61), the licensee proposed a revision to TS 2.1.1, "Reactor Core SLs," to address the addition of this penalty. Section 3.7.1 of this SE provides the staff's evaluation of the proposed TS revision. Because the licensee's proposed revision to TS 2.1.1 sufficiently addresses the addition of the .03 penalty, the staff finds the disposition of this limitation acceptable.

Limitation 9.6, R-Factor

> The plant specific R-factor calculation at a bundle level will be consistent with lattice axial void conditions expected for the hot channel operating state. The plant-specific EPU/MELLLA+ application will confirm that the R-factor calculation is consistent with the hot channel axial void conditions.

The licensee states that the limitation is applicable, and that R-factors may be appropriately determined with their existing methodology based on F_{eff} for SPCB (GE-14 fuel) and K-factor for ACE (AREVA ATRIUM 10XM fuel) documented in EMF-2245 (Reference 53), methodology. The NRC staff reviewed EMF-2245 and confirmed that the R-factor calculation is consistent with the hot channel axial void conditions and therefore finds this disposition acceptable.

Limitation 9.7, ECCS-LOCA 1

> For applications requesting implementation of EPU or expanded operating domains, including MELLLA+, the small and large break ECCS-LOCA analyses will include top-peaked and mid-peaked power shape in establishing the MAPLHGR and determining the PCT. This limitation is applicable to both the licensing bases PCT and the upper bound PCT. The plant-specific applications will report the limiting small and large break licensing basis and upper bound PCTs.

The licensee states that the limitation is applicable, and that their LOCA calculations include top-peaked and mid-peaked power shapes, as well as large and small break PCTs. However, the licensee states that they do not calculate upper bound PCT. The staff finds that the licensee provided the PCT relevant to the Monticello licensing bases and therefore, this disposition acceptable.

Limitation 9.8, ECCS-LOCA 2

> The ECCS-LOCA will be performed for all statepoints in the upper boundary of the expanded operating domain, including the minimum core flow statepoints, the transition statepoint, as defined in Reference 2, and the 55 percent core flow statepoint. The plant-specific application will report the limiting ECCS-LOCA results as well as the rated power and flow results. The SRLR [Supplemental Reload Licensing Report] will include both the limiting statepoint ECCS-LOCA results and the rated conditions ECCS-LOCA results.

The licensee states that the limitation is applicable and that calculations for the maximum and minimum core flow at rated EPU power and the low core flow MELLLA+ boundary (replacing the transition statepoint) have been performed for both power shapes. The NRC staff finds that the licensee has provided the information pertinent to this limitation and therefore the licensee's disposition acceptable.

Limitation 9.9, Transient LHGR 1

> Plant-specific EPU and MELLLA+ applications will demonstrate and document that during normal operation and core-wide AOOs, the T-M [thermal-mechanical] acceptance criteria as specified in Amendment 22 to GESTAR II will be met. Specifically, during an AOO, the licensing application will demonstrate that the: (1) loss of fuel rod mechanical integrity will not occur due to fuel melting and (2) loss of fuel rod mechanical integrity will not occur due to pellet–cladding mechanical interaction. The plant-specific application will demonstrate that the T-M acceptance criteria are met for both the UO_2 and the limiting Gd [Gadolinium] O_2 rods.

The licensee states that the limitation is applicable, and that compliance with the T-M acceptance criteria for AOOs has been demonstrated and documented using the most recent NRC-approved method in TR BAW-10247PA (Reference 54), including the use of RODEX4. The NRC staff reviewed BAW-10247PA to ensure that the T-M acceptance criteria has been demonstrated and finds this disposition acceptable.

Limitation 9.10, Transient LHGR 2

> Each EPU and MELLLA+ fuel reload will document the calculation results of the analyses demonstrating compliance to transient T-M acceptance criteria. The plant T-M response will be provided with the SRLR or COLR, or it will be reported directly to the NRC as an attachment to the SRLR or COLR.

The licensee states the limitation is applicable, that compliance to transient T-M acceptance has been documented in ANP-3295, and that T-M calculations will be performed each cycle and reported in the cycle-specific licensing analysis report as described in the limitation. Therefore, the NRC staff finds this disposition acceptable.

Limitation 9.11, Transient LHGR 3

> To account for the impact of the void history bias, plant-specific EPU and MELLLA+ applications using either TRACG or ODYN will demonstrate an equivalent to 10 percent margin to the fuel centerline melt and the 1 percent cladding circumferential plastic strain acceptance criteria due to pellet-cladding mechanical interaction for all of limiting AOO transient events, including equipment out-of-service. Limiting transients in this case, refers to transients where the void reactivity coefficient plays a significant role (such as pressurization events). If the void history bias is incorporated into the transient model within the code, then the additional 10 percent margin to the fuel centerline melt and the 1 percent cladding circumferential plastic strain is no longer required.

The licensee states that the limitation is not applicable, as the analysis used the GEH MOP/TOP criteria for GE14 fuel, and the most current NRC-approved T-M methods for AREVA ATRIUM XM fuel in BAW-10247PA. The response to RAI-29 compares COTRANSA2-calculated transient response to measured data for relatively recent transient events, which are more applicable to modern fuels and current reactor operating modes. The NRC staff has reviewed the reference and determined that the methods are applicable to MNGP. Therefore, the NRC staff finds this disposition acceptable.

Limitation 9.12, LHGR and Exposure Qualification

> In MFN 06-481, GE committed to submit plenum fission gas and fuel exposure gamma scans as part of the revision to the T-M licensing process. The conclusions of the plenum fission gas and fuel exposure gamma scans of GE 10x10 fuel designs as operated will be submitted for NRC staff review and approval. This revision will be accomplished through Amendment to GESTAR II or in a T-M licensing LTR. PRIME (a newly developed T-M code) has been submitted to the NRC staff for review (Reference A-3). Once the PRIME LTR and its application are approved, future license applications for EPU and MELLLA+ referencing LTR NEDC-33173P must utilize the PRIME T-M methods.

The licensee states that the limitation is not applicable, as they are using the most current NRC-approved methods in TR BAW-10247PA. The NRC staff has reviewed the reference and

determined that the methods are applicable to MNGP. Therefore, the staff finds this disposition acceptable.

Limitation 9.13, Application of 10 Weight Percent Gd

> Before applying 10 weight percent Gd to licensing applications, including EPU and expanded operating domain, the NRC staff needs to review and approve the T-M LTR demonstrating that the T-M acceptance criteria specified in GESTAR II and Amendment 22 to GESTAR II can be met for steady-state and transient conditions. Specifically, the T-M application must demonstrate that the T-M acceptance criteria can be met for TOP and MOP conditions that bounds the response of plants operating at EPU and expanded operating domains at the most limiting statepoints, considering the operating flexibilities (e.g., equipment out-of-service).
>
> Before the use of 10 weight percent Gd for modern fuel designs, NRC must review and approve TGBLA06 qualification submittal. Where a fuel design refers to a design with Gd-bearing rods adjacent to vanished or water rods, the submittal should include specific information regarding acceptance criteria for the qualification and address any downstream impacts in terms of the safety analysis. The 10 weight percent Gd qualifications submittal can supplement this report.

The licensee states that the limitation is not applicable, as they are using the most current NRC-approved T-M methods in TR BAW-10247PA and neutronics methods TR EMF-2158(P)(A). The NRC staff has reviewed the references and determined that the methods are applicable to MNGP. Therefore, the NRC staff finds this disposition acceptable.

Limitation 9.14, Part 21 Evaluation of GESTR-M Fuel Temperature Calculation

> Any conclusions drawn from the NRC staff evaluation of the GE's Part 21 report will be applicable to the GESTR-M T-M assessment of this SE for future license application. GE submitted the T-M Part 21 evaluation, which is currently under NRC staff review. Upon completion of its review, NRC staff will inform GE of its conclusions.

The licensee states that the limitation is not applicable, and that the AREVA methods for evaluating the impact of pellet thermal conductivity degradation using RODEX2 and RODEX4 have been approved for MNGP. The NRC staff has reviewed the references and determined that the methods are applicable to MNGP. In Appendix-A of this SER, Methods Extension review, Code Evaluation for EFW Applicability, the staff concludes that the use of RODEX2 and RODEX4 in the Monticello EFW domain is an acceptable extension of the existing approval. Therefore, the NRC staff finds this disposition acceptable.

Limitation 9.15, Void Reactivity 1

> The void reactivity coefficient bias and uncertainties in TRACG for EPU and MELLLA+ must be representative of the lattice designs of the fuel loaded in the core.

The licensee states that the limitation is applicable, and that related information is provided in responses to RAIs 9, 21, and 29, and in ANP-3224P. The NRC staff has reviewed the licensee's response to RAIs 9, 21, and 29, and in ANP-3224P and determined that the methods are applicable to MNGP. Therefore, the NRC staff finds this disposition acceptable.

Limitation 9.16, Void Reactivity 2

> A supplement to TRACG /PANAC11 for AOO is under NRC staff review (Reference A-4). TRACG internally models the response surface for the void coefficient biases and uncertainties for known dependencies due to the relative moderator density and exposure on nodal basis. Therefore, the void history bias determined through the methods review can be incorporated into the response surface "known" bias or through changes in lattice physics/core simulator methods for establishing the instantaneous cross-sections. Including the bias in the calculations negates the need for ensuring that plant-specific applications show sufficient margin. For application of TRACG to EPU and MELLLA+ applications, the TRACG methodology must incorporate the void history bias. The manner in which this void history bias is accounted for will be established by the NRC staff SE approving NEDE- 32906P, Supplement 3, "Migration to TRACG04/PANAC11 from TRACG02/PANAC10," May 2006 (Reference A-4). This limitation applies until the new TRACG/PANAC methodology is approved by the NRC staff.

The licensee states that the limitation is not applicable because the use of COTRANSA2 has been approved in ANF-913(P)(A) and that a comparison of COTRANSA2 calculated transient response with measured data is provided in the response to RAI 29. The NRC staff finds this evaluation acceptable because this limitation is specific to the use of TRACG.

Limitation 9.17, Steady-State 5 Percent Bypass Voiding

> The instrumentation specification design bases limit the presence of bypass voiding to 5 percent (LRPM levels). Limiting the bypass voiding to less than 5 percent for long-term steady operation ensures that instrumentation is operated within the specification. For EPU and MELLLA+ operation, the bypass voiding will be evaluated on a cycle-specific basis to confirm that the void fraction remains below 5 percent at all LPRM levels when operating at steady-state conditions within the MELLLA+ upper boundary. The highest calculated bypass voiding at any LPRM level will be provided with the plant-specific SRLR.

ANP-3434P provided a response to RAI 14 in which the licensee clarified that the limitation will continue to apply and the licensee will continue to evaluate bypass voiding on a cycle-specific basis to confirm that the void fraction remains below 5 percent at all LPRM levels. Based on its review of the response to RAI 14, the NRC staff finds this disposition acceptable.

Limitation 9.18, Stability Setpoints Adjustment

> The NRC staff concludes that the presence bypass voiding at the low-flow conditions where instabilities are likely can result in calibration errors of less than

> 5 percent for OPRM cells and less than 2 percent for APRM signals. These calibration errors must be accounted for while determining the setpoints for any detect and suppress long term methodology. The calibration values for the different long-term solutions are specified in the associated sections of this SE, discussing the stability methodology.

The licensee states that the limitation is not applicable, and that the issue of bypass voiding for MNGP under EFW conditions has been addressed in ANP-3135P and found to be negligible [[ ]] at the D level LPRM. The staff has reviewed the reference and determined that the 5 percent and 2 percent penalties are not applicable to MNGP because bypass voiding has been evaluated and found to be negligible. Therefore, the NRC staff finds this disposition acceptable.

Limitation 9.19, Void Quality Correlation 1

> For applications involving PANCEA/ODYN/ISCOR/TASC for operation at EPU and MELLLA+, an additional 0.01 will be added to the OLMCPR, until such time that GE expands the experimental database supporting the Findlay-Dix void-quality correlation to demonstrate the accuracy and performance of the void-quality correlation based on experimental data representative of the current fuel designs and operating conditions during steady state, transient, and accident conditions.

The licensee states that the limitation is applicable, and that the void-quality are documented in ANP-3224P and have been reviewed and approved for MNGP for EPU. The licensee provided significant void fraction benchmark data in ANP-3135P. The staff has reviewed the references and determined that the correlations are applicable to MNGP.

Based on a review of the experimental data provided, the staff has concluded that the 0.01 OLMCPR penalty is not applicable to Monticello because sufficient experimental data has been provided to validate the void fraction correlations for AREVA ATRIUM 10XM, including void fraction levels close to100 percent. Therefore, the NRC staff finds this disposition acceptable.

Limitation 9.20, Void Quality Correlation 2

> The NRC staff is currently reviewing Supplement 3 to NEDE-32906P, “Migration to TRACG04/PANAC11 from TRACG02/PANAC10,” dated May 2006 (Reference A-4). The adequacy of the TRACG interfacial shear model qualification for application to EPU and MELLLA+ will be addressed under this review. Any conclusions specified in the NRC staff SE approving Supplement 3 to LTR NEDC- 32906P (Reference A-4) will be applicable as approved.

The licensee states that the limitation is not applicable. The NRC staff finds the disposition of this limitation acceptable because this issue is specific to TRACG.

Limitation 9.21, Mixed Core Method 1

> Plants implementing EPU or MELLLA+ with mixed fuel vendor cores will provide plant-specific justification for extension of GE's analytical methods or codes. The content of the plant-specific application will cover the topics addressed in this SE as well as subjects relevant to application of GE's methods to legacy fuel. Alternatively, GE may supplement or revise LTR NEDC-33173P (Reference A-1) for mixed core application.

The licensee states that the original limitation does not apply because Monticello implemented EPU and MELLLA+ with a full core of GE14 fuel. The use of AREVA methods for mixed cores (GE14 and AREVA ATRIUM 10XM in this case) was addressed by the NRC staff in the core transition review. The staff finds this evaluation acceptable because the mixed core evaluation was performed for the core transition. Therefore, the NRC staff finds this disposition acceptable.

Limitation 9.22, Mixed Core Method 2

> For any plant-specific applications of TGBLA06 with fuel type characteristics not covered in this review, GE needs to provide assessment data similar to that provided for the GE fuels. The Interim Methods review is applicable to all GE lattices up to GE14. Fuel lattice designs, other than GE lattices up to GE14, with the following characteristics are not covered by this review:
> • square internal water channels water crosses
> • Gd rods simultaneously adjacent to water and vanished rods
> • 11x11 lattices
>
> The acceptability of the modified epithermal slowing down models in TGBLA06 has not been demonstrated for application to these or other geometries for expanded operating domains. Significant changes in the Gd rod optical thickness will require an evaluation of the TGBLA06 radial flux and Gd depletion modeling before being applied. Increases in the lattice Gd loading that result in nodal reactivity biases beyond those previously established will require review before the GE methods may be applied.

The licensee states that the limitation is applicable, and that mixed core analyses with GE14 and AREVA ATRIUM 10XM fuel have been documented and approved previously, including low flow/low power conditions; additionally, further analyses were performed in support of the EFW LAR. The analyses found very similar thermal hydraulics characteristics between GE14 and AREVA ATRIUM 10XM fuel, and the hydraulic characteristics of GE14 and AREVA ATRIUM 10XM fuel are explicitly modeled in all the mixed core calculations. Gadolinia treatment of cross sections using CASMO4 and of thermal-mechanical issues using RODEX2 and RODEX4 has been reviewed by the staff. The NRC staff has reviewed the references and determined that the methods are applicable to MNGP. The staff has reviewed the reference and finds the disposition of this limitation acceptable.

Limitation 9.23, MELLLA + Eigenvalue Tracking

> In the first plant-specific implementation of MELLLA+, the cycle-specific eigenvalue tracking data will be evaluated and submitted to NRC to establish the performance of nuclear methods under the operation in the new operating domain. The following data will be analyzed:
> • Hot critical eigenvalue,
> • Cold critical eigenvalue,
> • Nodal power distribution (measured and calculated TIP comparison),
> • Bundle power distribution (measured and calculated TIP comparison),
> • Thermal margin,
> • Core flow and pressure drop uncertainties, and
> • The MIP Criterion (e.g., determine if core and fuel design selected is expected to produce a plant response outside the prior experience base).
>
> Provision of evaluation of the core-tracking data will provide the NRC staff with bases to establish if operation at the expanded operating domain indicates: (1) changes in the performance of nuclear methods outside the EPU experience base; (2) changes in the available thermal margins; (3) need for changes in the uncertainties and NRC approved criterion used in the SLMCPR methodology; or (4) any anomaly that may require corrective actions.

The LAR stated that this limitation is applicable. Subsequently, by letter dated September 14, 2016 (Reference 61), the licensee revised the applicability of this limitation to be not applicable. The licensee stated that this limitation is not applicable because of the low power density and low-power to flow ratio and operating experience being similar to other plants currently using AREVA methods, therefore applying Limitation 9.23 to MNGP would not provide meaningful information. The staff has reviewed the licensee's disposition and finds the disposition of this limitation acceptable.

Limitation 9.24, Plant-Specific Applications

> The plant-specific applications will provide prediction of key parameters for cycle exposures for operation at EPU (and MELLLA+ for MELLLA+ applications). The plant-specific prediction of these key parameters will be plotted against the EPU Reference Plant experience base and MELLLA+ operating experience, if available. For evaluation of the margins available in the fuel design limits, plant-specific applications will also provide quarter core map (assuming core symmetry) showing bundle power, bundle operating LHGR, and MCPR for BOC, MOC [middle-of-cycle], and EOC [end-of-cycle]. Since the minimum margins to specific limits may occur at exposures other than the traditional BOC, MOC, and EOC, the data will be provided at these exposures.

The licensee states that the limitation is applicable, and that the first plant-specific implementation of AREVA methods for EFW is being reviewed as part of the MNGP EFW LAR. Since the necessary plant-specific information was contained in the LAR the NRC staff finds this disposition acceptable.

3.6.3 Limitations from NEDC-33006P, "Maximum Extended Load Line Limit Analysis Plus"

NEDC-33006P evaluates the impact of operation in the expanded operation domain (i.e. MELLLA+) for BWRs regarding safety systems and components capabilities and performance and reponse to the design bases and special events that demonstrate plants can meet the regulatory and safety requirements. Its associated SE contains limitations. Since, for Monticello, EFW and MELLLA+ are parametrically the same, it is necessary to review the limitations associated with NEDC-33006P to ensure that any limitations that could be applicable to EFW are addressed.

This section describes the NRC staff's evaluation of the limitations in the MELLLA+ SER (NEDC-33006P).

Limitation 12.1, GEXL PLUS

> The plant-specific application will confirm that for operation within the boundary defined by the MELLLA+ upper boundary and maximum CF range, the GEXL-PLUS experimental database covers the thermal-hydraulic conditions the fuel bundles will experience, including, bundle power, mass flux, void fraction, pressure, and subcooling. If the GEXL-PLUS experimental database does not cover the within bundle thermal-hydraulic conditions, during steady state, transient conditions, and DBA conditions, GHNE will inform the NRC at the time of submittal and obtain the necessary data for the submittal of the plant-specific MELLLA+ application. In addition, the plant-specific application will confirm that the experimental pressure drop database for the pressure drop correlation covers the pressure drops anticipated in the MELLLA+ range.
>
> With subsequent fuel designs, the plant-specific applications will confirm that the database supporting the CPR correlations covers the powers, flows and void fractions BWR bundles will experience for operation at and within the MELLLA+ domain, during steady state, transient, and DBA conditions. The plant-specific submittal will also confirm that the NRC staff reviewed and approved the associated CPR correlation if the changes in the correlation are outside the GESTAR II (Amendment 22) process. Similarly, the plant-specific application will confirm that the experimental pressure drop database does cover the range of pressures the fuel bundles will experience for operation within the MELLLA+ domain

The licensee states that this limitation is applicable. AREVA's critical heat flux (CHF) correlations have well-defined ranges of applicability that have been reviewed by the staff, including the applicability to co-resident fuel. The NRC staff finds that this limitation is applicable to MNGP EFW and has been addressed adequately.

