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

September 18, 2018

Mr. William R. Gideon, Vice President
Brunswick Steam Electric Plant
Duke Energy Progress, LLC
8470 River Rd., SE (M/C BNP001)
Southport, NC 28461

SUBJECT: BRUNSWICK STEAM ELECTRIC PLANT, UNITS 1 AND 2 – ISSUANCE OF AMENDMENT REGARDING CORE FLOW OPERATING RANGE EXPANSION (MELLLA+) (EPID L-2016-LLA-0009)

Dear Mr. Gideon:

The U.S. Nuclear Regulatory Commission (the Commission) has issued the enclosed Amendment Nos. 285 and 313 to Renewed Facility Operating License Nos. DPR-71 and DPR-62 for Brunswick Steam Electric Plant (BSEP), Units 1 and 2, respectively. These amendments are in response to your license amendment request dated September 6, 2016, as supplemented by letters dated November 9, 2016, April 6, 2017, November 1, 2017, February 5, 2018, February 14, 2018, March 1, 2018, March 14, 2018, March 29, 2018 and April 10, 2018.

The amendments approve a revision to the BSEP Technical Specifications to allow plant operation from the currently licensed Maximum Extended Load Line Limit Analysis (MELLLA) domain to operation in the expanded MELLLA+ domain under the previously approved Extended Power Uprate conditions, including a 2923 megawatt thermal rated core thermal power. The proposed request would expand the operating boundary without changing the maximum licensed core power and maximum licensed core flow.

Enclosure 4 contains sensitive unclassified non-safeguards information. When separated from Enclosure 4, this document is DECONTROLLED.

A copy of the related Safety Evaluation is also enclosed. Notice of Issuance will be included in the Commission's biweekly *Federal Register* Notice.

Sincerely,

Andrew Hon, Project Manager
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-325 and 50-324

Enclosures:
1. Amendment No. 285 to DPR-71
2. Amendment No. 313 to DPR-62
3. Non-Proprietary Safety Evaluation
4. Proprietary Safety Evaluation

cc w/o enclosure 4: Listserv

ENCLOSURE 1

DUKE ENERGY PROGRESS, LLC

DOCKET NO. 50-325

BRUNSWICK STEAM ELECTRIC PLANT, UNIT 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

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

DUKE ENERGY PROGRESS, LLC

DOCKET NO. 50-325

BRUNSWICK STEAM ELECTRIC PLANT, UNIT 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 285
Renewed License No. DPR-71

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

   A. The application for amendment filed by Duke Energy Progress, LLC, dated September 6, 2016, as supplemented by letters dated November 9, 2016, April 6, 2017, November 1, 2017, February 5, 2018, February 14, 2018, March 1, 2018, March 14, 2018, March 29, 2018, and April 10, 2018, 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 as indicated in the attachment to this license amendment, and Paragraph 2.C.(2) of Renewed Facility Operating License No. DPR-71 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 285, are hereby incorporated in the license. Duke Energy Progress, LLC shall operate the facility in accordance with the Technical Specifications.

3. In addition, the license is amended by changes as indicated in the attachment to this license amendment, and Paragraph 3 of Renewed Facility Operating License No. DPR-71 is hereby amended to read as follows:

   3. Additional Conditions contained in Appendix B, as revised through Amendment No. 285, are hereby incorporated into this license. Duke Energy Progress, LLC shall operate the facility in accordance with the Additional Conditions.

4. Renewed Facility Operating License No. DPR-71 is also amended by the addition of a new license condition to Appendix B, "Additional Conditions," as indicated in the attachment to this amendment, which reads as follows:

Amendment <u>Number</u>	<u>Additional Conditions</u>	<u>Implementation Date</u>
285	The licensee shall not operate the facility within the MELLLA+ operating domain with Feedwater Temperature Reduction (FWTR) as defined in the Core Operating Limits Report.	Upon implementation of Amendment No. 285

5. This license amendment is effective as of the date of its issuance and shall be implemented no later than 60 days following startup from the 2019 Unit 2 refueling outage.

FOR THE NUCLEAR REGULATORY COMMISSION

V. Booma

Booma Venkataraman, Acting Chief
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Changes to the Renewed
Operating License, Technical
Specifications, and Appendix B,
"Additional Conditions"

Date of Issuance: September 18, 2018

ATTACHMENT TO LICENSE AMENDMENT NO. 285

BRUNSWICK STEAM ELECTRIC PLANT, UNIT 1

RENEWED FACILITY OPERATING LICENSE NO. DPR-71

DOCKET NO. 50-325

Replace the following pages of Renewed Facility Operating License No. DPR-71, Appendix A, Technical Specifications, and Appendix B, Additional Conditions with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Renewed Facility Operating License