Limitation 12.2, Related LTRs

> Plant-specific MELLLA+ applications must comply with the limitations and conditions specified in and be consistent with the purpose and content covered in the NRC staff SEs approving the latest version of the following LTRs: NEDC- 33173P, NEDC-33075P, and NEDC- 33147.

The licensee states that this limitation is applicable. The licensee has reviewed the applicable limitations for GNF methods and addressed them satisfactorily. The NRC staff has reviewed the applicability of the MELLLA+ limitation and condition (L&C) and found the disposition acceptable (See Section 3.6 of this SER for the NRC staff's evaluation of the L&Cs for NEDC-33173P, NEDC-33075P, and NEDC-33147P (Reference 55)).

Limitation 12.3a, Concurrent Changes

> The plant-specific analyses supporting MELLLA+ operation will include all operating condition changes that are implemented at the plant at the time of MELLLA+ implementation. Operating condition changes include, but are not limited to, those changes that affect, an increase in the dome pressure, maximum CF, fuel cycle length, or any changes in the licensed operational enhancements. For example, with an increase in dome pressure, the following analyses must be analyzed: the ATWS analysis, the ASME overpressure analyses, the transient analyses, and the ECCS-LOCA analysis. Any changes to the safety system settings or any actuation setpoint changes necessary to operate with the increased dome pressure must be included in the evaluations (e.g., SRV setpoints).

The licensee states that this limitation is applicable. The LAR analyses comply with all operating condition changes that were implemented at MNGP in support of EPU and MELLLA+. Therefore, the NRC staff finds this disposition acceptable.

Limitation 12.3b

> For all topics in LTR NEDC-33006P that are reduced in scope or generically dispositioned, the plant-specific application will provide justification that the reduced scope or generic disposition is applicable to the plant. If changes that invalidate the LTR dispositions are to be implemented at the time of MELLLA+ implementation, the plant-specific application will provide analyses and evaluations that demonstrate the cumulative effect with MELLLA+ operation. For example, if the dome pressure is increased, the ECCS performance will be evaluated on a plant-specific basis.

The licensee states that this limitation is applicable. All fuel-related events that were included in the MNGP MELLLA+ application have been analyzed or dispositioned adequately for EFW. Therefore, the NRC staff finds this disposition acceptable.

Limitation 12.3c

> Any generic bounding sensitivity analyses provided in LTR NEDC-33006P will be evaluated to ensure that the key plant-specific input parameters and assumptions are applicable and bounded. If these generic sensitivity analyses are not applicable or additional operating condition changes affect the generic sensitivity analyses, a plant-specific evaluation will be provided. For example, with an increase in the dome pressure, the ATWS sensitivity analyses that model

> operator actions (e.g., depressurization if the HCTL is reached) needs to be reanalyzed, using the bounding dome pressure condition.

The licensee states that this limitation is applicable. Plant-specific calculations (including ATWS) have been performed using MNGP EFW conditions. Therefore, the NRC staff finds this disposition acceptable.

Limitation 12.3d

> If a new GE fuel product line or another vendor's fuel is loaded at the plant, the applicability of any generic sensitivity analyses supporting the MELLLA+ application shall be justified in the plant-specific application. If the generic sensitivity analyses cannot be demonstrated to be applicable, the analyses will be performed including the new fuel. For example, the ATWS instability analyses supporting the MELLLA+ condition are based on the GE14 fuel response. New analyses that demonstrate the ATWS instability performance of the new GE fuel or another vendor's fuel for MELLLA+ operation shall be provided to support the plant-specific application.

The licensee states that this limitation is applicable. Plant-specific calculations have been performed using MNGP EFW conditions and AREVA ATRIUM 10XM fuel. Therefore, the NRC staff finds this disposition acceptable.

Limitation 12.3e

> If a new GE fuel product line or another vendor's fuel is loaded at the plant prior to a MELLLA+ application, the analyses supporting the plant-specific MELLLA+ application will be based on a specific core configuration or bounding core conditions. Any topics that are generically dispositioned or reduced in scope in LTR NEDC-33006P will be demonstrated to be applicable, or new analyses based on the specific core configuration or bounding core conditions will be provided.

The licensee states that this limitation is applicable. Plant-specific calculations have been performed using MNGP EFW conditions and AREVA ATRIUM 10XM fuel. Therefore, the NRC staff finds this disposition acceptable.

Limitation 12.3f

> If a new GE fuel product line or another vendor's fuel is loaded at the plant prior to a MELLLA+ application, the plant-specific application will reference an NRC approved stability method supporting MELLLA+ operation, or provide sufficient plant-specific information to allow the NRC staff to review and approve the stability method supporting MELLLA+ operation. The plant-specific application will demonstrate that the analyses and evaluations supporting the stability method are applicable to the fuel loaded in the core.

The licensee states that this limitation is applicable. The licensee will use EO-III, which is an NRC approved stability method, ANP-10262(P)(A). Plant-specific calculations have been

performed using MNGP EFW conditions and AREVA ATRIUM 10XM fuel. Therefore, the NRC staff finds this disposition acceptable.

Limitation 12.3g

> For MELLLA+ operation, core instability is possible in the event a transient or plant maneuver places the reactor at a high power/low-flow condition. Therefore, plants operating at MELLLA+ conditions must have a NRC approved instability protection method. In the event the instability protection method is inoperable, the applicant must employ an NRC-approved backup instability method. The licensee will provide technical specification (TS) changes that specify the instability method operability requirements for MELLLA+ operation, including any backup stability protection methods.

The licensee states that this limitation is applicable. MNGP EFW will use the approved EO-III stability solution, including automated backup stability solution. Therefore, the NRC staff finds this disposition acceptable.

Limitation 12.4

> The plant-specific MELLLA+ application shall provide the plant-specific thermal limits assessment and transient analysis results. Considering the timing requirements to support the reload, the fuel and cycle dependent analyses including the plant-specific thermal limits assessment may be submitted by supplementing the initial M+SAR. Additionally, the SRLR for the initial MELLLA+ implementation cycle shall be submitted for NRC staff confirmation.

The licensee states that this limitation is applicable. By letter dated September 28, 2016 (Reference 62), the licensee clarified that a reload licensing report for the initial operating cycle with EFW will be submitted to the NRC when it is complete. Therefore, the NRC staff finds this disposition acceptable.

Limitation 12.5a, Operating Flexibility

> The licensee will amend the TS LCO [limiting condition for operation] for any equipment out-of-service (i.e., SLO) or operating flexibilities prohibited in the plant-specific MELLLA+ application.

The licensee states that this limitation is applicable. TSs were updated to support the EFW LAR and associated equipment out-of-service limitations. Therefore, the NRC staff finds this disposition acceptable.

Limitation 12.5b

> For an operating flexibility, such as FWHOOS, that is prohibited in the MELLLA+ plant-specific application but is not included in the TS LCO, the licensee will propose and implement a license condition.

The licensee states that this limitation is applicable. In response to SRXB RAI-4 (Reference 28), the licensee stated that due to hardware limitations the operation with FWHOOS is not possible at MNGP. Because operation with FWHOOS is not possible a TS LCO or license condition is not necessary. Therefore, the NRC staff finds this disposition acceptable.

Limitation 12.5c

> The power flow map is not specified in the TS; however, it is an important licensed operating domain. Licensees may elect to be licensed and operate the plant under plant-specific-expanded domain that is bounded by the MELLLA+ upper boundary. Plant-specific applications approved for operation within the MELLLA+ domain will include the plant-specific power/flow map specifying the licensed domain in the COLR.

The licensee states that this limitation is applicable. The power-flow operating map has been provided. Therefore, the NRC staff finds this disposition acceptable.

Limitation 12.6, SLMCPR Statepoints and CF Uncertainty

> Until such time when the SLMCPR methodology for off-rated SLMCPR calculation is approved by the staff for MELLLA+ operation, the SLMCPR will be calculated at the rated statepoint (120 percent P/100 percent CF), the plant specific minimum CF statepoint (e.g., 120 percent P/80 percent CF), and at the 100 percent OLTP at 55 percent CF statepoint. The currently approved off-rated CF uncertainty will be used for the minimum CF and 55 percent CF statepoints. The uncertainty must be consistent with the CF uncertainty.

The licensee states that this limitation is applicable. SLMCPR values have been provided at all EFW statepoint corners. Therefore, the NRC staff finds this disposition acceptable.

Limitation 12.7, Stability

> Manual operator actions are not adequate to control the consequences of instabilities when operating in the MELLLA+ domain. If the primary stability protection system is declared inoperable, a non-manual NRC approved backup protection system must be provided, or the reactor core must be operated below a NRC approved backup stability boundary specifically approved for MELLLA+ operation for the stability option employed.

The licensee states that this limitation is applicable. The EO-III solution provides for an automated backup stability solution, which is verified through STAIF calculations. Therefore, the NRC staff finds this disposition acceptable.

Limitation 12.8, Fluence Methodology and Fracture Toughness

> The applicant is to provide a plant-specific evaluation of the MELLLA+ RPV [reactor pressure vessel] fluence using the most up-to-date NRC-approved fluence methodology. This fluence will then be used to provide a plant-specific

> evaluation of the RPV fracture toughness in accordance with RG 1.99, Revision 2.

The licensee states that this limitation is applicable. However, the existing RPV fluence evaluation for the MELLLA+ LAR was used as basis for this evaluation. The licensee stated that the transition to AREVA ATRIUM 10XM fuel does not change the fast flux to the RPV; thus fluence values are not affected significantly by the change from GE14 to AREVA ATRIUM 10XM fuel. Therefore, the NRC staff finds this evaluation acceptable.

Limitation 12.9, Reactor Coolant Pressure Boundary

> MELLLA+ applicants must identify all other than Category "A" materials, as defined in NUREG-0313, Revision 2, that exist in its RCPB piping, and discuss the adequacy of the augmented inspection programs in light of the MELLLA+ operation on a plant-specific basis.

The licensee states that this limitation is applicable. The existing materials evaluation for the MELLLA+ LAR was used as basis for this evaluation. Because the operating domain for EFW is parabolicly identical to MELLLA+ the components of interest and inspection of those components is unchanged. Therefore, the NRC staff finds this disposition acceptable.

Limitation 12.10a, LOCA-Off-rated Multiplier

> The plant-specific application will provide the 10 CFR Part 50, Appendix K, and the nominal PCTs calculated at the rated EPU power/rated CF, rated EPU power/minimum CF, at the low-flow MELLLA+ boundary (Transition Statepoint). For the limiting statepoint, both the upper bound and the licensing PCT will be reported. The M+SAR will justify why the transition statepoint ECCS-LOCA response bounds the 55 percent CF statepoint. The M+SAR will provide discussion on what power/flow combination scoping calculations were performed to identify the limiting statepoints in terms of DBA-LOCA PCT response for the operation within the MELLLA+ boundary. The M+SAR will justify that the upper bound and licensing basis PCT provided is in fact the limiting PCT considering uncertainty applications to the non-limiting statepoints.

The licensee states that this limitation is applicable. LOCA calculations were performed for the maximum and minimum core flow at rated EPU power and the low core flow MELLLA+ boundary and reported in ANP-3295P. Therefore, the NRC staff finds this disposition acceptable.

Limitation 12.10b

> LOCA analysis is not performed on cycle-specific basis; therefore, the thermal limits applied in the M+SAR LOCA analysis for the 55 percent CF MELLLA+ statepoint and/or the transition statepoint must be either bounding or consistent with cycle-specific off-rated limits. The COLR and the SRLR will contain confirmation that the off-rated limits assumed in the ECCS-LOCA analyses bound the cycle-specific offrated limits calculated for the MELLLA+ operation. Every future cycle reload shall confirm that the cycle specific off-rated thermal

> limits applied at the 55 percent CF and/or the transition statepoints are consistent with those assumed in the plant-specific ECCS LOCA analyses.

The licensee states that this limitation is applicable. The LAR used a conservative OLMCPR value of 1.45 to determine the hot channel power for all the off-rated conditions. Off-rated (i.e., less conservative) OLMCPR values were not applied to establish the initial conditions for LOCA calculations, which is conservative. Therefore, the NRC staff finds this disposition acceptable.

Limitation 12.10c

> Off-rated limits will not be applied to the minimum CF statepoint.

The licensee states that this limitation is applicable. Off-rated limits were not applied to the minimum CF statepoint. Therefore, the NRC staff finds this disposition acceptable.

Limitation 12.10d

> If credit is taken for these off-rated limits, the plant will be required to apply these limits during core monitoring.

The licensee states that this limitation is applicable. Off-rated limits were not applied to the minimum CF statepoint. Therefore, the NRC staff finds this disposition acceptable.

Limitation 12.11, ECCS-LOCA Axial Power Distribution

> For MELLLA+ applications, the small and large break ECCS-LOCA analyses will include top-peaked and mid-peaked power shape in establishing the MAPLHGR and determining the PCT. This limitation is applicable to both the licensing bases PCT and the upper bound PCT. The plant-specific applications will report the limiting small and large break licensing basis and upper bound PCTs.

The licensee states that this limitation is applicable. Top- and mid-peaked power shapes were used in the plant-specific calculations reported in ANP-3295P and ANP-3092P (Reference 56). Therefore, the NRC staff finds this disposition acceptable.

Limitation 12.12a, ECCS-LOCA Reporting

> Both the nominal and Appendix K PCTs should be reported for all of the calculated statepoints, and The plant-variable and uncertainties currently applied will be used, unless the NRC staff specifically approves a different plant variable uncertainty method for application to the non-rated statepoints.

The licensee states that the AREVA methodology only calculates and reports Appendix K PCTs. The appendix K calculations are reported in ANP-3295P using the approved AREVA uncertainty methodology and the conservative OLMCPR limit (i.e., not the off-rated OLMCPR). The NRC staff finds this approach an acceptable methodology to evaluate LOCA criteria. Therefore, the NRC staff finds this disposition acceptable.

Limitation 12.12b, ECCS LOCA Reporting

> The plant-variable and uncertainties currently applied will be used, unless the NRC staff specifically approves a different plant variable uncertainty method for application to the non-rated statepoints.

The licensee states that the AREVA methodology only calculates and reports Appendix K PCTs. The appendix K calculations are reported in ANP-3295P using the approved AREVA uncertainty methodology and the conservative OLMCPR limit (i.e., not the off-rated OLMCPR). The NRC staff finds this approach an acceptable methodology to evaluate LOCA criteria. Therefore, the NRC staff finds this disposition acceptable.

Limitation 12.13, Small Break LOCA

> Small break LOCA analysis will be performed at the MELLLA+ minimum CF and the transition statepoints for those plants that: (1) are small break LOCA limited based on small break LOCA analysis performed at the rated EPU conditions; or (2) have margins of less than or equal to [[ ]] relative to the Appendix K or the licensing basis PCT.

The licensee states that this limitation is applicable. Small break analyses are performed for the maximum and minimum core flow at rated EPU power and the low core flow MELLLA+ boundary and are reported in ANP-3295P. Therefore, the NRC staff finds this disposition acceptable.

Limitation 12.14, Break Spectrum

> The scope of small break LOCA analysis for MELLLA+ operation relies upon the EPU small break LOCA analysis results. Therefore, the NRC staff concludes that for plants that will implement MELLLA+, sufficient small break sizes should be analyzed at the rated EPU power level to ensure that the peak PCT break size is identified.

The licensee states that this limitation is applicable. A large number of break sizes were evaluated at different flow rates and are reported in ANP-3295P and ANP-3211. Therefore, the NRC staff finds this disposition acceptable.

Limitation 12.15, Bypass Voiding Above the D-level

> Plant-specific MELLLA+ applications shall identify where in the MELLLA+ upper boundary the bypass voiding greater than 5 percent will occur above the D-level. The licensee shall provide in the plant-specific submittal the operator actions and procedures that will mitigate the impact of the bypass voiding on the TIPs and the core simulator used to monitor the fuel performance. The plant-specific submittal shall also provide discussion on what impact the bypass voiding greater than 5 percent will have on the NMS [neutron monitoring system] as defined in Section 5.1.1.5. The NRC staff will evaluate on plant-specific bases acceptability of bypass voiding above D level.

The licensee states that this limitation is applicable. Bypass boiling was evaluated for MNGP EFW operation and found to be [[ ]] at the D level LPRM, which is negligible and below the 5 percent acceptance criterion. Therefore, the NRC staff finds this disposition acceptable.

Limitation 12.16, Rod Withdrawal Error (RWE)

> Plants operating at the MELLLA+ operating domain shall perform RWE analyses to confirm the adequacy of the generic RBM [rod block monitor] setpoints. The M+SAR shall provide a discussion of the analyses performed and the results.

The licensee states that this limitation is applicable. The licensee performed rod withdrawal error (RWE) analyses and the results are reported in ANP-3295P. Therefore, the NRC staff finds this disposition acceptable.

Limitation 12.17, ATWS LOOP [loss of offsite power]

> As specified in LTR NEDC-33006P, at least two plant specific ATWS calculations must be performed: MSIVC and PRFO. In addition, if RHR capability is affected by LOOP, then a third plant-specific ATWS calculation must be performed that includes the reduced RHR capability. To evaluate the effect of reduced RHR capacity during LOOP, the plant-specific ATWS calculation must be performed for a sufficiently large period of time after HSBW injection is complete to guarantee that the suppression pool temperature is cooling, indicating that the RHR capacity is greater than the decay heat generation. The plant-specific application should include evaluation of the safety system performance during the long-term cooling phase, in terms of available NPSH.

The licensee states that this limitation is applicable. However, the ATWS evaluation for the MELLLA+ LAR was used as basis for this evaluation. The staff finds that the fuel transition from GE14 to AREVA ATRIUM 10XM does not impact the existing evaluation conclusions because the ATWS response (peak pressure and final containment temperature) is not affected significantly by the fuel transition. The licensee provided in ANP-3295P an evaluation of the impact that the changes in void reactivity coefficient would have on ATWS response and concluded that the changes from GE14 to AREVA ATRIUM 10XM fuel are smaller than normal changes during cycle burnup and are, thus, already accounted for in the analysis of record. The staff concurs with this evaluation because specifics of the fuel design (spacers, pressure drops, CHF correlations …) have little impact on the ATWS results, and, since RHR capability is not affected, the loss of offsite power (LOOP) case was not analyzed. Therefore, the NRC staff finds this disposition acceptable.

Limitation 12.18a, ATWS TRACG Analysis

> For plants that do not achieve hot shutdown prior to reaching the heat capacity temperature limit (HCTL) based on the licensing ODYN code calculation, plant specific MELLLA+ implementations must perform best estimate TRACG calculations on a plant-specific basis. The TRACG analysis will account for all plant parameters, including water-level control strategy and all plant-specific emergency operating procedure (EOP) actions.

The licensee states that this limitation is applicable. However, the ATWS evaluation for the MELLLA+ LAR was used as basis for this evaluation. The staff finds that the fuel transition from GE14 to AREVA ATRIUM 10XM does not impact the existing evaluation conclusions because the ATWS response is not affected significantly by the fuel transition. Therefore, the NRC staff finds this disposition acceptable.