Remove Page	Insert Page
6	6
10	10

Appendix A, Technical Specifications

Remove Page	Insert Page
3.1-22	3.1-22
3.1-23	3.1-23
3.3-3	3.3-3
3.3-4	3.3-4
3.3-5	3.3-5
3.3-6	3.3-6
3.3-7	3.3-7
3.3-8	3.3-8
3.3-9	3.3-9
3.3-10	3.3-10
3.4-1	3.4-1
3.4-2	3.4-2
5.0-20	5.0-20
5.0-21	5.0-21
5.0-22	5.0-22

Appendix B, Additional Conditions

Remove Page	Insert Page
App B-4	App B-4

(c) Transition License Conditions

1. Before achieving full compliance with 10 CFR 50.48(c), as specified by 2. below, risk-informed changes to the licensee's fire protection program may not be made without prior NRC review and approval unless the change has been demonstrated to have no more than a minimal risk impact, as described in 2. above.
2. The licensee shall implement the modifications to its facility, as described in Table S-1, "Plant Modifications Committed," of Duke letter BSEP 14-0122, dated November 20, 2014, to complete the transition to full compliance with 10 CFR 50.48(c) by the startup of the second refueling outage for each unit after issuance of the safety evaluation. The licensee shall maintain appropriate compensatory measures in place until completion of these modifications.
3. The licensee shall complete all implementation items, except item 9, listed in LAR Attachment S, Table S-2, "Implementation Items," of Duke letter BSEP 14-0122, dated November 20, 2014, within 180 days after NRC approval unless the 180th day falls within an outage window; then, in that case, completion of the implementation items, except item 9, shall occur no later than 60 days after startup from that particular outage. The licensee shall complete implementation of LAR Attachment S, Table S-2, Item 9, within 180 days after the startup of the second refueling outage for each unit after issuance of the safety evaluation.

C. This renewed license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations in 10 CFR Chapter I: Part 20, Section 30.34 of Part 30, Section 40.41 of Part 40, Sections 50.54 and 50.59 of Part 50, and Section 70.32 of Part 70; and is subject to all applicable provisions hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

The licensee is authorized to operate the facility at steady state reactor core power levels not in excess of 2923 megawatts thermal.

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 285, are hereby incorporated in the license. Duke Energy Progress, LLC shall operate the facility in accordance with the Technical Specifications.

For Surveillance Requirements (SRs) that are new in Amendment 203 to Renewed Facility Operating License DPR-71, the first performance is due at the end of the first surveillance interval that begins at implementation of Amendment 203. For SRs that existed prior to Amendment 203, including SRs with modified acceptance criteria and SRs whose frequency of

3. <u>Additional Conditions</u>

The Additional Conditions contained in Appendix B, as revised through Amendment No. 285, are hereby incorporated into this license. Duke Energy Progress, LLC shall operate the facility in accordance with the Additional Conditions.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

J. E. Dyer, Director
Office of Nuclear Reactor Regulation

Attachments:
1. Unit 1 – Technical Specifications – Appendices A and B

Date of Issuance: June 26, 2006

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.1.7.6	Verify each pump develops a flow rate ≥ 41.2 gpm at a discharge pressure ≥ 1190 psig.	In accordance with the INSERVICE TESTING PROGRAM
SR 3.1.7.7	Verify flow through one SLC subsystem from pump into reactor pressure vessel.	In accordance with the Surveillance Frequency Control Program
SR 3.1.7.8	Verify sodium pentaborate enrichment is ≥ 92 atom percent B-10.	Prior to addition to SLC tank