Limitation 12.18b, ATWS TRACG Analysis

> The TRACG calculation is not required if the plant increases the boron-10 concentration/enrichment so that the integrated heat load to containment calculated by the licensing ODYN calculation does not change with respect to a reference OLTP/75 percent flow ODYN calculation.

The licensee states that this limitation is applicable. However, the ATWS evaluation for the MELLLA+ LAR was used as basis for this evaluation. The NRC staff finds that the fuel transition from GE14 to AREVA ATRIUM 10XM does not impact the existing evaluation conclusions because the ATWS response is not affected significantly by the fuel transition. Therefore, the NRC staff finds this disposition acceptable.

Limitation 12.18c, ATWS TRACG Analysis

> Peak cladding temperature (PCT) for both phases of the transient (initial overpressure and emergency depressurization) must be evaluated on a plant specific basis with the TRACG ATWS calculation.

The licensee states that this limitation is applicable. However, the ATWS evaluation for the MELLLA+ LAR was used as basis for this evaluation. The NRC staff finds that the fuel transition from GE14 to AREVA ATRIUM 10XM does not impact the existing evaluation conclusions because the ATWS response is not affected significantly by the fuel transition. Therefore, the NRC staff finds this disposition acceptable.

Limitation 12.18d, ATWS TRACG Analysis

> In general, the plant-specific application will ensure that operation in the MELLLA+ domain is consistent with the assumptions used in the ATWS analysis, including equipment out of service (e.g., FWHOOS, SLO, SRVs, SLC pumps, and RHR pumps, etc.). If assumptions are not satisfied, operation in MELLLA+ is not allowed. The SRLR will specify the prohibited flexibility options for plant-specific MELLLA+ operation, where applicable. For key input parameters, systems and engineering safety features that are important to simulating the

> ATWS analysis and are specified in the Technical Specification (TS) (e.g., SLCS parameters, ATWS RPT, etc.), the calculation assumptions must be consistent with the allowed TS values and the allowed plant configuration. If the analyses deviate from the allowed TS configuration for long term equipment out of service (i.e., beyond the TS LCO), the plant-specific application will specify and justify the deviation. In addition, the licensee must ensure that all operability requirements are met (e.g., NPSH) by equipment assumed operable in the calculations.

The licensee states that this limitation is applicable. However, the ATWS evaluation for the MELLLA+ LAR was used as basis for this evaluation. The NRC staff finds that the fuel transition from GE14 to AREVA ATRIUM 10XM does not impact the existing evaluation conclusions because the ATWS response is not affected significantly by the fuel transition. Therefore, the NRC staff finds this disposition acceptable.

Limitation 12.18e, ATWS TRACG Analysis

> Nominal input parameters can be used in the ATWS analyses provided the uncertainty treatment and selection of the values of these input parameters are consistent with the input methods used in the original GE ATWS analyses in NEDE-24222. Treatment of key input parameters in terms of uncertainties applied or plant-specific TS value used can differ from the original NEDE-24222 approach, provided the manner in which it is used yields more conservative ATWS results.

The licensee states that this limitation is applicable. However, the ATWS evaluation for the MELLLA+ LAR was used as basis for this evaluation. The NRC staff finds that the fuel transition from GE14 to AREVA ATRIUM 10XM does not impact the existing evaluation conclusions because the ATWS response is not affected significantly by the fuel transition. Therefore, the NRC staff finds this disposition acceptable.

Limitation 12.18f ATWS TRACG Analysis

> The plant-specific application will include tabulation and discussion of the key input parameters and the associated uncertainty treatment.

The licensee states that this limitation is applicable. However, the ATWS evaluation for the MELLLA+ LAR was used as basis for this evaluation. The NRC staff finds that the fuel transition from GE14 to AREVA ATRIUM 10XM does not impact the existing evaluation conclusions because the ATWS response is not affected significantly by the fuel transition. Therefore, the NRC staff finds this disposition acceptable.

Limitation 12.19 Plant Specific ATWS Instability

> Until such time that NRC approves a generic solution for ATWS instability calculations for MELLLA+ operation, each plant-specific MELLLA+ application must provide ATWS instability analysis that satisfies the ATWS acceptance criteria listed in SRP Section 15.8. The plant-specific ATWS instability calculation must: (1) be based on the peak-reactivity exposure conditions,

> (2) model the plant-specific configuration important to ATWS instability response including mixed core, if applicable, and (3) use the regional-mode nodalization scheme. In order to improve the fidelity of the analyses, the plant-specific calculations should be based on latest NRC-approved neutronic and thermal-hydraulic codes such as GBLA06/PANAC11 and TRACG04.

The licensee states that this limitation is applicable. The ATWSI analyses are documented in ANP-3284P and ANP-3435P for the mitigated TT case and the 2RPT ATWSI case. Therefore, the NRC staff finds this disposition acceptable.

Limitation 12.20 Generic ATWS Instability

> Once the generic solution is approved, the plant-specific applications must provide confirmation that the generic instability analyses are relevant and applicable to their plant. Applicability confirmation includes review of any differences in plant design or operation that will result in significantly lower stability margins during ATWS such as: turbine bypass capacity, fraction of steam-driven feedwater pumps, any changes in plant design or operation that will significantly increase core inlet subcooling during ATWS events, significant differences in radial and axial power distributions, hot-channel power-to-flow ratio, fuel design changes beyond GE14.

The licensee states that this limitation is applicable. Plant specific ATWSI analyses were performed and documented. The NRC staff concurs with this disposition based on a similar evaluation as for Limitation 12.19.

Limitation 12.21, Individual Plant Examination

> Licensees that submit a MELLLA+ application should address the plant-specific risk impacts associated with MELLLA+ implementation, consistent with approved guidance documents (e.g., NEDC-32424P-A, NEDC-32523P-A, and NEDC-33004P-A) and the Matrix 13 of RS-001 and readdress the plant-specific risk impacts consistent with the approved guidance documents that were used in their approved EPU application and Matrix 13 of RS-001. If an EPU and MELLLA+ application come to the NRC in parallel, the expectation is that the EPU submittal will have incorporated the MELLLA+ impacts.

The licensee states that this limitation is applicable. The licensee stated that changing fuel types did not impact the associated risk of operation in EFW with AREVA methods vice the parametrically identical MELLLA+ region with GEH methods. The NRC staff finds that the fuel transition from GE14 to AREVA ATRIUM 10XM does not impact the existing evaluation conclusions because the ATWS response is not affected significantly by the fuel transition. Therefore, the NRC staff finds this disposition acceptable.

Limitation 12.22 IASCC [irradiation assisted stress-corrosion cracking]

> The applicant is to provide a plant-specific IASCC evaluation when implementing MELLLA+, which includes the components that will exceed the IASCC threshold of $5x10^{20}$ n/cm2 (E>1MeV), the impact of failure of these components on the

> integrity of the reactor internals and core support structures under licensing design bases conditions, and the inspections that will be performed on components that exceed the IASCC threshold to ensure timely identification of IASCC, should it occur.

The licensee states that this limitation is applicable. The licensee evaluated the impacts of the fuel transition from GE14 to AREVA ATRIUM 10XM of IASCC and found none of significance. Therefore, the NRC staff finds this evaluation acceptable.

12.23 Limitations from the ATWS RAI Evaluations

Limitation 12.23.1

> See limitation 12.18.d.

The licensee states that this limitation is not applicable. However, the ATWS evaluation for the MELLLA+ LAR was used as basis for this evaluation. The NRC staff finds this disposition acceptable based on a similar evaluation as for Limitation 12.17.

Limitation 12.23.2

> The plant-specific ODYN and TRACG key calculation parameters must be provided to the staff so they can verify that all plant-specific automatic settings are modeled properly.

The licensee states that this limitation is applicable. Plant-specific analyses using AREVA methods have been made available to the NRC staff for review. Therefore, the NRC staff finds this disposition acceptable.

Limitation 12.23.3

> The ATWS peak pressure response would be dependent upon SRVs upper tolerances assumed in the calculations. For each individual SRV, the tolerances used in the analysis must be consistent with or bound the plant-specific SRV performance. The SRV tolerance test data would be statistically treated using the NRC's historical 95/95 approach or any new NRC-approved statistical treatment method. In the event that current EPU experience base shows propensity for valve drift higher than pre- EPU experience base, the plant-specific transient and ATWS analyses would be based on the higher tolerances or justify the reason why the propensity for the higher drift is not applicable the plant's SRVs.

The licensee states that this limitation is applicable. The ATWS evaluation for the MELLLA+ LAR was used as basis for this evaluation. The NRC staff finds this disposition acceptable based on a similar evaluation as for Limitation 12.17.

Limitation 12.23.4

> EPG [emergency procedure guidelines]/SAG [severe accident guidelines] parameters must be reviewed for applicability to MELLLA+ operation in a plant-specific basis. The plant-specific MELLLA+ application will include a section that discusses the plant-specific EOPs [emergency operating procedures] and confirms that the ATWS calculation is consistent with the operator actions.

The licensee states that this limitation is applicable. The NRC staff reviewed the ATWS calculations key parameters and found them acceptable. Therefore, the NRC staff finds this disposition acceptable.

Limitation 12.23.5

> The conclusions of this LTR and associated SE are limited to reactors operating with a power density lower than 52.5 MW/MLBM/hr for operation at the minimum allowable CF at 120 percent OLTP. Verification that reactor operation will be maintained below this analysis limit must be performed for all plant-specific applications.

The licensee states that this limitation is applicable. The MNGP power-to-flow ratio is 49.94 MWt/Mlb/h, which is less than the limitation criteria of 52.5 MWt/Mlb/h. Therefore, the NRC staff concurs with this evaluation and finds the disposition acceptable.

Limitation 12.23.6

> For MELLLA+ applications involving GE fuel types beyond GE14 or other vendor fuels, bounding ATWS Instability analysis will be provided to the staff. Note: this limitation does not apply to special test assemblies.

The licensee states that this limitation is applicable. ATWSI analyses is documented in ANP-3284P for the unmitigated turbine trip (TT) case, and ANP-3435P for the mitigated TT case and the 2RPTATWSI case. Therefore, the NRC staff finds this disposition acceptable.

Limitation 12.23.7

> Limitation 12.23.7 is similar to Limitation 12.23.6 since it requires licensees provide bounding ATSW-I analyses if MELLLA+/EFW applications involve fuel types beyond GE14 since fuel responses to ATSW-I are dependent on the assumed fuel design.

The licensee states that this limitation is applicable. ATWSI analyses is documented in ANP-3284P for the unmitigated TT case, and ANP-3435P for the mitigated TT case and the 2RPT ATWSI case. Therefore, the NRC staff finds this evaluation acceptable.

Limitation 12.23.8

> The plant-specific ATWS calculations must account for all plant- and fuel design-specific features, such as the debris filters.

The licensee states that this limitation is applicable. ATWSI calculations and ATWS peak pressure calculations accounted for all plant- and fuel-design specific features. Therefore, the NRC staff finds this disposition acceptable.

Limitation 12.23.9

> Plant-specific applications must review the safety system specifications to ensure that all of the assumptions used for the ATWS SE indeed apply to their plant-specific conditions. The NRC staff review will give special attention to crucial safety systems like HPCI, and physical limitations like NPSH and maximum vessel pressure that RCIC and HPCI can inject. The plant-specific application will include a discussion on the licensing bases of the plant in terms of NPSH and system performance. It will also include NPSH and system performance evaluation for the duration of the event.

The licensee states that this limitation is applicable. However, the ATWS evaluation for the MELLLA+ LAR was used as basis for this evaluation. The NRC staff finds this disposition acceptable based on a similar evaluation as for Limitation 12.17.

Limitation 12.23.10

> Plant-specific applications must ensure that an increase in containment pressure resulting from ATWS events with EPU/MELLLA+ operation does not affect adversely the operation of safety-grade equipment.

The licensee states that this limitation is applicable. However, the ATWS evaluation for the MELLLA+ LAR was used as basis for this evaluation. The NRC staff finds this disposition acceptable based on a similar evaluation as for Limitation 12.17.

Limitation 12.23.11

> The plant-specific applications must justify the use of plant-specific suppression pool temperature limits for the ODYN and TRACG calculations that are higher than the HCTL limit for emergency depressurization.

The licensee states that this limitation is applicable. However, the ATWS evaluation for the MELLLA+ LAR was used as basis for this evaluation. The NRC staff finds this disposition acceptable based on a similar evaluation as for Limitation 12.17.

Limitation 12.24 Limitations from Fuel-Dependent Analyses RAI Evaluations

Limitation 12.24.1

> For EPU/MELLLA+ plant-specific applications that use TRACG or any code that has the capability to model in-channel water rod flow, the supporting analysis will use the actual flow configuration.

The licensee states that this limitation is applicable. The in-channel water rod (GE14) or water channel (AREVA ATRIUM 10XM) flow will be modeled in the AREVA codes that have that capability. Therefore, the NRC staff finds this disposition acceptable.

Limitation 12.24.2

> The EPU/MELLLA+ application would provide the exit void fraction of the high-powered bundles in the comparison between the EPU/MELLLA+ and the pre-MELLLA+ conditions.

The licensee states that this limitation is applicable. Exit void fraction conditions were provided in ANP-3224P. Therefore, the NRC staff finds this disposition acceptable.

Limitation 12.24.3

> For the Fuel-Dependent Analyses RAI Evaluations (Appendix B of NEDC-33006P), the same SLMCPR limitation applies as limitation 12.6.

The licensee states that this limitation is applicable. SLMCPR values have been provided at all EFW statepoint corners. Therefore, the NRC staff finds this disposition acceptable.

Limitation 12.24.4

> For the Fuel-Dependent Analyses RAI Evaluations (Appendix B of NEDC-33006P), the same requirements as limitiation 12.18.d are applicable. This includes the same plant-specific application assumptions, SRLR requirements, and Tech Spec Requirements as limitation 12.18.d.

The licensee states that this limitation is applicable. However, the ATWS evaluation for the MELLLA+ LAR was used as basis for this evaluation. The staff finds this disposition acceptable based on a similar evaluation as for Limitation 12.17.

3.6.4 Limitations from NEDC-33075P, “General Electric Boiling Water Reactor Detect and Suppress Solution – Confirmation Density”

Limitation 4.1 HW Platform for Stability Solution

> The NRC staff has reviewed on a separate report the implementation of DSS-CD using the approved GENE Option III firmware and software and found it acceptable. Implementations on other Option III platforms will require plant-specific review.

The licensee states that this limitation is applicable to the MNGP long term stability solution EO-III, which uses the existing DSS-CD hardware currently approved for MNGP. Details of the EO-III implementation for MNGP have been provided in ANP-3295NP. Therefore, the NRC staff finds this disposition acceptable.

Limitation 4.2 Stability Solution Applicability Checklist

> Tables 6.1 and 6.2 of NEDC-33075P, Revision 5, document a plant-specific applicability checklist, which contains specific criteria that must be reviewed and satisfied for each core reload. This methodology is a technically acceptable process for plant- and cycle specific reviews of DSS-CD applicability.

The licensee states that this limitation is not applicable. EO-III plant-specific calculations are performed every cycle and were reported in ANP-3295 for the first MNGP EFW core. The staff finds this evaluation acceptable because it is specific to DSS-CD, which is not applicable to EFW.

Limitation 4.3 Stability Solution Applicability for New Fuels

> For situations where the plant applicability checklist is not satisfied (e.g., introduction of a new fuel type), Tables 6.3 and 6.4 of NEDC- 33075P, Revision 5, describe a technically acceptable procedure to extend the future applicability of DSS-CD.

The licensee states that this limitation is not applicable because EO-III relies on plant-specific calculations as opposed to the applicability checklist for DSS-CD. Because DSS-CD is not used, the NRC staff finds this disposition acceptable.

Limitation 4.4 TS Changes

> Section 8 of NEDC-33075P, Revision 5, provides a description of required changes to TSs and an example is provided in Appendix A. The proposed TSs are acceptable for the implementation of DSS-CD.

The licensee states that this limitation is applicable. TSs have been reviewed and modified for the transition from DSS-CD to EO-III. The NRC staff concurs with this evaluation. The proposed TS changes have been reviewed by the staff and found acceptable.

Limitation 4.5 Fuel Transition

> Table 6.5 of NEDC-33075P, Revision 5, describes the fuel transition scenarios, which are subject to a plant specific review for each application.

The licensee states that this limitation is not applicable because EO-III relies on plant-specific calculations as opposed to the applicability checklist for DSS-CD, and fuel transitions are handled by the cycle-specific calculations. The NRC staff finds this evaluation acceptable since the EO-III solution is re-analyzed each cycle such that it will consider a fuel transition scenario.

Limitation 4.6 Generic CDA Setpoints

> Application of an alternative to the generic CDA setpoints with respect to the susceptibility of a plant's intrinsic noise will require a plant specific review.

The licensee states that this limitation is not applicable because EO-III relies on plant-specific calculations as opposed to the applicability checklist for DSS-CD, and EO-III setpoints are handled automatically by the cycle-specific calculations. The NRC staff finds this evaluation acceptable since the setpoints are handled by the cycle-specific calculations.

Limitation 4.7 Stability HW Platform

> The hardware components required to implement DSS-CD are expected to be those currently used for the approved Solution III. If the DSS-CD hardware implementation deviates significantly from the approved Solution III, a hardware review by the NRC staff may be necessary.

The licensee states that this limitation is applicable. The hardware components used to implement EO-III are those already approved and installed for DSS-CD. Therefore, the NRC staff concurs with this evaluation and finds the disposition acceptable.

Limitation 4.8 Fixed and Adjustable Stability Parameters

> The NRC staff concludes that the plant-specific settings for eight of the FIXED parameters and three of the ADJUSTABLE parameters appear to be licensing basis values. The process by which these values will be controlled must be addressed by licensees.

The licensee states that this limitation is not applicable because it is specific to DSS-CD and EO-III relies on cycle-specific calculations. The NRC staff finds this disposition acceptable because it is only applicable when DSS-CD is used.

Limitation 4.9 Stability Solution Licensing Basis

> The NRC staff concludes that if plants other than Brunswick Steam Electric Plant, Units 1 and 2, use the DSS-CD trip function, those plant licensees must ensure the DSS-CD trip function is applicable in their plant licensing bases, including the optional BSP trip function, if it is to be installed.

The licensee states that this limitation is not applicable because it is specific to DSS-CD and EO-III relies on cycle-specific calculations. The NRC staff finds this disposition acceptable because it is only applicable when DSS-CD is used.

3.6.5 Limitations from NEDC-33147P, “DSS-CD TRACG Application”

Limitation 4.1 MCPR Uncertainty

> The NRC staff will require a submittal for review if any significant change in the bounding uncertainty or any change in the process to bound the uncertainty in the MCPR is proposed.

The licensee states that this limitation is not applicable because it is specific to the use of TRACG and DSS-CD, which are not applicable to MNGP EFW. The NRC staff concurs with this evaluation and therefore finds this disposition acceptable.

3.7 Technical Specification Changes

The licensee submitted changes to the MNGP TSs to support its EFW license amendment request in the LAR (Reference 1) and letter dated September 20, 2015 (Reference 4). The proposed TS changes are primarily associated with implementation of the AREVA EO-III stability solution, and described in ANP-10262(P)(A). The NRC staff's review of the proposed changes is discussed below.

3.7.1 TS 2.1.1, Reactor Core Safety Limits (SLs)

By letter dated September 18, 2016, the licensee submitted a proposed revision to TS 2.1.1, "Reactor Core SLs." The letter proposed revision to 2.1.1.2, addition of 2.1.1.3, and renumeration of 2.1.1.3 to 2.1.1.4 to reflect the addition of the new 2.1.1.3. The propsed TS revision addresses Limitation 9.5 related to NEDC-33173P, "Applicability of GE Methods to Expanded Operating Domains" which set forth a SLMCPR penalty of 0.03. The staff's evaluation of this Limitation is discussed in Section 3.6.2 of this SE.