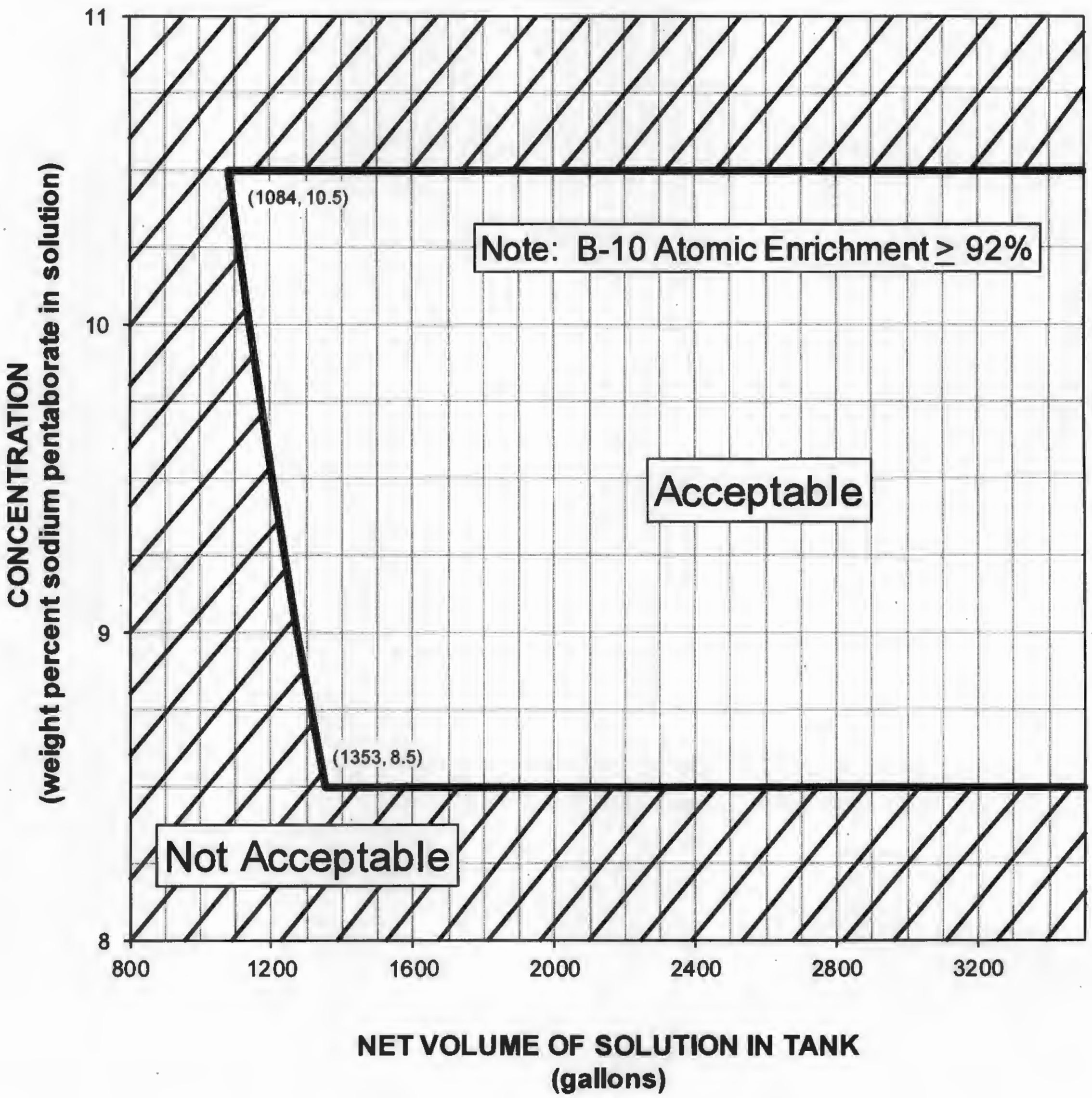

Figure 3.1.7-1 (page 1 of 1)
Sodium Pentaborate Solution Volume
Versus Concentration Requirements

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
F. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	F.1	Be in MODE 2.	6 hours
G. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	G.1	Be in MODE 3.	12 hours
H. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	H.1	Initiate action to fully insert all insertable control rods in core cells containing one or more fuel assemblies.	Immediately
I. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	I.1	Initiate action to implement the Manual BSP Regions defined in the COLR.	Immediately
	AND		
	I.2	Implement the Automated BSP Scram Region using the modified APRM Simulated Thermal Power – High Scram setpoints defined in the COLR.	12 hours
	AND		
	I.3	Initiate action in accordance with Specification 5.6.7.	Immediately

(continued)

ACTIONS (continued)

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

SURVEILLANCE REQUIREMENTS

--NOTES--

1. Refer to Table 3.3.1.1-1 to determine which SRs apply for each RPS Function.
2. When a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided the associated Function maintains RPS trip capability.

SURVEILLANCE		FREQUENCY
SR 3.3.1.1.1	(Not used.)	
SR 3.3.1.1.2	Perform CHANNEL CHECK.	In accordance with the Surveillance Frequency Control Program
SR 3.3.1.1.3	---------------------------NOTE--------------------------- Not required to be performed until 12 hours after THERMAL POWER ≥ 23% RTP. --- Adjust the average power range monitor (APRM) channels to conform to the calculated power while operating at ≥ 23% RTP.	In accordance with the Surveillance Frequency Control Program
SR 3.3.1.1.4	---------------------------NOTE--------------------------- Not required to be performed when entering MODE 2 from MODE 1 until 12 hours after entering MODE 2. --- Perform CHANNEL FUNCTIONAL TEST.	In accordance with the Surveillance Frequency Control Program

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.1.5	Perform a functional test of each automatic scram contactor.	In accordance with the Surveillance Frequency Control Program
SR 3.3.1.1.6	Verify the source range monitor (SRM) and intermediate range monitor (IRM) channels overlap.	Prior to withdrawing SRMs from the fully inserted position
SR 3.3.1.1.7	---NOTE--- Only required to be met during entry into MODE 2 from MODE 1. --- Verify the IRM and APRM channels overlap.	In accordance with the Surveillance Frequency Control Program
SR 3.3.1.1.8	Calibrate the local power range monitors.	In accordance with the Surveillance Frequency Control Program
SR 3.3.1.1.9	Perform CHANNEL FUNCTIONAL TEST.	In accordance with the Surveillance Frequency Control Program
SR 3.3.1.1.10	Calibrate the trip units.	In accordance with the Surveillance Frequency Control Program