The proposed TS revision to 2.1.1.2 adds a parenthetical to note that its applicability is to GEH methods. New TS 2.1.1.3 adds MCPR applicable for the range of operating domains to account for the .03 penalty proposed in Limitation 9.5 to account for the limited data in certain operating regions. The proposed TS is consistent with the staff's finding that the SLMCPR penalty should be imposed. Therefore, the NRC staff finds the proposed revisions to TS 2.1.1 accetable.

3.7.2 TS 3.3.1.1, RPS Instrumentation

The proposed changes to Conditions A and B are editorial changes to address section renumbering to account for the addition of new function 2.g to the notes of non-applicability. The NRC staff finds that these changes are editorial and nature, and, therefore acceptable.

Condition I

The proposed changes to Condition I revise Required Actions associated with the OPRM Upscale function, replacing the implementation of Manual Backup Scram Protection (BSP) Regions with an action to initiate the alternate method to detect and suppress thermal hydraulic instability oscillations.

Condition I is entered if the OPRM Upscale function (i.e., the PBDA OPRM scram) is declared inoperable. Under these conditions, the revised TSs require:

- Initiate alternative methods of stability detection within 12 hours
- Restore the channel as operable within 120 days

The alternative methods referred to in Condition I are the actions recommended by the BWROG should the original Solution III be declared inoperable. They involve stationing an operator close to the OPRM display and manually monitor for instabilities.

The NRC staff has reviewed the proposed TS changes and finds them acceptable because they are the same procedures that have been used successfully for the original Solution III, and the

deviations between Solution III and EO-III are minimal and follow the approved EO-III SER, making them acceptable.

Condition J

The proposed changes to Condition J revise the Condition, Required Actions, and Completion Time to implement the LCO associated with the proposed new EFWS trip function, removing the manual actuation of BSP Regions that was required for M+ operation.

Condition J is entered if the EFWS is inoperable. Under these conditions, the TS require:

- Exit the EFW region (i.e., reduce power below the MELLLA boundary)

The NRC staff has reviewed the proposed TS changes and finds them acceptable because they are the same procedures that have been used successfully for the original Solution III, and the deviations between Solution III and EO-III are minimal and follow the approved EO-III SER, making them acceptable.

Condition K

The proposed change to Condition K is to invoke the Required Action and Completion times when the Required Actions and Completion Times of Condition I (OPRM Upscale function) cannot be met. Condition K requires a reduction of thermal power to less than 20 percent rated thermal power within 4 hours which is identical to the required action and completion time as previously approved for operation in the MELLLA+ region.

The NRC staff reviewed the proposed TS change and finds it acceptable because derating of the plant is appropriate when there is an unrecoverable loss of the instability protection system and the completion time is consistent with the previously approved MELLLA+ amendment.

3.7.3 SR 3.3.1.1.16

The proposed TS revision adds a new surveillance requirement (SR) for the OPRM upscale function that is reinstated with the proposed amendment. The proposed SR is similar to that previously approved for MNGP prior to EFW, but with some editorial improvement. The acronym “RTP” was added to clarify that the simulated thermal power value is associated with rated thermal power, and the word “recirculation” was removed where it preceded “drive flow.”

This SR ensures that scrams initiated from OPRM upscale function will not be inadvertently bypassed when the thermal power is >25 percent and core flow <60 percent. Based on its review of the proposed SR revision, the NRC staff finds this revision acceptable.

3.7.4 Table 3.3.1.1-1

The proposed TS revision deletes note (h) and the annotation that applies the note to Function 2.b, Simulated Thermal Power- High. The note had provided reference to the COLR to determine scram setpoints under certain conditions (when the OPRM Upscale Function 2.f is inoperable).

The revision inserts a new SR for OPRM Upscale and revise the Allowable Value column to refer to the COLR. The proposed SR is identical to that previously approved for MNGP prior to EFW.

The revision inserts a new Function (Function 2.g) for EFWS trip to be implemented above the MELLLA line (in the EFW domain) and to apply all the appropriate SRs to ensure operability of the RPS instrumentation that supports the Function. The proposed revision requires the Allowable Value to be specified in the COLR.

The revision deletes note (e) to eliminate reference to DSS-CD operability, and delete the associated annotation that applies the note to Function 2.f.

The current version of the MNGP TSs is based on the implementation of a long term solution (LTS) in M+. DSS-CD is a GEH technology that is applicable under MELLLA+ licensing bases. The equivalent AREVA technology is EO-III, which is approved for use in the EFW domain. To disable DSS-CD and implement EO-III, MNGP proposes to revert to the TSs that were in place in Monticello with the original Solution III and add steps for the implementation of the additional EO-III features. Based on its review of the revisions, the NRC staff finds the proposed changes acceptable.

TS Pages 3.3.1.1-9 and 3.3.1.1-10 have been administratively revised due to requirements from previous pages (Function 4 on Page 3.3.1.1-9 and Function 8 on Page 3.3.1.1-10) rolling onto a new page because of other revisions in Table 3.3.1.1-1. These changes are administrative and do not alter the requirements of the TSs, therefore, the NRC staff finds these changes acceptable.

3.7.5 TS 3.4.1 Statement

The proposed editorial change to LCO 3.4.1 reflects the nominal change in extended flow operating domain from the “MELLLA+” terminology to the "EFW" terminology.

The NRC staff finds that the proposed terminology change is acceptable because it reflects the current licensing basis for Monticello operation in the EFW domain. After implementation of this LAR, MNGP will no longer be licensed under the MELLLA+ terminology.

3.7.6 TS 5.6.3.A.6

The proposed revisions to TS 5.6.3.A.6 would eliminate a class of COLR limits associated with the M+ amendment, and replace them with a class of limits associated with the AREVA EO-III methodology. In effect, the EO-III approach reinstates the PBDA trip setpoints that were associated with the OPRM Upscale function (Function 2.f) that was instituted prior to EFW, and includes a new COLR limit associated with new TS Table 3.3.1.1-1 reactor protection function associated with channel instability (new Function 2.g).

The prior licensing basis (MELLLA+) required the use of the GEH proprietary DSS-CD stability long term solution (LTS). By implementing EFW, MNGP removes the use of DSS-CD from TS and installs AREVA’s EO-III, which is based on the original Solution III that relies on the PBDA. To provide adequate protection under EFW conditions, EO-III institutes a CIER and associated EFWS trip. The NRC staff finds the proposed TS change is acceptable because EO-III has

been reviewed by the staff and found to be and acceptable long term stability solution in the EFW domain.

3.7.7 TS 5.6.3.B

The proposed changes to TS 5.6.3.B would remove one methodology that is not approved for use with AREVA fuel (listed as item 5.6.B.4), and add four new methodologies and one Engineering evaluation. Two of these methodologies had been used at MNGP prior to EFW and are necessary to support the transition to EO-III. The other two methodologies are previously-approved licensing topical reports associated with AREVA's EO-III methodology for the EFW domain.

The proposed TS change introduces the following changes to implement the approved EO-III LTS as the licensing bases in the EFW domain:

1. Adds NEDO-31960-A (the BWROG LTS Methodologies SER). This modification is acceptable because Monticello intends to use EO-III as the LTS in EFW, and EO-III is an extension of the old Solution III LTS, which is documented in this SER.

2. Adds NEDO-32465-A which is an integral part of EO-III methodology and its inclusion is, thus, acceptable.

3. Removes as a TS reference NEDO-33075-A (the DSS-CD SER). This modification is acceptable because DSS-CD is no longer used under EFW licensing bases and is replaced by EO-III.

4. Adds Engineering Evaluation EC 25987. This evaluation establishes the calculational framework of the setpoint methodology for the EFW Stability protection setpoints and is therefore acceptable.

5. References 8 and 9 (XN-NF-80-19(P)(A) Volume 1 and Volume 4) move from page 5.6-2 to 5.6-3. The references are unchanged and have moved pages due to modification to previous references. This change is administrative and therefore acceptable.

6. Reference 21 (ANP-10307P-A) moves from page 5.6-3 to 5.6-4. The reference is unchanged and has moved pages due to modification to previous references. This change is administrative and therefore acceptable.

7. References 22 and 24 (BAW-10255(P)(A)), which is the approved SER for AREVA's DIVOM methodology using the RAMONA5-FA code, moves from page 5.6-3 to 5.6-4. Its inclusion is acceptable because it is an integral part of the EO-III methodology.

8. Adds ANP-10262PA, which is the approved SER for EO-III and its inclusion is, thus, acceptable.

3.7.8 TS 5.6.6

The proposed change to TS 5.6.6 would eliminate a reporting requirement that was included specifically due to implementation of DSS-CD functions for the EFW amendment.

The proposed TS change removes a requirement that is specific to DSS-CD implementations and does not apply to EO-III. The proposed change is acceptable because the "OPRM" report is only a requirement of the DSS-CD SER and not of the EO-III SER. EO-III implementations have implicitly the backup stability option always armed by using the CIER function; thus an inoperable OPRM report is not required.

3.8 NRC Staff Evaluation Conclusion

The NRC staff conclusion from the review of the MNGP EFW LAR is that the use of AREVA ATRIUM 10XM fuel and AREVA methods by MNGP in the EFW operating domain with the solutions proposed in the MNGP Safety Analysis Report in NEDC-33435P and ANP-3295P are technically acceptable to satisfy the regulatory criteria. The following solutions are proposed to maintain acceptable safety margin under EFW:

1. Operation in the EFW domain will require having all SRVs except one in service. This restriction is implemented through administrative controls and is necessary to demonstrate compliance to peak vessel pressure limits during ATWS events.

2. FWHOOS will not be allowed in the EFW domain because analyses have not been performed to demonstrate compliance with applicable criteria under these conditions.

3. SLO is not allowed in the EFW domain.

4. The method for determining the LHGR letdown value and MAPLHGR will not be changed.

5. The Long Term Stability Solution OPRM amplitude set point [[ ]] will not be changed unless additional OLMCPR margin is imposed based on the AREVA calculations documented in the SAR.

6. There is no change in operator action times. Operator actions to initiate reduction of reactor vessel water level have been assumed to occur within 90 seconds of the ATWS initiation.

7. Operation in the EFW region increases the core-average void fraction and insufficient operating experience is available in this region. Therefore a 0.03 operating margin consistent with the MNGP MELLLA+ LAR is necessary.

The results of the licensee analyses (summarized in Table 5-1 of ANP-3295P, indicate that the limiting AOOs result in larger delta-CPR when initiated at nominal conditions than inside the EFW domain; therefore, additional OLPMCPR margin is not required for operation in the EFW domain.

The NRC staff concludes that the licensee adequately accounted for the effects of the proposed EFW 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 staff finds the proposed EFW operating domain operation 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 installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes SRs. The NRC staff has determined that the amendments involve 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 amendments involve no significant hazards consideration (80 FR 38775). Accordingly, the amendments meet 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 amendments.

6.0 CONCLUSION

The Commission 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 amendments will not be inimical to the common defense and security or to the health and safety of the public.

7.0 ADVISORY COMMITTEE ON REACTOR SAFEGAURDS REVIEW

During its 638th meeting which occurred on November 3-5, 2016, the Advisory Committee on Reactor Safeguards (ACRS) completed its review of the EFW license amendment request for MNGP. The Committee provided its report on the MNGP license amendment request to Chairman Stephen G. Burns in a letter dated November 15, 2016 (Reference 63).

8.0 REFERENCES

1. Fili, Karen D., "License Amendment Request for AREVA Extended Flow Window," October 3, 2014, Agencywide Documents Access and Management System (ADAMS) Accession No..

2. Fili, Karen D., "License Amendment Request for AREVA Extended Flow Window Supplement to Respond to NRC Staff Questions," January 9, 2015, ADAMS Accession No. ML15022A165.

3. Gardner, Peter A., "License Amendment Request for AREVA Extended Flow Window Supplement to Respond to NRC Staff Questions (TAC No. MF5002)," August 26, 2015, ADAMS Accession No. ML15348A222.

4. Gardner, Peter A., "License Amendment Request for AREVA Extended Flow Window Supplement to Respond to NRC Staff Questions (TAC No. MF5002)," September 29, 2015, ADAMS Accession No. ML15274A473.

5. Gardner, Peter A., "License Amendment Request for AREVA Extended Flow Window Supplement to Provide Revised Analysis of Anticipated Transient Without Scram Instability (TAC No. MF5002)," December 8, 2015, ADAMS Accession No. ML15345A407.

6. Gardner, Peter A., "License Amendment Request for AREVA Extended Flow Window Supplement to Disposition Changes to Non-Limiting Transient Analysis (TAC No. MF5002)," February 29, 2016, ADAMS Accession No. ML16063A033.

7. Gardner, Peter A., "Revised Commitment to Reconcile Analysis of Bypass Voiding For Transition to AREVA Analysis Methodology (TAC No. MF5002)," April 29, 2016, ADAMS Accession No. ML16120A299.

8. Gardner, Peter A., "License Amendment Request for AREVA Extended Flow Window Supplement to Provide Revised Analysis of Anticipated Transient Without Scram Instability (TAC No. MF5002)," August 4, 2016, ADAMS Accession No. ML16221A273.

9. Beltz, Terry A., "Issuance of Amendment No. 176 to Renewed Facility Operating License Regarding Extended Power Uprate (TAC No. MD9990)," December 9, 2013, ADAMS Accession No. ML13316C459.

10. Beltz, Terry A., "Issuance of Amendment No. 180 to Renewed Facility Operating License Regarding Maximum Extended Load Line Limit Analysis Plus (TAC No. ME3145)," March 28, 2014, ADAMS Accession No. ML14070A042.

11. ANP-3274NP, "Analytical Methods for Monticello ATWS-I," Revision 1, July 31, 2014, ADAMS Accession No. ML14283A120.

12. ANP-3284NP, "Results of Analysis & Benchmarking of Methods for Monticello ATWS-I," April 30, 2014, ADAMS Accession No. ML14283A121.

13. ANP-3295NP, “Monticello Licensing Analysis for EFW (EPU/EFW),” Revision 2, September 30, 2014, ADAMS Accession No. ML14283A122.

14. ANP-3135P, “Applicability of AREVA BWR Methods to Extended Flow Window for Monticello,” April 30, 2014, ADAMS Accession No. ML14283A197.

15. Beltz, Terry A., “Issuance of Amendment to Transition to AREVA ATRIUM 10XM Fuel and AREVA Safety Analysis Methods (TAC No. MF2479)” June 5, 2015, ADAMS Accession No. ML15072A143.

16. NEDC-33322P, “Safety Analysis Report for Monticello Constant Pressure Power Uprate,” November 5, 2008, ADAMS Accession No. ML083230125.

17. NEDC-33435P, Revision 1, “Safety Analysis Report for Monticello Maximum Extended Load Line Limit Analysis Plus,” December 2009, ADAMS Accession No. ML100280566.

18. ANP-3295P, “Monticello Licensing Analysis for EFW (EPU/EFW),” Revision 2, September 30, 2014, ADAMS Accession No. ML14283A196.

19. NUREG-0800 “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, LWR Edition,” March 2007, ADAMS Accession No. ML070660036

20. Review Standard RS-001 “Review Standard for Extended Power Uprates,” December 2003, ADAMS Accession No. ML033640024.

21. ANP-3213P, Rev.2 Monticello Fuel Transition Cycle 28 Reload Licensing Analysis (EPU/MELLLA), AREVA NP June 2013.

22. ANP-3158P, Revision 2, Fuel Rod Thermal-Mechanical Design for Monticello ATRIUM IOXM Fuel Assemblies, Cycle 28, AREVA NP, August 2013.

23. EMF-2209(P)(A) Revision 3, “SPCB Critical Power Correlation,” September 2009.

24. ANP-10298PA, Supplement 1P, Revision 0, “Improved K-factor Model for ACE/ATRIUM 10XM Critical Power Correlation,” AREVA NP, December 2011.

25. ANP-3224P, Rev 2, “Applicability of AREVA NP BWR Methods to Monticello,” June 2013 ADAMS Accession No. ML13200A201.

26. EMF-2158(P)(A) Rev 0, “Siemens Power Corporation Methodology for Boiling Water Reactors: Evaluation and Validation of CASMO-4/MICROBURN-B2, Siemens Power Corporation,” October 1999.

27. ANP-10307PA Rev 0, “AREVA MCPR Safety Limit Methodology for Boiling Water Reactors, AREVA NP,” June 2011.

28. ANP-3434P Rev 1, “AREVA Responses to RAI from SRXB and SNPB on MNGP EFW LAR,” August 2015.

29. NEDO 31960-A, Supplement 1, “BWR Owners’ Group Long-Term stability Solutions Licensing Methodology (Supplement 1), GE Nuclear Energy, November 1995, ADAMS Accession No. ML14093A211.

30. NEDO 32465-A, “Licensing Topical Report, Reactor Stability Detect and Suppress Solutions Licensing Basis Methodology for Reload Applications,” GE Nuclear Energy, August 1996, ADAMS Accession No. ML14093A210.

31. ANP-10262PA Rev 0, "Enhanced Option III Long Term Stability Solution," AREVA NP, May 2008, ADAMS Accession No. ML081270199.

32. NEDC 33075P, Revision 6, “Detect and Suppress Solution -Confirmation Density Licensing Topical Report,” dated January 2008, ADAMS Accession No. ML080310384.

33. ANP-10262Q1P, Revision 000, “Response to Request for Additional Information – ANP-10262(P),” August 2007, ADAMS Accession No. ML072280224.

34. ANF-913(P)(A), Volume 1, Revision 1, and Volume 1, Supplements 2, 3 and 4, COTRANSA2: A Computer Program for Boiling Water Reactor Transient Analyses, Advanced Nuclear Fuels Corporation, August 1990.

35. XN-NF-80-19(P)(A), Volume 3, Revision 2, “Exxon Nuclear Methodology for Boiling Water Reactors, THERMEX: Thermal Limits Methodology Summary Description,” Exxon Nuclear Company, January 1987.

36. XN-NF-84-105(P)(A), Volume 1, and Volume 1, Supplements 1 and 2, “XCOBRA-T: A Computer Code for BWR Transient Thermal-Hydraulic Core Analysis,” Exxon Nuclear Company, February 1987.

37. XN-NF-81-58(P)(A,) Revision 2 and Supplements 1 and 2, “RODEX2 Fuel Rod Thermal-Mechanical Response Evaluation Model,” Exxon Nuclear Company, March 1984.

38. NEDC-33006PA, Revision 3, “General Electric Boiling Water Reactor Maximum Extended Load Line Limit Analysis Plus,” GE-Hitachi. June 2009.

39. ANP-3211, Revision 1, “Monticello EPU LOCA Break Spectrum Analysis for ATRIUM™ 10XM Fuel,” July 2013.

40. ANP-3212P Revision 0 “Monticello EPU LOCA-ECCS Analysis MAPLHGR Limits for ATRIUM (TM) 10XM Fuel,” May 2013.

41. EMF-2361 (P)(A) Revision 0, “EXEM BWR-2000 ECCS Evaluation Model,” Framatome ANP, May 2001.

42. Letter, M.A. Schimmel (NSPM) to Document Control Desk (NRC), “License Amendment Request for Fuel Storage Changes,” L-MT-12-076, October 30, 2012, ADAMS Accession No. ML12307A433.

43. ANP-3376NP, “Supplement to Xcel Energy License Amendment Request for AREVA Extended Flow Window, Revision 0,” December 2014, ML15022A167.

44. L. Mark Padovan, “Issuance of Amendment RE: Elimination of Requirements for Hydrogen Recombiners and Hydrogen and Oxygen Monitors (TAC No. MC1902)” May 21, 2004, ADAMS Accession No. ML041180612.

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

46. SRM on SECY 93-087, “Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced LWR Designs,” July 21, 1993, ADAMS Accession No. ML003708056.

47. Engineering Evaluation EC 25987, “Calculation Framework for the Extended Flow Window Stability (EFWS) Setpoints, August 18, 2015, ADAMS Accession No. ML15274A474.

48. Schimmel, Mark, License Amendment Request (LAR) to Adopt the Alternate Source Term Methodology,” October 27, 2009, ADAMS Accession No. ML093160605.

49. Regulatory Guide 1.183, “Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors,” July 2000, ADAMS Accession No. ML003716792.

50. ANP-10262Q2P, Revision 000, “Response to Request for Additional Information – ANP-10262(P), April 2008, ADAMS Accession No. ML080990064.

51. BAW-10255PA, Revision 2, “Cycle-Specific DIVOM Methodology Using the RAMONA5-FA Code,” AREVA NP, May 2008.

52. NEDC-33173P, “Applicability of GE Methods to Expanded Operating Domains,” dated July 21, 2009, ADAMS Accession No. ML083520464.

53. EMF-2245(P)(A), Revision 0, “Application of Siemens Power Corporation's Critical Power Correlations to Co-Resident Fuel,” August 2000.

54. BAW-10247PA, Revision 0, “Realistic Thermal-Mechanical Fuel Rod Methodology for Boiling Water Reactors,” AREVA NP, February 2008.

55. NEDC-33147P, Revision 2, “DSS-CD TRACG Application,” dated November 2007, ADAMS Accession No. ML073120638

56. ANP-3092(P), Revision 0, “Monticello Thermal-Hydraulic Design Report for ATRIUM™ 10XM Fuel Assemblies,” July 2012.

57. ANP-2637, Revision 6, “Boiling Water Reactor Licensing Methodology Compendium,” October 2014.

58. EMF-CC-074(P)(A), Volume 4, Revision 0, “BWR Stability Analysis - Assessment of STAIF with Input from MICROBURN-B2,” Siemens Power Corporation, August 2000.

59. XN-NF-85-92(P)(A), "Exxon Nuclear Uranium Dioxide/Gadolinia Irradiation Examination and Thermal Conductivity Results," Exxon Nuclear Company, November 1986.

60. SSP-09/444-C, “Monticello NPP Power Distribution Uncertainties,” July 2009, ADAMS Accession No. ML110450403.

61. Gardner, Peter A., “License Amendment Request for AREVA Extended Flow Window Supplement to Address Power Distribution Uncertainties (TAC No. MF5002),” September 14, 2016, ADAMS Accession No. ML16258A480.

62. Gardner, Peter A., “License Amendment Request for AREVA Extended Flow Window Supplement to Revise the Applicability of a Topical Report Limitation (TAC No. MF5002),” September 28, 2016, ADAMS Accession No. ML16272A328.

63. Bley, Dennis C.,” Monticello Nuclear Generating Plant Licensing Amendment Request For Operation in the Extended Flow Window Domain,” November 15, 2016, ADAMS Accession No. ML16312A407.

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Appendices:
A. AREVA Codes Used for Monticello EFW LAR and Code Evaluation for EFW Applicability
B. Request for Additional Information Evaluation
C. List of Acronyms

APPENDIX A

AREVA CODES USED FOR MONTICELLO EXTENDED FLOW WINDOW (EFW) LICENSE AMENDMENT REQUEST (LAR) AND CODE EVALUATION FOR EFW APPLICABILITY

The Monticello Nuclear Generating Plant (MNGP)-specific AREVA EFW methodology LAR is a first-of-a kind review for the NRC staff since the AREVA analyses methodology are not explicitly approved for EFW (or MELLLA+) conditions. The staff with technical support from Oak Ridge National Laboratory (ORNL) verified the validity of application of AREVA methods for EFW analyses for MNGP. A complete list of methodology and evaluation models for cycle-specific reload analyses is given in ANP-3295, Revision 2, "Monticello Licensing Analysis for EFW (EPU/MELLLA+)." The review examined whether the evaluations, models, and codes listed in ANP-3295 Table 2-3 are qualified and benchmarked for application at high void fractions and low flow conditions expected during EFW operating conditions for MNGP. ANP-3135P, Revision 0, "Applicability of AREVA BWR Methods to Extended Flow Window for Monticello," reviews the AREVA licensing methodologies to demonstrate that they are applicable to operation of the MNGP including EPU conditions as well as the EFW. The applicability of the AREVA licensing methodologies for EPU at MNGP was addressed in ANP-3224P, Revision 2, "Applicability of AREVA NP BWR Methods to Monticello." The staff's evaluation of the applicability of these AREVA codes and methodologies for EFW is described in this Appendix.

<u>Summary of AREVA Codes</u>

In the response to RAI-27 (Reference 28), the licensee listed all the codes used for the MNGP EFW evaluation. ANP-2637 (Reference 57) contains a compendium of AREVA methods for BWRs and their roles in the safety analysis methodology. The following is a list of the approved codes:

1. CASMO4/MICROBURN-B2 is the approved EMF-2158(P)(A) steady state core simulator. CASMO4 generates the lattice cross sections as function of instantaneous void and temperature and the histories. MICROBURN-B2 performs 3D neutronic calculations and couples them to the thermal-hydraulic (TH) solution.

2. SAFLIM3D is the approved code for AREVA safety limit methodology for BWRs.

3. XCOBRA is the steady state detailed thermal-hydraulic analysis code. Note that XCOBRA has not been explicitly approved by the staff, but its use has been found to be acceptable in the context of the THERMEX thermal limits methodology, XN-NF-80-19(P)(A).

4. XCOBRA-T is the approved XN-NF-84-105(P)(A) transient thermal hydraulic analysis code. It performs analyses of transient heat transfer behavior in BWR assemblies.

5. COTRANSA2 is the approved transient coupled neutronic thermal-hydraulics code used for transient analyses, including AOOs and is described in ANF-913(P)(A). The COTRANSA2 code is used to calculate BWR system behavior for steady-state and transient conditions. This behavior is then used to provide input to the XCOBRA-T and XCOBRA codes, from which critical power ratios are determined for limiting transients.

6. STAIF is the approved (EMF-CC-074 (P)(A) (Reference 58)) frequency domain stability code, used for exclusion region calculations.

7. RAMONA5-FA is the approved (BAW -10255(P)(A) Revision 2) code for calculating DIVOM correlation parameters to define EO-III scram setpoints.

8. RODEX2 is the approved (XN-NF-81-58(P)(A)) code for thermal-mechanical fuel performance. It is used mainly to generate input parameters (e.g. fuel gap conductance) for the transient codes such as COTRANSA2 and for LOCA calculations. See response to RAI-13 (Reference 28) for complete details.

9. RODEX4 is the approved code (BAW-10247PA) for thermal-mechanical fuel performance of BWR fuel designs. RODEX4 is used in the thermal-mechanical licensing and safety calculations for normal operation and AOOs to demonstrate compliance with the 1 percent strain increment and centerline melting criteria. See response to RAI-13 for complete details.

10. RELAX is the approved code (EMF-2361(P)(A), (Reference 43)) code that calculates the system and hot channel blowdown transient. It is part of the EXEM/BWR ECCS evaluation suite of codes.

11. HUXY is the approved code (EMF-2361(P)(A)) that takes input from the RELAX system calculation results and computes the fuel heatup of the maximum power assembly at the plane of interest over the entire LOCA transient. It is part of the EXEM/BWR ECCS evaluation suite of codes, and it is used to develop a planar heat transfer model including rod-to-rod radiation.

In addition to the above approved codes, two codes are used for the ATWSI calculation. These codes are referenced in the Monticello EFW LAR and are applicable in the EFW domain.

1. AISHA described in ANP-3274P and ANP-3284P is a coupled neutronics TH code used to calculate the transient response [[ ]] It is used primarily for stability calculations during ATWS-I events.

2. SINANO described in ANP-3274P and ANP-3284P is a single channel detailed TH model that uses boundary conditions from AISHA to calculate the fuel response.

<u>CASMO4/MICROBURN-B2</u>

CASMO4/MICROBURN-B2 is the AREVA steady state core simulator. CASMO4 generates the lattice cross sections as function of instantaneous void and temperature and the histories. MICROBURN-B2 performs 3D neutronic calculations and couples them to the TH solution. EMF-2158(P)(A), which includes Topical Report and the approving safety evaluation report (SER), has the following limitations, which are implemented by AREVA as engineering guidelines:

1. The CASMO-4/MICROBURN-B2 code systems shall be applied in a manner that predicted results are within the range of the validation criteria (Tables 2.1 and 2.2) and measurement uncertainties (Table 2.3) presented in EMF-2158(P)(A).

2. The CASMO-4/MICROBURN-B2 code system shall be validated for analyses of any new fuel design which departs from current orthogonal lattice designs and/or exceed gadolinia and U-235 enrichment limits.

3. The CASMO-4/MICROBURN-B2 code system shall only be used for BWR licensing analyses and BWR core monitoring applications.

4. The review of the CASMO-4/MICROBURN-B2 code system should not be construed as a generic review of the CASMO-4 or MICROBURN-B2 computer codes.

5. The CASMO-4/MICROBURN-B2 code system is approved as a replacement for the CASMO-3G/MICROBURN-B code system used in NRC-approved AREVA BWR licensing methodology and in AREVA BWR core monitoring applications. Such replacements shall be evaluated to ensure that each affected methodology continues to comply with its SER restrictions and/or conditions.

6. AREVA shall notify any customer who proposes to use the CASMO-4/MICROBURN-B2 code system independent of any AREVA fuel contract that conditions 1 through 4 above must be met. AREVA's notification shall provide positive evidence to the NRC that each customer has been informed by AREVA of the applicable conditions for using the code system.

The NRC staff has reviewed the applicable limitations and finds that operation in the EFW regime does not invalidate any of the limitations.

As discussed earlier in this SER, EFW operating conditions at MNGP are bounded in terms of void fraction, power, and flow by other reactors in the fleet where the use of CASMO4/MICROBURN-B2 is currently approved and demonstrate good benchmarks against plant data.

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Therefore, the NRC staff concludes that the use of CASMO4/MICROBURN-B2 for MNGP in the EFW domain is an acceptable extension of the existing approval.

SAFLIM3D

SAFLIM3D is the code used by AREVA safety limit methodology for BWRs. The SLMCPR methodology is determined using a statistical analysis that employs a Monte Carlo process that perturbs key input parameters used in the MCPR calculation. The Monte Carlo process is implemented by the SAFLIM3D code, which was approved in ANP-10307PA for referencing without limitations.

SAFLIM3D uses a Monte Carlo approach to sampling the number of rods that are in boiling transition, and it is used to define the SLMCPR. Monticello operation in the EFW domain does not impact the process; therefore, the staff concludes that the use of SAFLIM3D in the Monticello EFW domain is an acceptable extension of the existing approval.

XCOBRA

XCOBRA is the steady state detailed thermal-hydraulic analysis code. Note that XCOBRA has not been explicitly approved by the staff, but its use has been found to be acceptable in the context of the THERMEX thermal limits methodology, XN-NF-80-19PA. The only limitation from that evaluation, which is not applicable to the use of XCOBRA, is:

- Monitoring systems other than POWERPLEX® CMSS may be used provided that the associated power distribution uncertainties are identified and appropriate operating parameters compatible with ENC transient safety analyses are monitored. Whatever monitoring system is used should be specifically identified in plant submittals.

AREVA notes that some of the computer codes referenced in the topical report have been superseded by other NRC-approved codes (e.g., COTRANSA with COTRANSA2, XTGBWR with MICROBURN-B2) and the XN-3 CHF correlation has been supplemented with the NRC-approved SPCB and ACE CHF correlations.

As discussed earlier in this SER, EFW operating conditions for MNGP are bounded in terms of void fraction, power, and flow by other reactors in the fleet where the use of XCOBRA is currently approved and demonstrate good benchmarks against plant data.
Therefore, the NRC staff concludes that the use of XCOBRA in the MNGP EFW domain is an acceptable extension of the existing approval.

XCOBRA-T

XCOBRA-T is the transient thermal hydraulic analysis code. It performs analyses of transient heat transfer behavior in BWR assemblies.

XN-NF-84-105(P)(A) which contains the Topical Report and the approval SER, contains the following limitations, which are enforced by AREVA through engineering guidelines:

1. XCOBRA-T was found acceptable for the analysis of only the following licensing basis transients:
    a. Load rejection without bypass
    b. Turbine trip without bypass
    c. Feedwater controller failure
    d. Steam isolation valve closure without direct scram
    e. Loss of feedwater heating or inadvertent HPCI actuation
    f. Flow increase transients from low-power and low-flow operation

2. XCOBRA-T analyses that result in any calculated downflow in the bypass region will not be considered valid for licensing purposes.

3. XCOBRA-T licensing calculations use NRC approved default options for void quality relationship and two-phase multiplier correlations.

4. The use of XCOBRA-T is conditional upon a commitment by ENC to a follow-up program to examine the XCOBRA-T void profile against experimental data from other sources.

As discussed in this SER EFW operating conditions in MNGP are bounded in terms of void fraction, power, and flow by other reactors in the fleet where the use of XCOBRA-T is currently approved and demonstrate good benchmarks against plant data.

AREVA has performed void fraction measurements to specifically assess the impact of the AREVA ATRIUM 10XM fuel design attributes on void fraction predictions by AREVA codes. These were performed at the KATHY test facility using two prototypical BWR test assemblies (including AREVA ATRIUM 10XM) with part-length rods and mixing vane spacer grids. In addition, AREVA has used reference void fraction data from FRIGG-2 and FRIGG-3. These data are summarized in ANP-3224. XCOBRA-T uses the Ohkawa-Lahey correlation, which is not as accurate as the [[ ]] results, and it shows a small negative bias (~-0.05) for void fractions between 40 percent and 80 percent, with unbiased agreement for $\alpha<40$ percent and $\alpha>80$ percent. However, ANP-3224 documents a sensitivity study where the void-quality correlation is biased up and down (±0.05 void) based on the scatter observed between calculation and measurements. The sensitivity results indicate that biasing the void-quality correlations results in a very small increase in ΔCPR, a very small decrease in SLMCPR, and a

very small increase in OLMCPR for this study; therefore, the impact of the correlation bias is small.

Operation in EFW does not impact significantly whether a transient will have downflow in the bypass region, which requires a very low core flow rate. Thus, the applicability of XCOBRA-T in the EFW domain is not likely to be impacted by reverse bypass flow.

Operation in the EFW domain increases the core average void fraction, but Monticello EFW conditions are bounded by experience if other operating plants that use approved AREVA methods successfully. In addition, AREVA has demonstrated little sensitivity to void-quality correlation biases.

Therefore, the NRC staff concludes that the use of XCOBRA-T in the MNGP EFW domain is an acceptable extension of the existing approval.

COTRANSA2

COTRANSA2 is the transient coupled neutronic TH's code used for transient analyses, including AOOs. The COTRANSA2 code is used to calculate BWR system behavior for steady-state and transient conditions. This behavior is then used to provide input to the XCOBRA-T and XCOBRA codes, from which critical power ratios are determined for limiting transients.

The approval SER for ANF-913(P)(A), which is included with the Topical Report in ANF-913(P)(A), contains the following limitations, which are implemented by engineering guidelines and automation tools:

1. Use of COTRANSA2 is subject to limitations set forth for methodologies described and approved for XCOBRA-T and COTRAN.

2. The COTRANSA2 code is not applicable to the analysis of any transient for which lateral flow in a bundle is significant and non-conservative in the calculation of system response.

3. For those analyses in which core bypass is modeled, the effect of a computed negative flow in the core bypass region should be shown to make no significant non-conservative contribution in the system response.

4. Licensing applications referencing the COTRANSA2 methodology must include confirmation that sensitivity to the time step selection has been considered in the analysis.

COTRANSA2 is approved for the following Chapter 15 analysis:

15.1.1 – 15.1.3 Decrease in Feedwater Temperature, Increase in Feedwater Flow, and Increase in Steam Demand

15.2.1 – 15.2.5 Loss of External Load, Turbine Trip, Loss of Condenser Vacuum, Closure of Main Steam Isolation Valve (BWR), and Steam Pressure Regulator Failure (Closed)

15.2.7	Loss of Normal Feedwater Flow 15.3.1-15.3.2 Loss of Forced Reactor Coolant Flow Including Trip of Pump Motor and Flow Controller Malfunctions
15.3.3-15.3.4	Reactor Coolant Pump Rotor Seizure and Reactor Coolant Pump Shaft Break
15.4.4 – 15.4.5	Startup of an Inactive Loop or Recirculation Loop at an Incorrect Temperature, and Flow Controller Malfunction Causing an Increase in BWR Core Flow Rate
15.5.1	Inadvertent Operation of ECCS that Increases Reactor Coolant Inventory
15.6.1	Inadvertent Opening of a PWR Pressure Relief Valve and BWR Pressure Relief Valve
15.8	ATWS (the Initial Pressurization Only)

The COTRANSA2 SER restrictions are similar to those for XCOBRA-T, and a similar evaluation applies. AREVA has provided void data for up to 100 percent void and a sensitivity analysis showing little sensitivity to void bias. Other reactors in the fleet bound the EFW operating conditions in MNGP. And none of the limitations in the original COTRANSA2 SER are violated by MNGP operation in the EFW domain.

Therefore, the NRC staff concludes that the use of COTRANSA2 in the MNGP EFW domain is an acceptable extension of the existing approval.

STAIF

STAIF, EMF-CC-074 (P)(A), is the frequency domain stability code, used for exclusion region calculations. The approval SER contains, is found with the Topical Report in EMF-CC-074 (P)(A), contains the following limitations:

1. The core model must be divided into a minimum of 24 axial nodes.

2. The core model must be divided into a series of radial nodes (i.e., thermal-hydraulic regions or channels) in such a manner that:
    a. No single region can be associated with more than 20 percent of the total core power generation. This requirement guarantees a good description of the radial power shape, especially for the high power channels.
    b. The core model must include a minimum of three regions for every bundle type that accounts for significant power generation.
    c. The model must include a hot channel for each significant bundle type with the actual conditions of the hot channel.

3. Each of the TH regions must have its own axial power shape to account for 3-D power distributions. For example, high power channels are likely to have more bottom peaked shapes.

4. The collapsed 1-D cross sections must represent the actual conditions being analyzed as closely as possible, including control rod positions.

5. The STAIF calculation must use the "shifted Nyquist" or complex pole search feature to minimize the error at low DR conditions.

The limitations are implemented in the code itself when it collects data from MICROBURN-B2 and collapses it for calculation. STAIF is used primarily to calculate the exclusion regions that define the EFWS channel instability region and the backup stability solutions. For all plants, and specifically for Monticello, these exclusion regions are located at low flow conditions outside the EFW region. In addition, operation in the EFW region does not violate any of the SER limitations. Therefore, operation in EFW does not affect the applicability of STAIF for its intended purpose.

RAMONA5-FA

RAMONA5-FA is the code for calculating DIVOM correlation parameters to define EO-III scram setpoints.

The approving SER, which is contained with the Topical Report in BAW-10255PA, Revision 2, has the following limitations, which are implemented by AREVA as engineering guidelines:

1. If a reduced scope of parameter variations is used to define the cycle-specific DIVOM slope as described in Section 7 of the TR, the scope must be justified and documented for NRC staff review.