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.1.11	-----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.	In accordance with the Surveillance Frequency Control Program
SR 3.3.1.1.12	Perform CHANNEL FUNCTIONAL TEST.	In accordance with the Surveillance Frequency Control Program
SR 3.3.1.1.13	-----NOTES----- 1. Neutron detectors are excluded. 2. For Function 1, not required to be performed when entering MODE 2 from MODE 1 until 12 hours after entering MODE 2. 3. For Functions 2.b and 2.f, the recirculation flow transmitters that feed the APRMs are included. ---------- Perform CHANNEL CALIBRATION.	In accordance with the Surveillance Frequency Control Program
SR 3.3.1.1.14	(Not used.)	
SR 3.3.1.1.15	Perform LOGIC SYSTEM FUNCTIONAL TEST.	In accordance with the Surveillance Frequency Control Program

(continued)

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
SR 3.3.1.1.16	Verify Turbine Stop Valve—Closure and Turbine Control Valve Fast Closure, Trip Oil Pressure—Low Functions are not bypassed when THERMAL POWER is ≥ 26% RTP.	In accordance with the Surveillance Frequency Control Program
SR 3.3.1.1.17	-------------------------------NOTES----------------------------- 1. Neutron detectors are excluded. 2. For Functions 3 and 4, the sensor response time may be assumed to be the design sensor response time. --- Verify the RPS RESPONSE TIME is within limits.	In accordance with the Surveillance Frequency Control Program
SR 3.3.1.1.18	Adjust recirculation drive flow to conform to reactor core flow.	Once within 7 days after reaching equilibrium conditions following refueling outage

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

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Intermediate Range Monitors					
a. Neutron Flux—High	2	3	G	SR 3.3.1.1.2 SR 3.3.1.1.4 SR 3.3.1.1.5 SR 3.3.1.1.6 SR 3.3.1.1.7 SR 3.3.1.1.13 SR 3.3.1.1.15	≤ 120/125 divisions of full scale
	5(a)	3	H	SR 3.3.1.1.2 SR 3.3.1.1.4 SR 3.3.1.1.5 SR 3.3.1.1.13 SR 3.3.1.1.15	≤ 120/125 divisions of full scale
b. Inop	2	3	G	SR 3.3.1.1.4 SR 3.3.1.1.5 SR 3.3.1.1.15	NA
	5(a)	3	H	SR 3.3.1.1.4 SR 3.3.1.1.5 SR 3.3.1.1.15	NA
2. Average Power Range Monitors					
a. Neutron Flux—High (Setdown)	2	3(c)	G	SR 3.3.1.1.2 SR 3.3.1.1.5 SR 3.3.1.1.7 SR 3.3.1.1.8 SR 3.3.1.1.11 SR 3.3.1.1.13	≤ 22.7% RTP
b. Simulated Thermal Power—High	1	3(c)	F	SR 3.3.1.1.2 SR 3.3.1.1.3 SR 3.3.1.1.5 SR 3.3.1.1.8 SR 3.3.1.1.11 SR 3.3.1.1.13 SR 3.3.1.1.18	≤ 0.61W + 65.2% RTP(b),(e) and ≤ 117.1% RTP

(continued)

(a) With any control rod withdrawn from a core cell containing one or more fuel assemblies.
(b) ≤ [0.55 (W - ΔW) + 62.6% RTP] when reset for single loop operation per LCO 3.4.1, "Recirculation Loops Operating." The value of ΔW is defined in plant procedures.
(c) Each APRM channel provides inputs to both trip systems.
(e) With OPRM Upscale (Function 2.f) inoperable, the Automated BSP Scram Region setpoints are implemented in accordance with Action I of this Specification.