2. The NRC staff imposes a condition to perform a full code review of RAMONA5-FA, including constitutive relations, numerics, neutronic methods, and benchmarks before RAMONA5-FA can be used to calculate DIVOM curves in EFW operating domains without the 10 percent penalty on DIVOM slopes, as noted in Limitation and Condition No. 3 below.

3. The NRC staff imposed an interim 10 percent penalty on DIVOM slopes calculated using the RAMONA5-FA methodology under EFW conditions. This was an interim restriction that is revised because the full RAMONA5-FA Code review is complete.

RAMONA5-FA is used to calculate the DIVOM slope for EO-III setpoint calculations. Thus, even though the solution involves the whole core, the application is for a single hot channel where the power and CPR oscillation amplitudes are correlated. Thus, RAMONA5-FA DIVOM calculations are not affected to first order by operation in the EFW domain because the hot channel conditions are limited by its own CPR limits and are likely to be similar inside and outside the EFW domain.

In addition, as with XCOBRA-T and COTRANSA2 above, a similar evaluation applies. AREVA has provided void data for up to 100 percent void and a sensitivity analysis showing little sensitivity to void bias. Other reactors in the fleet bound the EFW operating conditions in Monticello. And none of the limitations in the original RAMONA5-FA SER are violated by Monticello operation in the EFW domain.

Therefore, the NRC staff concludes that the use of RAMONA5-FA to calculate DIVOM parameters in the MNGP EFW domain is an acceptable extension of the existing approval.

RODEX2

RODEX2 is the code for thermal-mechanical fuel performance. It is used mainly to generate input parameters (e.g., fuel gap conductance) for the transient codes such as COTRANSA2 and for LOCA calculations. See response to RAI-13 and ANP-2637 for complete details.
The approving SER, which is contained with the Topical Report in XN-NF-81-58(P)(A,) Revision 2 and Supplements 1 and 2, has the following limitations, which are implemented by AREVA as engineering guidelines and computer code controls:

1. The NRC concluded that the RODEX2 fission gas release model was acceptable to burnups up to 60 MWd/KgU. This implies a burnup limit of 60 MWd/KgU (nodal basis). (This restriction no longer applies. The exposure limits for BWR fuel were increased to 54 MWd/kgU for an assembly and to 62 MWd/kgU for a rod).

2. The creep correlation accepted by the NRC is the one with the designation MTYPE = 0.

XN-NF-85-92(P)(A) (Reference 59) justifies Gd fuel properties for up to 8 wt percent Gd with RODEX2 methods, which covers the expected operation of Monticello in EFW.

Operation in the EFW domain increases the core-average void fraction, but does not change the operating power; thus, fuel rod temperatures and conditions are similar. Therefore, operation in EFW is not expected to impact the validity of the RODEX2 models.

Therefore, the NRC staff concludes that the use of RODEX2 in the MNGP EFW domain is an acceptable extension of the existing approval.

RODEX4

RODEX4 is the code for thermal-mechanical fuel performance of BWR fuel designs. RODEX4 is used in the thermal-mechanical licensing and safety calculations for normal operation and AOOs to demonstrate compliance with the 1 percent strain increment and centerline melting criteria. See response to RAI-13 and ANP-2637 for complete details.

RODEX4 is approved for modeling BWR fuel rods with the following conditions:

1. Peak rod average burnup limit of 62 GWd/MTU (full length rod).

2. Solid UO_2 fuel pellet with a maximum Gd content of 10.0 weight percent.

3. CWSR Zr-2 fuel clad material

The approving SER, which is contained with the Topical Report in BAW-10247PA, has the following limitations:

1. Due to limitations within the FGR model, the analytical fuel pellet grain size shall not exceed 20 microns 3-D when the as-manufactured fuel pellet grain size could exceed 20 microns 3-D.

2. RODEX4 shall not be used to model fuel above incipient fuel melting temperatures.

3. The hydrogen pickup model within RODEX4 is not approved for use.

4. Due to the empirical nature of the RODEX4 calibration and validation process, the specific values of the equation constants and tuning parameters derived in TR BAW-10247(P), Revision 0 (as updated by RAI responses) become inherently part of the approved models. Thus, these values may not be updated without necessitating further NRC review.

5. RODEX4 has no crud deposition model. Due to the potential impact of crud formation on heat transfer, fuel temperature, and related calculations, RODEX4 calculations must account for a design basis crud thickness. The level of deposited crud on the fuel rod surface should be based upon an upper bound of expected crud and may be based on plant-specific history. Specific analyses would be required if an abnormal crud or corrosion layer (beyond the design basis) is observed at any given plant. For the purpose of this evaluation, an abnormal crud/corrosion layer is defined by a formation that increases the calculated fuel average temperature by more than 25°C beyond the design basis calculation.

The implementation of these limitations by AREVA is documented in ANP-2637. The NRC staff has reviewed these limitations and concludes that they will be satisfied in the EFW domain in Monticello without changes.

As with the RODEX2 evaluation above, operation in the EFW domain increases the core-average void fraction, but does not change the operating power; thus fuel rod temperatures and conditions are similar. Therefore, operation in EFW is not expected to impact the validity of the RODEX4 models.

Therefore, the NRC staff concludes that the use of RODEX4 in the MNGP EFW domain is an acceptable extension of the existing approval.

<u>RELAX</u>

RELAX is the code that calculates the system and hot channel blowdown transient. It is part of the EXEM/BWR ECCS evaluation suite of codes.

The approving SER, which is contained with the Topical Report in EMF-2361(P)(A), has only one limitation that is no longer applicable because the FLEX code is no longer used:

- Counter-current flow limit correlation coefficients used in FLEX for new fuel designs that vary from fuel cooling test facility (FCTF) measured test configurations must be justified

LOCA results are mostly driven by decay heat, which is proportional to operating power, and not affected significantly by EFW operation. Thus, the staff concludes that the use of RELAX in the Monticello EFW domain is an acceptable extension of the existing approval.

HUXY

HUXY is a code that takes input from the RELAX system calculation results and computes the fuel heatup of the maximum power assembly at the plane of interest over the entire LOCA transient. It is part of the EXEM/BWR ECCS evaluation suite of codes, and it is used to develop a planar heat transfer model including rod-to-rod radiation.

The approving SER in XN-CC-33A has the following limitations, which are implemented by AREVA as engineering guidelines and code modifications.

1. The NRC staff, however, will require that a conservative reduction of 10 percent be made in the (spray heat transfer) coefficients specified in 10 CFR 50 Appendix K for 7x7 assemblies when applied to ENC 8x8 assemblies.

2. In each individual plant submittal employing the Exxon model the applicant will be required to properly take rod bowing in account.

3. Since GAPEX is not identical to HUXY in radial nodding or solution scheme, it is required that the volumetric average fuel temperature for each rod be equal to or greater than that in the approved version of GAPEX. If it is not, the gap coefficient must be adjusted accordingly.

4. It has been demonstrated that the (2DQ local quench velocity) correlation gives hot plane quench time results that are suitably conservative with respect to the available data when a coefficient behind the quench front of 14000 Btu/(hr-ft2-OF) is used.

5. It (Appendix K) requires that heat production from the decay of fission products shall be 1.2 times the value given by K. Shure as presented in ANS 5.1 and shall assume infinite operation time for the reactor.

6. It is to be assumed for all these heat sources (fission heat, decay of actinides and fission product decay) that the reactor has operated continuously at 102 percent of licensed power at maximum peaking factors allowed by the TSs.

7. For small and intermediate size breaks, the applicability of the fission power curve used in the calculations will be justified on a case by case basis. This will include justification of the time of scram (beginning point in time of the fission power decrease) and the rate of fission power decrease due to voiding, if any.

8. The rate of (metal water) reaction must be calculated using the Baker-Just equation with no decrease in reaction rate due to the lack of steam. This rate equation must be used to calculate metal-water reactions both on the outside surface of the cladding, and if ruptured, on the inside surface of the cladding. The reaction zone must extend axially at least three inches.

9. The initial oxide thickness (that affects the zirconium-water reaction rate) used should be no larger than can be reasonably justified, including consideration of the effects of manufacturing processes, hot-functional testing and exposure.

10. Exxon has agreed to provide calculations on a plant by plant basis to demonstrate that the plane of interest assumed for each plant is the plane in which peak cladding temperatures occur for that plant.

LOCA results are mostly driven by decay heat, which is proportional to operating power, and not affected significantly by EFW operation. In addition, the NRC staff has reviewed the SER limitations and operation in the EFW domain does not affect them. Thus, the staff concludes that the use of HUXY in the Monticello EFW domain is an acceptable extension of the existing approval.

EVALUATION OF CODES FOR ATWSI CALCULATIONS

AISHA

AISHA is a computer code that couples neutronics and THs and is used for analyzing BWR transients such as ATWSI. The code is optimized for simulating large power and flow oscillations associated with ATWS-I (ANP-3274P, Appendix A). AISHA is constructed from a selective mix of models from three parent codes; RAMONA5-FA, AISHA-10, and SINANO. AISHA has the following models: [[

]] The neutron cross section is coupled through MICROBURN-B2. [[

]]

Axial variations in flow area and hydraulic diameter are accounted for to accurately simulate bundles with pat-length fuel rods.

[[

]]

The input to AISHA are mostly automatic via coupling to MICROBURN-B2.
[[

]]

The major assumptions in the AISHA code are:
[[

]]

SINANO

SINANO is a single channel two phase flow THs code (ANP-3274P, Appendix B). [[

]] The dryout correlation coefficient is obtained from CPROM dryout correlation.

The key assumptions in the SINANO model are:
[[

]]

Major attributes to the transient one-dimensional THs model are:

1. [[ ]]
2. 1-D two phase flow,
3. [[ ]]
4. [[ ]]
5. [[ ]]
6. Post-dryout heat transfer in average and limiting rods, and
7. [[ ]]

The dryout and rewetting model consists of mass balance between the deposition from bulk flow as a source term and a combination of entrainment of liquid from the film into the bulk flow and evaporation as the loss term. The net rate of change of the liquid film mass is equated to the difference between the source and loss rates. The heat transfer coefficients for the dry and wet states and the time constants for the dry and wet states are and the time constants for the transition between the two states are obtained from the KATHY stability tests.

SINANO TH system comprises of field equations for [[

]] The NRC staff has

found that the accuracy of the [[ ]] that is used in the calculation of heat transfer coefficient was acceptable.

[[

]]

Steady state Dryout Correlation CPROM

Critical Power Reduced Order Model (CPROM) is a correlation developed by AREVA and is similar to AREVA's ACE correlations applicable to their ACE ATRIUM 10XM CHF correlation. CPROM correlation is well suited to fitting in to transient models of post-dryout that also includes dryout and rewetting with possible failure to rewet. CPROM is an integral part of the SINANO transient model. The correlation coefficients for a given BWR fuel type are obtained by fitting to the dryout testing database for that fuel type. [[

]]

The critical power obtained through CPROM process has the following characteristics:
[[

]]

The NRC staff has determined that the CPROM correlation for use in the ATWSI analysis is developed using NRC-approved methods and is acceptable.

Co-Resident Fuel Critical Heat Flux (CHF) Correlation

Critical power correlations for co-resident fuel are developed by AREVA using the approved methodology in EMF-2245. The following limitations were placed for the application of this methodology:

- Technology transfer to licensees who may be responsible for using these processes will be accomplished through AREVA and licensee procedures consistent with the requirements of GL 83-11, Supplement 1. This process includes the performance of an independent benchmarking calculation by AREVA for comparison to licensee-generated results to verify that the application of AREVA CHF correlations is properly applied for the first application by a licensee.

In the case of MNGP, AREVA generated SPCB CHF correlation parameters for GE14 based on simulated CHF conditions. The simulated conditions cover the range of applicability of the correlation in terms of pressure, flow, and subcooling.

The CHF correlation performance is not affected by operation in the EFW region because it is only dependent on the local conditions in the bundle. Therefore, the NRC staff concludes that the use of the EMF-2245 methodology in the EFW domain is an acceptable extension of the approved methodology.

KATHY Experimental ATWSI Data

AREVA obtained data from the Karlstein hydraulic loop (KATHY) with a full scale electrically heated AREVA ATRIUM 10XM bundle, which was tested under realistic ATWSI conditions of severe unstable density waves with simulated reactivity and power feedback. Experimental data representative of ATWSI conditions was collected and used to generate a new CPROM. CPROM and the SINANO models were benchmarked against the KATHY experimental data, both steady state and under oscillatory conditions.

Facility Description

KATHY is a BWR simulator located in Karlstein, Germany. The active core region consists of a full size AREVA ATRIUM 10XM electrically heated bundle. [[

]] Figure 9 shows a schematic of the facility.
[[

]]

Simulated Reactivity Feedback

In the KATHY experiments, the flow oscillations are self-induced by an instability of the tested bundle. In this way, the frequency and shape of the flow oscillation simulates accurately the oscillations that would be expected during an ATWSI event.

The rod power also oscillates in a fashion consistent with expectations during an ATWSI event. To this end, a reactivity feedback loop is simulated by the KATHY electronic control system.
[[

]] The code SINAN was adapted to implement the power reactivity feedback in KATHY.

[[

]]

Thus, the flow oscillations in KATHY are not forced by the modulation of a valve or pump, but by establishing an instability in the KATHY bundle and allowing it to develop. Large flow oscillations, leading to reverse flow at the core inlet were measured with reverse flow as large as [[

]]

Using the above experimental setup, data was collected for three different types of conditions:
[[

]]

Experimental Data

[[

]]

The [[ ]] conditions measured provide a significant experimental database for model development, and they represent the conditions that would be expected in a severe ATWSI event.

Experimental Heat Transfer Coefficient

Based on the KATHY experiments, AREVA has developed models for heat transfer coefficient (HTC) for:

1. Wet conditions at various fluid and surface conditions

2. Asymptotic value post dryout (stable film boiling)

AREVA reports that the measured wet heat transfer coefficient is represented accurately by the [[ ]], as seen in Figure 10.

The experimental data also supports the use of [[ ]], as seen in Figure 11. This is a very relevant value, because it controls the final asymptotic temperature of the hot node if failure to rewet exists. [[

]]

CPROM correlation development

The CPROM correlation provides a dynamic equation for the thickness of the liquid film surrounding the heated rod. [[
]] This form of the correlation has been used to benchmark the KATHY oscillatory flow/power tests described in the next section.

For steady state conditions, AREVA has obtained a closed form analytical equation to calculate the critical power using the coefficients of the dynamic CPROM correlation. Figure shows the calculated versus the measured critical power for the steady state AREVA database of critical power measurements for AREVA ATRIUM 10XM showing a standard deviation of [[ ]] percent. Figure 13 shows a similar analysis for GE14 showing a standard deviation of [[ ]] percent In the case of GE14, the data is not experimental, but generated using a suite of conditions provided by AREVA, which then used the GEH CPR correlation to provide the simulated data.

Based on these data, the NRC staff concludes that AREVA's CPROM correlation reproduces accurately (within [[ ]] percent) the experimental critical power for a wide range of operating conditions representative of normal reactor operation and ATWSI conditions.
[[

]]

KATHY Results

Section 3.0 of ANP-3284P documents relevant results from the KATHY periodic dryout conditions experiments and a comparison of the SINANO predictions for these conditions.

[[

]]

Figure 15 also shows the clad temperature calculated by SINANO using the CPROM correlation (black line in the figure)

[[

]]

Similar figures are provided in Section 3.0 of ANP-3284P for other tests. Figure 16 and Figure 17 show test results for cases with SINAN power feedback. For the power feedback cases, the power and flow oscillations are significantly larger, [[ ]] These very large oscillation amplitudes are representative of ATWSI conditions, and these test series requires a larger steady state CPR margin comparable to the margin representative of ATWSI conditions.
[[

]]

Evaluation

Overall, the SINANO predictions of fuel clad temperature (black lines in the figures) reproduce the trends in the data accurately. [[

]]

The NRC staff concludes that the SINANO model accurately predicts the conditions when failure to rewet causes a large temperature excursion in the KATHY oscillatory tests.

The SINANO models are a novel interpretation in the newly acquired data and have not been validated against experimental data other than this series of KATHY oscillatory tests, and the

formulation is significantly different than the one used on more established codes; thus, the generic applicability of these models to all possible conditions in a reactor may need additional work. However, the KATHY tests provide a sufficient benchmark because:

1. They are full-size 1-to-1 and do not require scaling to reactor conditions

2. Represent accurately the postulated conditions during ATWSI events

3. Have taken care to establish self-consistent oscillations with the proper frequency and phase delays between components (e.g., flow and power) at conditions representative of ATWSI events.

4. Results are not contradicted by any other experimental data available for similar conditions

5. [[

]]

Therefore, the NRC staff concludes that the SINANO models may be used to predict the hot fuel rod conditions during postulated ATWSI events.

Operating Conditions in MNGP Compared to Fleet Experience

Comparison to High-Power-Density Reactors in the Fleet

In the response to RAI 12 (Reference 28), AREVA provided operating data for high-power-density plants using approved AREVA methods in the fleet. The NRC staff has reviewed this information and reached the conclusion that the current fleet operating conditions bound the conditions expected in the EFW domain in MNGP.

Figure 19 shows a comparison of the axial distribution of the core-average void fraction for a high-power-density BWR6 that uses AREVA methods and MNGP at the 100 percent power and 80 percent core flow corner. Figure 20 shows a composite of the core exit void fraction for all the channels in the core. Figure 21 shows a comparison of flow and exit quality for Monticello operating conditions (yellow points) and other reactors in the fleet (blue points) along with the test points from the KATHY tests (red – void fraction, and magenta – delta-P). A comparison of the above results indicates that the operating reactor fleet operating conditions bound the expected operation in MNGP.

Figure 22 shows the evaluation of power distribution uncertainties for the high-power-density BWR6 reactor using AREVA methodology. The data covers 16 cycles of operation and no significant trends observed. The average 3D power distribution uncertainty is ~5 percent even with core exit void fractions as high as ~90 percent. This indicates that AREVA methods can: (1) predict the actual void fraction level accurately; (2) produce accurate cross sections; and (3) estimate the steady state power distribution accurately. Since MNGP operation in EFW results in lower void fraction levels, similar results should be expected.