Table 3.3.1.1-1 (page 2 of 3)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
2. Average Power Range Monitors (continued)					
c. Neutron Flux—High	1	3(c)	F	SR 3.3.1.1.2 SR 3.3.1.1.3 SR 3.3.1.1.5 SR 3.3.1.1.8 SR 3.3.1.1.11 SR 3.3.1.1.13	≤ 118.7% RTP
d. Inop	1,2	3(c)	G	SR 3.3.1.1.5 SR 3.3.1.1.11	NA
e. 2-Out-Of-4 Voter	1,2	2	G	SR 3.3.1.1.2 SR 3.3.1.1.5 SR 3.3.1.1.11 SR 3.3.1.1.15 SR 3.3.1.1.17	NA
f. OPRM Upscale	≥ 18% RTP(f)	3(c)	I	SR 3.3.1.1.2 SR 3.3.1.1.5 SR 3.3.1.1.8 SR 3.3.1.1.11 SR 3.3.1.1.13 SR 3.3.1.1.18	(d)
3. Reactor Vessel Steam Dome Pressure—High	1,2	2	G	SR 3.3.1.1.2 SR 3.3.1.1.5 SR 3.3.1.1.9 SR 3.3.1.1.10 SR 3.3.1.1.13 SR 3.3.1.1.15 SR 3.3.1.1.17	≤ 1077 psig
4. Reactor Vessel Water Level—Low Level 1	1,2	2	G	SR 3.3.1.1.2 SR 3.3.1.1.5 SR 3.3.1.1.9 SR 3.3.1.1.10 SR 3.3.1.1.13 SR 3.3.1.1.15 SR 3.3.1.1.17	≥ 153 inches
5. Main Steam Isolation Valve—Closure	1	8	F	SR 3.3.1.1.5 SR 3.3.1.1.9 SR 3.3.1.1.13 SR 3.3.1.1.15 SR 3.3.1.1.17	≤ 10% closed
6. Drywell Pressure—High	1,2	2	G	SR 3.3.1.1.2 SR 3.3.1.1.5 SR 3.3.1.1.9 SR 3.3.1.1.10 SR 3.3.1.1.13 SR 3.3.1.1.15	≤ 1.8 psig

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(c) Each APRM channel provides inputs to both trip systems.
(d) See COLR for OPRM Confirmation Density Algorithm (CDA) setpoints.
(f) Following DSS-CD implementation, DSS-CD is not required to be armed while in the DSS-CD Armed Region during the first reactor startup and during the first controlled shutdown that passes completely through the DSS-CD Armed Region. However, DSS-CD is considered OPERABLE and shall be maintained OPERABLE and capable of automatically arming for operation at recirculation drive flow rates above the DSS-CD Armed Region.

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.1 Recirculation Loops Operating

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

OR

One recirculation loop may be in operation provided the plant is not operating in the MELLLA+ operating domain, as 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;

c. LCO 3.2.3, "LINEAR HEAT GENERATION RATE (LHGR)," single loop operation limits specified in the COLR; and

d. LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation," Function 2.b (Average Power Range Monitors 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.	6 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Operation in the MELLLA+ domain with a single recirculation loop in operation.	B.1 Initiate action to exit the MELLLA+ operating domain.	Immediately
C. Required Action and associated Completion Time of Condition A or B not met. OR No recirculation loops in operation.	C.1 Be in MODE 3.	12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.1.1	-----------------------------NOTE----------------------------- Not required to be performed until 24 hours after both recirculation loops are in operation. --- Verify recirculation loop jet pump flow mismatch with both recirculation loops in operation: a. ≤ 10% of rated core flow when operating at < 75% of rated core flow; and b. ≤ 5% of rated core flow when operating at ≥ 75% of rated core flow.	In accordance with the Surveillance Frequency Control Program

5.6 Reporting Requirements (continued)

5.6.5 CORE OPERATING LIMITS REPORT (COLR)

a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:

1. The AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR) for Specification 3.2.1;

2. The MINIMUM CRITICAL POWER RATIO (MCPR) for Specification 3.2.2;

3. The LINEAR HEAT GENERATION RATE (LHGR) for Specification 3.2.3;

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

5. The Allowable Values and power range setpoints for Rod Block Monitor Upscale Functions for Specification 3.3.2.1.

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

1. NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel."

2. XN-NF-81-58(P)(A), RODEX2 Fuel Rod Thermal-Mechanical Response Evaluation Model.

3. XN-NF-85-67(P)(A), Generic Mechanical Design for Exxon Nuclear Jet Pump BWR Reload Fuel.

4. EMF-85-74(P) Supplement 1(P)(A) and Supplement 2(P)(A), RODEX2A (BWR) Fuel Rod Thermal-Mechanical Evaluation Model.

5. ANF-89-98(P)(A), Generic Mechanical Design Criteria for BWR Fuel Designs.

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5.6 Reporting Requirements (continued)

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

6. XN-NF-80-19(P)(A) Volume 1, Exxon Nuclear Methodology for Boiling Water Reactors – Neutronic Methods for Design and Analysis.

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

8. EMF-2158(P)(A), Siemens Power Corporation Methodology for Boiling Water Reactors: Evaluation and Validation of CASMO-4/MICROBURN-B2.

9. XN-NF-80-19(P)(A) Volume 3, Exxon Nuclear Methodology for Boiling Water Reactors, THERMEX: Thermal Limits Methodology Summary Description.

10. XN-NF-84-105(P)(A) Volume 1, XCOBRA-T: A Computer Code for BWR Transient Thermal-Hydraulic Core Analysis.

11. ANP-10307PA, AREVA MCPR Safety Limit Methodology for Boiling Water Reactors, Revision 0, June 2011.

12. ANF-913(P)(A) Volume 1, COTRANSA2: A Computer Program for Boiling Water Reactor Transient Analyses.

13. ANF-1358(P)(A), The Loss of Feedwater Heating Transient in Boiling Water Reactors.

14. EMF-2209(P)(A), SPCB Critical Power Correlation.

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

16. EMF-2361(P)(A), EXEM BWR-2000 ECCS Evaluation Model.

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

18. EMF-CC-074(P)(A) Volume 4, BWR Stability Analysis – Assessment of STAIF with Input from MICROBURN-B2.

19. NEDC-33075P-A, GE Hitachi Boiling Water Reactor, Detect and Suppress Solution - Confirmation Density, Revision 8, November 2013.

(continued)

5.6 Reporting Requirements (continued)

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

20. BAW-10247PA, Realistic Thermal-Mechanical Fuel Rod Methodology for Boiling Water Reactors, Revision 0, April 2008.

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

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.

5.6.6 Post Accident Monitoring (PAM) Instrumentation Report

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

5.6.7 Oscillation Power Range Monitor (OPRM) Report

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

Amendment Number	Additional Conditions	Implementation Date
282	During the extended EDG Completion Times authorized by Amendment No. 282, designated NLOs will be briefed, each shift, regarding cross-tying 480 V E7 bus to the 480 V E8 bus per 0AOP-36.1, *Loss of Any 4kV OR 480V Bus.*	Upon implementation of Amendment No. 282.
282	During the extended EDG Completion Times authorized by Amendment No. 282, designated NLOs will be briefed, each shift, regarding starting and tying the SUPP-DG to 4160 V emergency bus E4 per plant procedure 0EOP-01-SBO-08, *Supplemental DG Alignment.*	Upon implementation of Amendment No. 282.
282	During the extended EDG Completion Times authorized by Amendment No. 282, designated NLOs will be briefed, each shift, regarding load shed procedures and alignment of the FLEX diesel generators.	Upon implementation of Amendment No. 282.
282	During the extended EDG Completion Times authorized by Amendment No. 282, a continuous fire watch shall be established for the Unit 1 Cable Spread Room and for the Balance of Plant busses in the Unit 1 Turbine Building 20 foot elevation.	Upon implementation of Amendment No. 282.
285	The licensee shall not operate the facility within the MELLLA+ operating domain with Feedwater Temperature Reduction (FWTR), as defined in the Core Operating Limits Report.	Upon implementation of Amendment No. 285

ENCLOSURE 2

DUKE ENERGY PROGRESS, LLC

DOCKET NO. 50-324

BRUNSWICK STEAM ELECTRIC PLANT, UNIT 2

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

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

DUKE ENERGY PROGRESS, LLC

DOCKET NO. 50-324

BRUNSWICK STEAM ELECTRIC PLANT, UNIT 2

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 313
Renewed License No. DPR-62

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

   A. The application for amendment filed by Duke Energy Progress, LLC, dated September 6, 2016, as supplemented by letters dated November 9, 2016, April 6, 2017, November 1, 2017, February 5, 2018, February 14, 2018, March 1, 2018, March 14, 2018, March 29, 2018, and April 10, 2018, 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 as indicated in the attachment to this license amendment, and Paragraph 2.C.(2) of Renewed Facility Operating License No. DPR-62 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 313, are hereby incorporated in the license. Duke Energy Progress, LLC shall operate the facility in accordance with the Technical Specifications.

3. In addition, the license is amended by changes as indicated in the attachment to this license amendment, and Paragraph 3 of Renewed Facility Operating License No. DPR-62 is hereby amended to read as follows:

   3. Additional Conditions contained in Appendix B, as revised through Amendment No. 313, are hereby incorporated into this license. Duke Energy Progress, LLC shall operate the facility in accordance with the Additional Conditions.

4. Renewed Facility Operating License No. DPR-62 is also amended by the addition of a new license condition to Appendix B, "Additional Conditions," as indicated in the attachment to this amendment, which reads as follows:

Amendment Number	Additional Conditions	Implementation Date
313	The licensee shall not operate the facility within the MELLLA+ operating domain with Feedwater Temperature Reduction (FWTR) as defined in the Core Operating Limits Report.	Upon implementation of Amendment No. 313

4. This license amendment is effective as of the date of its issuance and shall be implemented no later than 60 days following startup from the 2019 Unit 2 refueling outage.