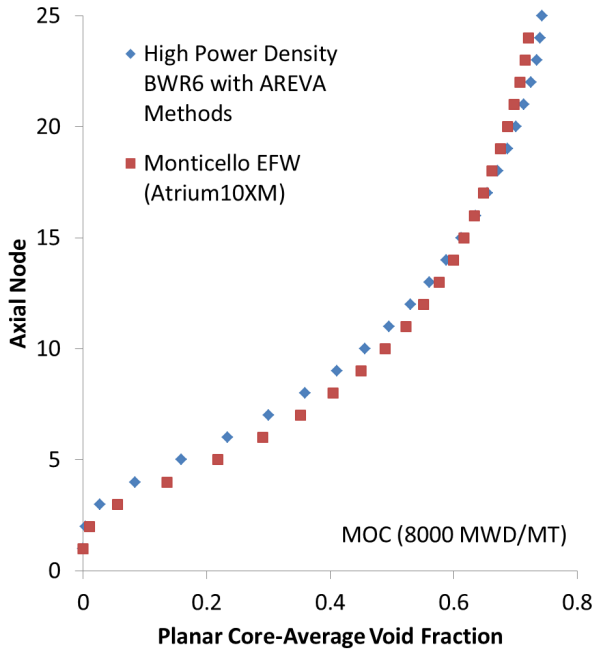

Figure 19 – Axial core-average void fraction in Monticello versus a high-power-density BWR6

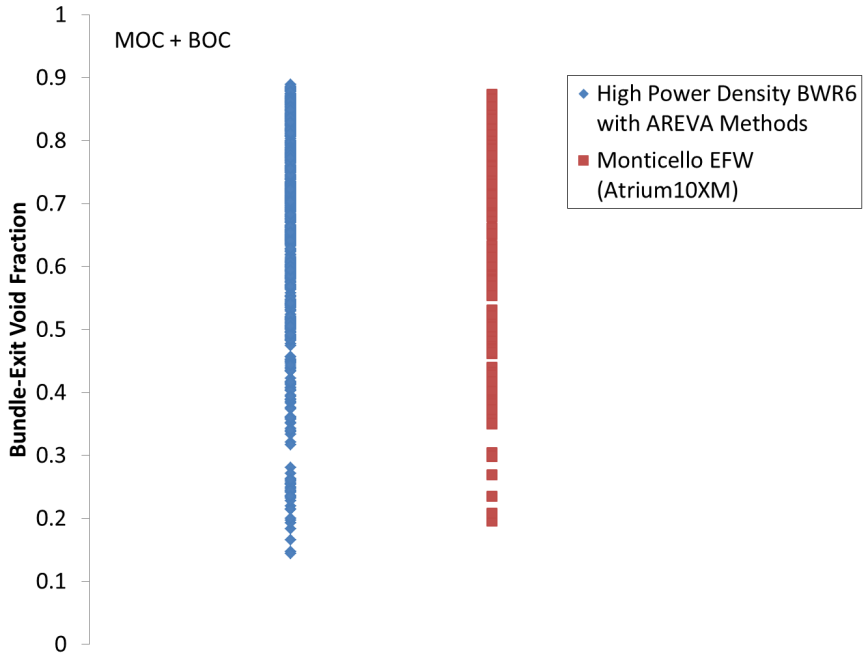

Figure 20 – Exit void fraction of all bundles in Monticello versus a high-power-density BWR6

[[

]]

TIP data pre- and post-EPU upgrades

In the response to RAI-1 (Reference 28), AREVA provided TIP-based power distribution uncertainties for two plants that have implemented a 20 percent EPU power increase in their last cycle. Following standard procedure, TIP data has been collected post-EPU and analyzed

using AREVA methods. As seen in Figure 23 and Figure 24, no significant trend can be observed pre- and post-EPU on the power distribution uncertainty. The NRC staff notes that an EPU power uprate does not increase the maximum allowed power-to-flow density because it simply follows the MELLLA line. However, operation pre-EPU occurs at 100 percent OLTP with variable flow, while post EPU, the operation occurs at the maximum power-to-flow density because the flow is fixed at ~100 percent. Thus, operation at EPU conditions reflects a higher average void fraction than pre-EPU for most of the cycle.

In the response to RAI 1, AREVA provided Figure 25, which shows the TIP statistical uncertainty for the EPU plants in their fleet. The red points are TIP uncertainties pre-EPU, and the green points represent the uncertainty post-EPU. This figure shows no discernible trend pre- and post-EPU. The figure also confirms that a few TIP measurements were collected up to a power-to-flow ratio value of 52 MWth/Mlb/hr, which exceeds the operating conditions in Monticello (<50 MWth/Mlb/h) and no discernible trend can be observed pre- and post-EPU on the power distribution uncertainty.

The NRC staff has reviewed this information and concludes that AREVA methods are likely to perform similarly in MNGP in the EFW domain. No significant increases in power distribution uncertainties should be expected.
[[

]]

Void Fraction measurements and Sensitivity Analysis

AREVA has performed void fraction measurements to specifically assess the impact of the AREVA ATRIUM 10XM fuel design attributes on void fraction predictions by AREVA codes. These were performed at the KATHY test facility using two prototypical BWR test assemblies (including AREVA ATRIUM 10XM) with part-length rods and mixing vane spacer grids. In

addition, AREVA has used reference void fraction data from FRIGG-2 and FRIGG-3. These data is summarized in ANP-3224.

AREVA shows an excellent agreement between the experimental data and the steady-state core simulator void fraction predictions, which uses the [[ ]] correlation. The agreement for the AREVA transient methods, which use the Ohkawa-Lahey correlation, is not as accurate as the [[ ]] results, and it shows a small negative bias (~-0.05) for void fractions between 40 percent and 80 percent, with unbiased agreement for α<40 percent and α>80 percent (see Figure 3 and Figure 4) Appendix D of ANP-3224 provides a sensitivity study where the void-quality correlation is biased up and down (±0.05 void) based on the scatter observed between calculation and measurements. The sensitivity results indicate that biasing the void-quality correlations results in a very small increase in ΔCPR, a very small decrease in SLMCPR, and a very small increase in OLMCPR for this study; therefore, the impact of the correlation bias is small (see Table D-2 of ANP-322)

NRC Staff Evaluation for AISHA and SINANO

AISHA code was developed by AREVA to simulate severe power oscillations associated with core instabilities unsuppressed by scram. AISHA has been benchmarked against measured stability data obtained from KATHY loop for AREVA ATRIUM 10XM fuel type which is very similar to AREVA ATRIUM 10XM fuel design. AISHA has been benchmarked against all the regional instabilities in actual BWR plants.

SINANO code models post-dryout heat transfer for the calculation of cladding temperature excursion in the highest power rod. SINANO models are benchmarked against data obtained from KATHY facility where a full scale electrically heated AREVA ATRIUM 10XM bundle was tested under realistic ATWSI conditions of severe unstable density waves with reactivity feedback.

The NRC staff has reviewed all aspects of these codes, such as, the neutron kinetics model, thermal hydraulic models including heat conduction and gap conductance, dryout and rewetting reduced order model, CPROM dryout correlation. Based on its review, the NRC staff has determined that the use of the code for ATWSI analysis in connection with the EFW LAR at MNGP is acceptable.

Conclusion

The NRC staff has reviewed the applicability of the following codes for use in MNGP EFW domain and has documented the review in the body of this SER.

1. CASMO4/MICROBURN-B2.
2. SAFLIM3D.
3. XCOBRA.
4. XCOBRA-T.
5. COTRANSA2.

6. STAIF.
7. RAMONA5-FA.
8. RODEX2.
9. RODEX4.
10. RELAX.
11. HUXY
12. AISHA
13. SINANO

All these codes are already approved for use up to the EFW line outside the EFW domain, but operating conditions in the fleet bound the expected conditions in the Monticello EFW domain. AREVA methods benchmark adequately against operating data in the fleet. In addition, AREVA has provided experimental data that benchmarks their methods to measurements with very high void fraction levels (close to 100 percent) and gamma scan data for AREVA ATRIUM 10XM.

Based on the NRC staff evaluation of the information provided, the staff concludes that the use of these codes in the MNGP EFW domain is an acceptable extension of the existing approval.

APPENDIX B

REQUEST FOR INFORMATION EVALUATION (RAI)

This Appendix provides a summary of the NRC staff's evaluation of the licensee's responses to RAIs documented in ANP-3434P, ANP-3435, and Xcel letter L-MT-15-065.

The NRC staff has issued a number of RAIs related to the review of the MNGP EFW LAR. A partial set of responses is documented in ANP-3434P. This section provides the staff evaluation of these responses.

RAI 1: TIP Uncertainty Data

Please provide Traversing Incore Probe (TIP) uncertainty data plots for high power-density plants with AREVA fuel (e.g. Kuosheng, Gunndremingen, Brunswick and Susquehanna) for a minimum of 3 cycles. For each plant provided, identify the pre- and post-Extended Power Uprate (EPU) cycles.

Evaluation

TIP comparisons (measured versus calculated with AREVA methods) are presented for 6 high-power-density plants. Two of the plants presented (BWR-H and BWR-I, see Figure 23 and Figure 24) have power densities larger than MNGP in EFW, and have implemented EPU in their last cycle and the TIP data shows no unusual trends, indicating that AREVA methods predict the power distribution accurately at power densities higher than expected in MNGP. This RAI is closed.

RAI 2: TIP Uncertainty Update Process

Monticello is expected to collect TIP data in the EFW domain.

a) Provide a short description of the TIP calibration process and how it is reflected in the Safety Limit Minimum Critical Power Ratio (SLMCPR) uncertainty calculations.
b) Provide a description of the process that would be used to reflect the higher uncertainty in the SLMCPR analyses if higher uncertainties are measured during EFW operation.

Evaluation

At MNGP, TIP measurements are performed periodically to: (1) calibrate LPRM detectors, and (2) provide data to the Gardel core monitoring system to calculate adaptive thermal margins. Three TIP machines are in use at MNGP.

In the response to the RAI, the licensee provided a summary of Studsvik report (Reference 60). The licensee stated that if during the routine TIP measurement process differences are found, by procedure, corrective action program action would be initiated by the Nuclear Analysis and

Design group. If the new power uncertainties are greater than those in the safety analysis, Xcel Energy would notify the vendor and an appropriate CPR penalty would be determined. This RAI issue is resolved.

RAI 3: EO-III Solution

a) Provide a roadmap and explanation of how the Extended Flow Window Stability (EFWS) trip is defined and implemented in technical specifications and COLR.
b) Define the methodology/process to calculate the EFWS trip on cycle specific basis.
c) Provide a justification for the removal of manual Backup Stability Protection (BSP) from section 5.6.3 of Technical Specifications.

Evaluation

In the response, the licensee describes the implementation of EO-III at MNGP. The EFWS setpoint is calculated on a cycle specific basis, and it corresponds to the cycle-specific Region1, which is reported in the COLR.

OPRM inoperable conditions require an immediate scram if the reactor operating conditions enter the exclusion region (Region I). Both Region I and Region II (exit region) are defined in the COLR.

EFWS Trip Inoperable requires an exit from the EFW region in <12 hours because the Channel Instability Exclusion Region (CIER) and Region I are not automatically protected. Operability requirements and armed conditions for EFWS are provided in the RAI response. This RAI is closed.

RAI 4: Equipment Out of Service

> The Monticello LAR states that Feedwater Heaters Out of Service (FWHOOS) is not allowed; however, all the LAR calculations have been performed with a +5F -10F FW [feedwater] temperature band. Clarify how the FWHOOS requirement implementation in the plant will be implemented.

Evaluation

MNGP clarifies that, due to hardware restrictions, operation with FWHOOS is not possible in MNGP and, thus, is not allowed. To cover small variations in temperature during normal operation, all calculations were performed with a +5F -10F feedwater temperature band. The response to this RAI is acceptable and the RAI is closed.

RAI 5: SLMCPR/OLMCPR Penalties

> Provide a tabular summary of the OLMCPR and SLMCPR penalties, if any, proposed for Monticello with AREVA methods. Provide a comparison between the current (GEH) penalties and proposed (AREVA) penalties, if any.

Evaluation

In the RAI response, MNGP proposes to remove all OLMCPR and SLMCPR penalties associated with their existing MELLLA+ implementation.

Currently, Monticello has a 0.01 OLMCPR penalty due to lack of experimental data supporting the [[ ]] void-quality correlation to demonstrate its accuracy, especially at high void fractions. However, the void fraction benchmarks provided in the response to RAI-9 against prototypical BWR assemblies operated at high void fractions along with sensitivity analysis documented in Appendix D of ANP-3224 provide the basis for the removal of the existing 0.01 OLMCPR limit for MNGP.

Currently, MNGP has a 0.03 SLMCPR penalty for power densities greater than 42MWt/Mlbm/hr because of lack of experimental data supporting the accuracy of power distributions, especially at high void fractions. Gamma scan data and TIP comparisons for high power densities have been provided in the response to RAIs 1, 9, and 10. This information does not show significant error trend with increased power-to-flow ratio. However, very little data has been collected for US reactors and no U.S. data has been provided to the NRC, further the 2 years of TIP data collected for MNGP in MELLLA+ was not provided and no data has been collected at MNGP at the lower corner of the EFW region. Therefore, the NRC staff concludes that it is prudent to maintain the 0.03 operating margin.

The NRC staff concludes that the new information provided by MNGP does not sufficiently justify the removal of the 0.03 SLMCPR penalty and, therefore, an additional 0.03 operating margin is necessary.

The NRC staff notes that there is currently an additional penalty for MNGP operation in the MELLLA+ domain. The penalty imposes a requirement to use the SLO uncertainties when calculating the SLMCPR inside the MELLLA+ domain. The justification for this penalty was that the uncertainty in evaluating the core flow during SLO conditions should bound the uncertainty associated with the large void fraction operation, for which there was no experimental data available for benchmarking. For operation in the EFW domain, this uncertainty remains in place as a license condition. This RAI is closed.

RAI 6: STAIF Input Data

> Provide data from sample STAIF calculations in sufficient detail to perform LAPUR confirmatory calculations.

Evaluation

Detailed data from sample STAIF calculations has been provided electronically. Using this data, the staff has performed confirmatory calculations with the LAPUR code. STAIF was used to determine the exclusion regions that define the EFWS channel instability region and the backup stability solutions. This determination relies on the decay ratio (DR) values calculated by STAIF, which are given separately for the core-wide and regional oscillation modes. The LAPUR confirmatory calculations have been carried out using identical axial nodalization, radial channel grouping, assumed 3-D power shape, and thermal hydraulic specifications as the STAIF calculations. The calculations using LAPUR have shown satisfactory agreement with the DR values calculated by STAIF for both the core-wide and regional modes. The NRC staff finds that the STAIF code is acceptable for use in defining exclusion region boundaries for MNGP EFW. This RAI is closed.

RAI 7: KATHY Axial Conduction

> Provide a discussion and sample calculation of the possible error incurred in the KATHY facility test measurements by discarding axial conduction to a possibly-nearby quench front.

Evaluation

The licensee provided both a detailed analysis and a bounding one for the error incurred by neglecting the axial conduction term in the KATHY experimental data processing. The licensee concludes that the error is at most [[ ]], which is negligible and can safely be ignored in the data processing. This RAI is closed.

RAI 8: Mitigated ATWSI Calculations

> ATWSI calculations in the Monticello LAR are for unmitigated (i.e., no operator actions) assumptions.
>
> a. Provide realistic ATWSI calculations under the expected conditions crediting operator actions. Provide sensitivity results for later operator action time.
> b. Provide a discussion of uncertainty treatment (e.g., hGap, inlet friction).
> c. Describe the methodology used by AISHA to excite the oscillation when the DR is close to 1.0.

Evaluation:

The licensee provided an evaluation of the mitigated ATWSI event in ANP-3435P. The conclusions can be summarized as:

1. When nominal feedwaterW temperature transient is assumed and the required operator action of 90 seconds is performed on time, the turbine trip ATWs event does not result in unstable oscillations.

2. Sensitivities were performed to identify how much margin exists if the operator actions are not performed in the required 90 seconds. The results indicate that Monticello operators have an additional [[ ]] of additional margin (i.e., initiate water level reduction at [[ ]]) before the oscillations become large enough to result in clad temperature excursions. Figure 8-3 of ANP-3435P (reproduced here as Figure 26) shows the sensitivity to operator actions

[[

]]

ANP-3435P also provides a sensitivity analysis to input parameters known to affect the stability of the system. Analysis of the turbine trip with bypass (TTWBP) ATWSI event was performed with perturbations in the gap heat transfer coefficient, core inlet friction, and the [[ ]] in AISHA. As expected, varying the relative stability of the system affects the starting time of the oscillations but not the ultimate PCT, which is controlled by the average power and the heat transfer coefficient. These results are presented in Figures 8-6, 8-7, and 8-8 of ANP-3435P.

In the response to RAI 8c in ANP-3435P, the licensee documents the procedure used to excite the oscillation to identify the DR. [[

]] This RAI is closed.

<u>RAI 9: Gamma-Densitometer Data – Void Fraction Uncertainty</u>

Currently, Monticello has a 0.01 OLMCPR penalty because of void uncertainty.

Provide a technical justification for an appropriate void uncertainty penalty for AREVA methods. Include available gamma densitometer data from KATHY for steady state void methods benchmark.

Evaluation:

AREVA has performed void fraction measurements to specifically assess the impact of the AREVA ATRIUM 10XM fuel design attributes on void fraction predictions by AREVA codes. These were performed at the KATHY test facility using two prototypical BWR test assemblies (including AREVA ATRIUM 10XM) with part-length rods and mixing vane spacer grids. In addition, AREVA has used reference void fraction data from FRIGG-2 and FRIGG-3.

AREVA shows an excellent agreement between the experimental data and the steady-state core simulator void fraction predictions, which uses the [[ ]] correlation. The agreement for the AREVA transient methods, which use the Ohkawa-Lahey correlation, is not as accurate as the [[ ]] results, and it shows a small negative bias (~-0.05) for void fractions between 40 percent and 80 percent, with unbiased agreement for $\alpha<40$ percent and $\alpha>80$ percent.

The void fraction benchmarks against prototypical BWR assemblies operated at high void fractions along with sensitivity analysis documented in Appendix D of ANP-3224 provide the basis for the removal of the existing 0.01 OLMCPR limit for MNGP. This RAI is closed.

<u>RAI 10: Gamma-Scan Data – Power Distribution Uncertainty</u>

Currently, Monticello has a 0.03 SLMCPR penalty for power densities greater than 42MWt/Mlbm/hr. Provide a technical justification for an appropriate power distribution uncertainty penalty supported by available data using AREVA methods. Include the available gamma scan data in the response.

Evaluation:

Gamma Scan data for modern fuel designs was presented in the 1999 CASMO-4/MICROBURN-B2 TR EMF-2158, Section 8. In addition, in the response to this RAI, AREVA presents the results of more recent gamma scan measurements in 48 assemblies, including AREVA ATRIUM 10XM.

The RAI response also includes a summary of TIP data comparisons as function of operating power-to-flow ratio. Most of the TIP data are between ~32 to ~42 MWth-hr/Mlb, but a few points were measured at power-flow ratios as high as 52 MWth-hr/Mlb. Since the maximum power-to-flow ratio at the lower-flow EFW corner in MNGP is 50 MWth-hr/Mlb, the TIP data provided by MNGP covers the entire MNGP operating domain. The TIP data does not show any significant error trend with respect to power-to-flow ratio.

This information, along with the TIP uncertainty provided on the response to RAI 1, has been used by the staff in its evaluation of the applicability of the 0.03 SLMCP penalty. Insufficient

information has been provided to remove this penalty so it is necessary for operation in the EFW domain. This RAI is closed.

RAI 11: Maximum Heat Flux Under Steam Cooling

AREVA has measured the steam heat transfer coefficient (HTC) using data from the KATHY oscillatory tests.

a) Using this HTC, provide the maximum heat flux without reaching peak cladding temperature (PCT) limits in absolute units and also in terms of percent the core-average heat flux and the limiting linear heat generation rate (LHGR).
b) Provide additional experimental data supporting the HTC value measured by AREVA.
c) Provide a justification for extrapolation of the HTC to higher pressures than those tested.

Evaluation:

A simple calculation using Monticello geometry and [[ ]] (which is supported by KATHY data) indicates that the hot rod will not reach limits (1204C, or 2200F) as long as the linear heat flux is less than [[

]] This RAI is closed.

RAI 12: MICROBURN-B2 Output Files

a. Provide the MICROBURN-B2 output file for the Monticello equilibrium cycle step-through with detailed edits of power and void fraction.
b. Provide the MICROBURN-B2 output file for an operating cycle of a high power density plant with detailed edits of power, void fraction and calculated TIP responses. Also, provide the corresponding measured TIP's.