FOR THE NUCLEAR REGULATORY COMMISSION

V. Booma

Booma Venkataraman, Acting Chief
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Renewed Operating License, Technical Specifications, and Appendix B, "Additional Conditions"

Date of Issuance: September 18, 2018

ATTACHMENT TO LICENSE AMENDMENT NO. 313

BRUNSWICK STEAM ELECTRIC PLANT, UNIT 2

RENEWED FACILITY OPERATING LICENSE NO. DPR-62

DOCKET NO. 50-324

Replace the following pages of Renewed Facility Operating License No. DPR-62, Appendix A, Technical Specifications, and Appendix B, Additional Conditions with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Renewed Facility Operating License

Remove Page	Insert Page
6	6
10	10

Appendix A, Technical Specifications

Remove Page	Insert Page
3.1-22	3.1-22
3.1-23	3.1-23
3.3-3	3.3-3
3.3-4	3.3-4
3.3-5	3.3-5
3.3-6	3.3-6
3.3-7	3.3-7
3.3-8	3.3-8
3.3-9	3.3-9
3.3-10	3.3-10
3.4-1	3.4-1
3.4-2	3.4-2
5.0-20	5.0-20
5.0-21	5.0-21
5.0-22	5.0-22

Appendix B, Additional Conditions

Remove Page	Insert Page
App B-4	App B-4

(c) Transition License Conditions

1. Before achieving full compliance with 10 CFR 50.48(c), as specified by 2. below, risk-informed changes to the licensee's fire protection program may not be made without prior NRC review and approval unless the change has been demonstrated to have no more than a minimal risk impact, as described in 2. above.

2. The licensee shall implement the modifications to its facility, as described in Table S-1, "Plant Modifications Committed," of Duke letter BSEP 14-0122, dated November 20, 2014, to complete the transition to full compliance with 10 CFR 50.48(c) by the startup of the second refueling outage for each unit after issuance of the safety evaluation. The licensee shall maintain appropriate compensatory measures in place until completion of these modifications.

3. The licensee shall complete all implementation items, except Item 9, listed in LAR Attachment S, Table S-2, "Implementation Items," of Duke letter BSEP 14-0122, dated November 20, 2014, within 180 days after NRC approval unless the 180th day falls within an outage window; then, in that case, completion of the implementation items, except item 9, shall occur no later than 60 days after startup from that particular outage. The licensee shall complete implementation of LAR Attachment S, Table S-2, Item 9, within 180 days after the startup of the second refueling outage for each unit after issuance of the safety evaluation.

C. This renewed license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations in 10 CFR Chapter I: Part 20, Section 30.34 of Part 30, Section 40.41 of Part 40, Sections 50.54 and 50.59 of Part 50, and Section 70.32 of Part 70; 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

The licensee is authorized to operate the facility at steady state reactor core power levels not in excess of 2923 megawatts (thermal).

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 313, are hereby incorporated in the license. Duke Energy Progress, LLC shall operate the facility in accordance with the Technical Specifications.

For Surveillance Requirements (SRs) that are new in Amendment 233 to Renewed Facility Operating License DPR-62, the first performance is due at the end of the first surveillance interval that begins at implementation of Amendment 233. For SRs that existed prior to Amendment 233,

M. Mitigation Strategy License Condition

Develop and maintain strategies for addressing large fires and explosions and that include the following key areas:

(1) Fire fighting response strategy with the following elements:
1. Pre-defined coordinated fire response strategy and guidance
2. Assessment of mutual aid fire fighting assets
3. Designated staging areas for equipment and materials
4. Command and control
5. Training of response personnel

(2) Operations to mitigate fuel damage considering the following:
1. Protection and use of personnel assets
2. Communications
3. Minimizing fire spread
4. Procedures for implementing integrated fire response strategy
5. Identification of readily-available pre-staged equipment
6. Training on integrated fire response strategy
7. Spent fuel pool mitigation measures

(3) Actions to minimize release to include consideration of:
1. Water spray scrubbing
2. Dose to onsite responders

N. The licensee shall implement and maintain all Actions required by Attachment 2 to NRC Order EA-06-137, issued June 20, 2006, except the last action that requires incorporation of the strategies into the site security plan, contingency plan, emergency plan and/or guard training and qualification plan, as appropriate.

3. Additional Conditions

The Additional Conditions contained in Appendix B, as revised through Amendment No. 313, are hereby incorporated into this license. Duke Energy Progress, LLC shall operate the facility in accordance with the Additional Conditions.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