Evaluation:

MICROBURN-B2 files were provided. The NRC staff has processed the information on these files and concluded that the operating void fraction in MNGP at EFW is bounded by the operating reactor. The staff notes, that TIP evaluations provided in the response to RAI-1 show that AREVA methods provide valid uncertainty levels for the operating reactors with high void density. Thus, AREVA methods are likely to produce similar uncertainty results for MNGP. This RAI is closed.

RAI 13: Clarification of RODEX2 and RODEX4 Methodology Use

> Clarify the COLR analysis methodology given that the RODEX2 and RODEX4 methods are referenced. Specifically, clarify which code will used for each transient in the COLR.

Evaluation:

RODEX 2 is the legacy T-M licensing and safety analysis code that is used for following approved methodologies:

1. Core average pellet to clad gap HTC used in COTRANSA2 for AOO and over-pressurization analyses.

2. LOCA analysis parameters, including initial stored energy, pellet to clad gap heat transfer coefficients and mechanical parameters needed for clad ballooning calculations.

3. Stability analysis using STAIF or RAMONA-5FA

4. Fire analyses based on RELAX/RODEX2/HUXY

RODEX4 is used in the T-M licensing and safety calculations for AOOs. RODEX4 calculations explicitly include the effects of thermal conductivity degradation (TCD), but RODEX2 calculations do not account explicitly for TCD. The use of non-TCD RODEX2 methods was evaluated in Appendix E of ANP-3224P. This RAI is closed.

RAI 14: MELLLA+ Methods and TRACG Limitations and Conditions

> NEDC-33435P, "Safety Analysis Report for Monticello Maximum Extended Load Line Limit Analysis Plus," dated December 2009, Appendices A, B, C, and D provide disposition of the 80 limitations and conditions for GEH codes and methods.
>
> Provide a table listing all the limitations and conditions listed and explain in detail how these conditions and limitations are satisfied with the use of AREVA codes and methods.

Evaluation:

Tables with the disposition of the limitations and conditions for GEH codes and methods were provided and form the basis for the SER evaluations. This RAI is closed

RAI 15: EO-III 5 percent Hot Channel Oscillation Magnitute (HCOM) Penalty

> Provide additional justification on the adequacy of the 5 percent HCOM penalty to maintain margin in the EFW operating domain given that oscillations may grow at a faster rate (higher DR) in the EFW operating domain.

Evaluation:

In the response to this RAI, the licensee addresses the impact of [[

]]

The response to this RAI is acceptable, and provides the basis to conclude that the 0.005 HCOM penalty was an overly conservative limitation imposed by the staff in the absence of detailed calculation at the time, and it is no longer required. This RAI is closed.

RAI 16: Axial Reactivity Variation during ATWSI

> Describe the impact of axial variation of reactivity between upper and lower parts of the core on the ATWSI analysis.

Evaluation:

The axial swings in reactivity are accounted for explicitly by the [[

]] This RAI is closed.

RAI 17: Dryout Phenomena in AISHA

> Provide a justification for the AISHA system code not calculating dryout phenomena.

Evaluation:

The licensee describes the AISHA model with respect to dryout. [[

]]

The NRC staff concurs with the licensee's evaluation of AISHA's treatment of hot rod dryout and bulk coolant conditions. This RAI is closed.

RAI 18: Time Dependent Gap Conductance

> ANP 3274 Appendix A states that "time dependent gap conductance could not be used in the analysis even under very large oscillation amplitudes during the transient." Describe the impact of this assumption in ATWSI analysis where the time dependent gap conductance is not accounted for.

Evaluation:

[[

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The NRC staff finds that the licensee's evaluation of time-dependent gap conductance acceptable. This RAI is closed.

RAI 19: Neutron Kinetics Theory

The following questions are related to the neutron kinetics theory presented in ANP-3274.

a) Explain why the kinetics theory is named "Adaptive Kinetics Theory."
b) Provide the boundary between the two neutron energy groups.
c) Explain how the adaptive reactivity (Equation A-13) and the adaptive function (Equation A-19) are applied in the AISHA code during the analysis.
d) Confirm that neutron conservation has been maintained in the finite difference solution scheme of the diffusion equation used in AISHA code. If not, provide a justification for the appropriateness of this approach.

Evaluation:

(a) [[ ]]
(b) The licensee responded that the boundary between the two neutron energy groups in the kinetics model is [[ ]]
(c) [[ ]]
(d) The licensee provided a response that since the AISHA kinetics code uses [[ ]]

This RAI is closed.

RAI 20: CPROM Model Development

a) Section B.2.5 of ANP-3274 introduces the "anchoring process" through which the CPROM correlation has been fit to the ACE ATRIUM 10XM and the SPCB correlations at a given value. Provide a detailed explanation of this "anchoring process," and include a justification for the value selected for "anchoring."
b) Describe how the radial peaking factor is absorbed by the anchoring process as mentioned in Section C.2 of ANP-3274P. Is this equivalent to the K-factor determination for the ACE ATRIUM 10XM correlation?
c) Provide details of how the pin-wise values of θ0 were generated and how their values are related to the additive constants for the ACE correlation.

d) Please provide details of the statistical analysis performed to determine the uncertainty and standard deviation for the CPROM correlation for the ATRIUM 10XM and GE14 fuel designs.

Evaluation:

(a) CPROM (Critical Power Reduced Order Model) is a dryout-rewet correlation. [[

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This RAI is closed.

RAI 21: Void History Bias

Provide a disposition to the potential effect of void history bias (like that noted in Limitation 9.11 of NEDC-33173-A) and applicability of AREVA methods to high void fraction conditions. Include a discussion on whether the transient models incorporate void history bias.

Evaluation:

The licensee stated that they have not identified any void-history bias and the void coefficient provides the best possible information for the transient analysis. No penalty is needed. This RAI is closed.

RAI 22: Boundary Condition Transfer

Provide a description of how the boundary conditions are passed between TRACG, AISHA, and SINANO.

Evaluation:

The licensee's RAI response describes the process to transfer boundary conditions from [[

]] This RAI is closed.

RAI 23: Relative CPR Performance of AREVA ATRIUM 10XM and GE14

Provide the dual recirculation pump trip (2RPT) results (e.g., MCPR/IMCPR and MCPR) vs. time for GE14 and ATRIUM 10XM.

Evaluation:

In the response to this, the licensee provided the CPR performance of GE14 and AREVA ATRIUM 10XM following a 2RPT. Even though AREVA ATRIUM 10XM does not recover as much CPR margin as GE 14 (AREVA ATRIUM 10XM CPR flow-induced recovery is lower by ~15 percent than GE14 at natural circulation), the differences are minimal indicating that the AREVA ATRIUM 10XM CPR performance is not degraded significantly at off-nominal flows. This RAI is closed.

RAI 24: Pin Peaking Factor

Describe the methodology for selecting the pin peaking factor for the ATWSI calculation.

Evaluation:

The hot-rod peaking factor for ATWSI calculations [[

]] This RAI is closed.

[[

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RAI 25: AOO Event Results

> Provide Table 2.2 listed in AREVA document, "AOO Event Results Summary – TSSS," FS1-0015233, "Monticello Cycle 28 EPU/MELLLA+ (EFW) Reload Licensing Report Support"

Evaluation:

The table was provided and the data has been incorporated by the NRC staff on the review of Section 5.0 of ANP-3295P. This RAI is closed.

RAI 26: ATWS Long Term Evaluation

> Provide a technical justification for not performing an ATWS Long-Term Evaluation for the transition to ATRIUM 10XM from GE14 (to EFW from MELLLA+).

Evaluation:

The RAI response and Section 7 of ANP-3295P provide justification for maintaining the GE14 ODYN analysis of record for ATWS long term. The staff has reviewed this information and finds the licensee's position that GE14 ODYN analysis of record is applicable for ATWS long term during EFW domain operation acceptable. This RAI is closed.

RAI 27: AREVA Codes and Methods

> Provide a list of the AREVA codes and methods used for Monticello EFW and their GEH counterparts

Evaluation:

A table documenting the equivalent AREVA codes to be used in the EFW domain evaluation was provided in the RAI response. The information in this RAI and Table 2.3 of ANP-3295P has been used by the NRC staff for the code applicability evaluation. This RAI is closed.

RAI 28 SLMCPR Calculation in the EFW Region

> Provide a justification for the core flow rate, assembly radial peaking, and nodal power uncertainties used to calculate SLMCPR at the EFW corners.

Evaluation:

The SLO flow uncertainties are used in the EFW domain (points L & M) for calculation of the SLMCPR. MNGP specific Gardel core monitoring system power distribution uncertainties were used. The SLMCPR methodology uses cycle-specific power distributions that take into account the assembly radial peaking. This RAI is closed.

RAI 29: Benchmarks against Plant Transients

> Provide benchmarks of analytical suite against modern transients (e.g., Susquehanna recirculation runback) to justify extension of methods.

Evaluation:

The LAR uses COTRANSA2 code for modeling the vessel and core response during transient analyses. The licensee provided results for the Peach Bottom turbine trip tests and for the recent Susquehanna flow reduction event. The benchmarks are successful and good agreement is observed between calculations and experiment. This RAI is closed.

RAI 30: Figures of Merit for Limiting Cases

> Provide a Table of figures of merit of the most limiting cases for ATWSI, ATWS, Overpressure, transients, and LOCA. This comparison is required to identify the limiting cases for each core loading (GE-14, ATRIUM 10XM and GE-14, and ATRIUM 10XM).

Evaluation:

In the RAI response, the licensee provided a table (Table 30-1 in ANP-3434P) with a comparison of the main figures of merit for the representative cycle 28. A comparison is performed for the transition core (GE14 + AREVA ATRIUM 10XM) versus the previous results with 100 percent GNF fuel. The safety and operating limits (SLMCPR & OLMCPR) are slightly reduced for the mixed transition core, but the results are acceptable. LOCA performance is slightly more conservative for AREVA ATRIUM 10XM core than for the 100 percent GNF core, and both result in acceptable results (PCT<2200F). ATWSI results are shown using the up-to-

date AREVA methodology (AISHA/SINANO), and they are acceptable (PCT<2200F) even for the unmitigated case. This RAI is closed.

RAI 31: O-III versus EO-III Setpoints

> Provide a discussion of the process followed by AREVA to ensure that the Enhanced Option III (EO-III) setpoints are conservative with respect to the BWROG, Option III, setpoints.

Evaluation:

In the response to this RAI, the licensee stated that the BSP regions calculated using the EO-III methodology are equal or more conservative than if they had been calculated with the standard Option III. The basis of licensee's assessment is that the BSP is the line where DR > (1 - uncertainties) and the methodology does not change the position of that line. Thus, the BSP regions calculated using the EO-III methodology may be used for Option III implementation before the EFWS (channel instability region) region is armed outside the EFW region. The response to this RAI is acceptable and the RAI is closed.

RAI 32: 2RPT ATWS

> The ATWSI analysis of record in the LAR is TTWBP. When operator actions are credited, the TTWBP does not show significant power oscillations and the limiting ATWSI transient becomes the 2RPT with failure to scram. Provide the results of 2RPT event with failure to scram. Describe the basis for boundary conditions and operator actions assumed for the analysis.

Evaluation:

The information was provided in the response to RAI 32 in ANP-3435P. [[

]] The results of these simulations are documented in Figures 32-, 32-2, and 32-3 (reproduced here as Figure 28). Even without operator actions, PCT values remain well below the customary acceptance criterion of 1204ºC (2200ºF). This RAI is closed.
[[

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RAI 33: Reactivity Model

Section 2.6 of ANP-3274P indicates that as density wave propagates upward, the total reactivity as well as the axial reactivity distribution changes and there are oscillating reactivity differences between the upper and the lower parts of the core.

a) Provide an estimate of the magnitude of the change in reactivity between the upper and lower part of the core and the frequency with which the change occurs.
b) Explain the impact of the reactivity swing between the upper and lower parts of the core on the ATWS instability event.
c) Explain how this swing in the reactivity in axial mode is accounted for by the neutron kinetics model used in the analysis.

Evaluation:
[[

]] This RAI is closed

RAI 34: ATWSI Analysis Assumptions

Provide a justification for the differences in the assumptions listed in Section 5 of Calculation notebook 32-9196882-001, "Monticello ATWS/INSTABILITY ANALYSES," and Section A.2.1 of ANP-3274P.

Evaluation:

The RAI response points to Section A.2.1 of ANP-3274P, where the AISHA assumptions are evaluated. This RAI is closed

RAI 35: Gadolinia (Gd) Rod Treatment in Hot Rod Selection

The ANP-3139P indicates that there will be Gd rods in the MNGP core. Describe the impact, if any, of Gd rods on the selection of average power rod and peak power-hot rod. Provide a justification for this treatment, or lack thereof, given that the melting temperature of Gd rods is less than a pure uranium oxide rod.

Evaluation:

The licensee stated that Gd rod could conceivably be limiting because the melting temperature is lower. However, if during the design process a Gd rod is found to be limiting, then changes are made to the nuclear design to lower the local peaking of the Gd rod until the design criteria are satisfied. The result is a design that meets the thermal-mechanical criteria using LHGR limits that will be applied during subsequent operation of the fuel. The RODEX4 methodology explicitly considers all of the fuel rods in the cycle design, including Gd rods, and identifies when criteria is not satisfied. This RAI is closed.

RAI 36: Heater Rods versus Fuel Rods

a) Provide a comparison between the test rods used in KATHY facility to the actual rods of GE-14 and ATRIUM 10XM fuel, including similarities and differences.
b) Explain how the differences between KATHY test rods and actual fuel rods impacts the benchmarking of the MNGP ATWSI analysis that is modelled using the AISHA/SINANO methodology. Include a discussion of the impact of using stainless steel rods and using skin heater rods.

Evaluation:

The RAI response provided the details of the KATHY rods and the differences between real uranium fuel rods and the KATHY heater. The response stated that the differences in time constant between fuel and heater rods are [[

]] This RAI is closed.

RAI 37: SINANO Benchmarking Process

> Please provide a summary of how the stability is determined for tests that include stable operation under different power and subcooling conditions using "noise analysis techniques" as mentioned in Section 3.0 of ANP-3284.

Evaluation:

The RAI response describes that two codes are used to determine the relative stability when self-sustaining oscillations do not develop (i.e., KATHY test point is stable). [[

]] This methodology is well-understood and has been applied over the years and benchmarked to stability tests. This RAI is closed.

APPENDIX C
LIST OF ACRONYMS

ACRONYM	DEFINITION
2RPT	two-pump recirculation pump trip
ΔCPR	delta critical power ratio
ACE	AREVA Critical power Evaluator
ADAMS	Agencywide Documents Access and Management System
AEC	Atomic Energy Commission
AOO	anticipated operational occurrences
APRM	average power range monitor
ASME	American Society of Mechanical Engineers
ASME Code	ASME Boiler and Vessel Pressure Code
AST	alternate source term
ATWS	anticipated transient without scram
ATWSI	anticipated transient without scram with instability
BSP	backup stability protection
BTP	Branch Technical Position
BTU	British Thermal Unit
BWR	boiling-water reactor
BWROG	Boiling-Water Reactors Owners Group
CCF	common-cause failure
CDA	Confirmation Density Algorithm
CF	core flow
CFR	*Code of Federal Regulations*
CHF	critical heat flux
CIER	channel instability exclusion region
COLR	Core Operating Limits Report
CPR	critical power ratio
CRDA	control rod drop accident
CRDS	control rod drive system
CS	Core Spray
D3	defense-in-depth and diversity
DBA	Design-Basis Accident
DIVOM	Delta Initial MCPR Versus Oscillation Magnitude
DSS-CD	Detect and Suppress Solution - Confirmation Density
EAB	exclusive area boundary
ECCS	Emergency Core Cooling System
EFW	extended flow window
EFWS	extended flow window stability
EOC	end-of-cycle
EOPs	emergency operating procedures
EPFOD	extended power/flow operating domain

ACRONYM	DEFINITION
EPG	emergency procedure guidelines
EPU	extended power uprate
°F	Fahrenheit
FHA	fuel handling accident
FW	Feedwater
FWCF	Feedwater Controller Failure
FWHOOS	Feedwater Heater Out-of-Service
Gd	Gadolinium
GDC	General Design Criterion/Criteria
GE	General Electric
GEH	GE-Hitachi Nuclear Energy
GL	Generic Letter
GNF	Global Nuclear Fuel
HCTL	heat capacity temperature limit
HPCI	high pressure coolant injection
HSBW	hot shutdown boron weight
HVAC	Heating, Ventilating, and Air Conditioning
IASCC	irradiation assisted stress-corrosion cracking
IMCPR	initial minimum critical power ratio
kW/ft	kilowatts per foot
KATHY	Karlstein hydraulic loop
LCO	limiting condition for operation
LAR	license amendment request
LHGR	linear heat generation rate
LOCA	loss-of-coolant accident
LOOP	loss of offsite power
LPCI	low pressure coolant injection
LPRM	local power range monitor
LPZ	low-population zone
LTR	licensing topical report
MAPLHGR	maximum average planar linear heat generation rate
MCPR	minimum critical power ratio
M+	Short for MELLLA+
M&E	Mass and energy
MELLLA	Maximum Extended Load Line Limit Analysis
MELLLA+	Maximum Extended Load Line Limit Analysis Plus
Mlbm/hr	million pounds mass per hour
MNGP	Monticello Nuclear Generating Plant
MOC	middle-of-cycle
MSIV	main steam isolation valve
MSIVC	main steam isolation valve closure
MSLB	main steam line break
MWt	megawatts thermal

ACRONYM	DEFINITION
NMS	neutron monitoring system
NPSH	net positive suction head
NPSHa	net positive suction head available
NRC	U.S. Nuclear Regulatory Commission
NUMAC	Nuclear Measurement Analysis and Control
OLMCPR	operating limit minimum critical power ratio
OLTP	original licensed thermal power
OPRM	oscillation power range monitor
ORNL	Oak Ridge National Laboratory
PBDA	period-based detection algorithm
PCT	peak cladding temperature
PRFO	pressure regulator failure open
PRNMS	power range neutron monitoring system
psi	pounds per square inch
psia	pounds per square inch atmospheric
psig	pounds per square inch gauge
RAI	request for additional information
RBM	rod block monitor
RCIC	reactor core isolation cooling
RCPB	reactor coolant pressure boundary
RCS	reactor coolant system
RG	Regulatory Guide
RHR	residual heat removal
RMS	root mean squared
RPS	reactor protection system
RPT	recirculation pump trip
RPV	reactor pressure vessel
RS	Review Standard
RTP	rated thermal power
RWE	rod withdrawal error
SAFDL	specified acceptable fuel design limit
SAG	severe accident guidelines
SAR	Safety Analysis Report
SBO	station blackout
SE	safety evaluation
SER	safety evaluation report
SLC	standby liquid control
SLCS	standby liquid control system
SLMCPR	Safety Limit Minimum Critical Power Ratio
SPCB	Siemens Power Corporation B
SLO	single loop operation
SR	surveillance requirement
SRLR	Supplemental Reload Licensing Report

ACRONYM	DEFINITION
SRP	Standard Review Plan
SRV	safety relief valve
SRVOOS	safety relief valve out-of-service
SSCs	structures, systems, and components
TH	Thermal Hydraulic
T-M	Thermal Mechanical
TIP	traversing incore probes
TLO	Two Loop Operation
TS	Technical Specification
UFSAR	Updated Final Safety Analysis Report
USAR	Updated Safety Analysis Report
Zr	Zirconium

SUBJECT: MONTICELLO NUCLEAR GENERATING PLANT - ISSUANCE OF AMENDMENT RE: EXTENDED FLOW WINDOW (CAC NO. MF5002) DATED FEBRUARY 23, 2017

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