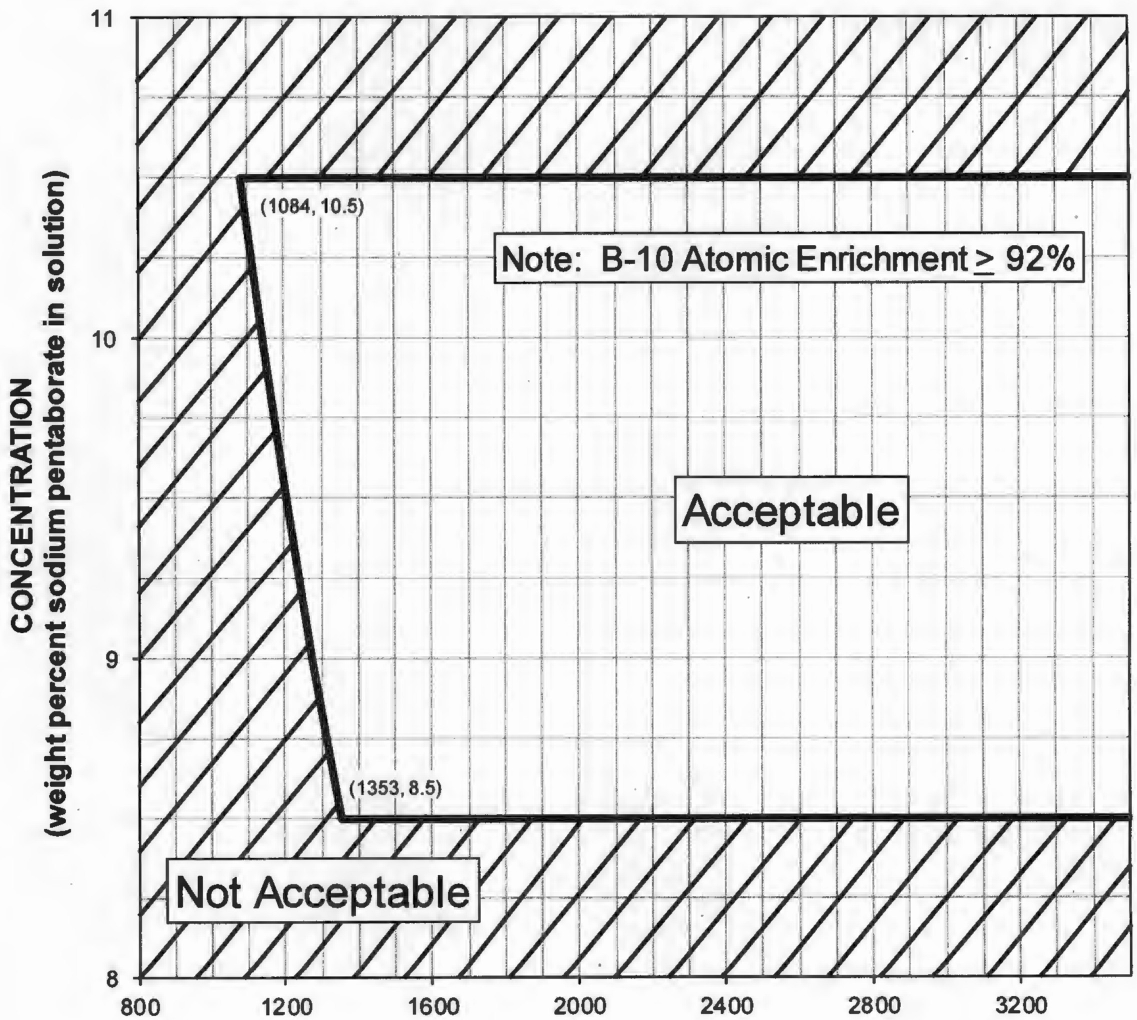
J. E. Dyer, Director
Office of Nuclear Reactor Regulation

Attachments:
1. Unit 2 – Technical Specifications – Appendices A and B

Date of Issuance: June 26, 2006

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.1.7.6	Verify each pump develops a flow rate ≥ 41.2 gpm at a discharge pressure ≥ 1190 psig.	In accordance with the INSERVICE TESTING PROGRAM
SR 3.1.7.7	Verify flow through one SLC subsystem from pump into reactor pressure vessel.	In accordance with the Surveillance Frequency Control Program
SR 3.1.7.8	Verify sodium pentaborate enrichment is ≥ 92 atom percent B-10.	Prior to addition to SLC tank



NET VOLUME OF SOLUTION IN TANK
(gallons)

Figure 3.1.7-1 (page 1 of 1)
Sodium Pentaborate Solution Volume
Versus Concentration Requirements

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
F. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	F.1	Be in MODE 2.	6 hours
G. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	G.1	Be in MODE 3.	12 hours
H. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	H.1	Initiate action to fully insert all insertable control rods in core cells containing one or more fuel assemblies.	Immediately
I. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	I.1	Initiate action to implement the Manual BSP Regions defined in the COLR.	Immediately
	<u>AND</u>		
	I.2	Implement the Automated BSP Scram Region using the modified APRM Simulated Thermal Power – High Scram setpoints defined in the COLR.	12 hours
	<u>AND</u>		
	I.3	Initiate action in accordance with Specification 5.6.7.	Immediately

(continued)

ACTIONS (continued)

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

SURVEILLANCE REQUIREMENTS

--NOTES--

1. Refer to Table 3.3.1.1-1 to determine which SRs apply for each RPS Function.
2. When a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided the associated Function maintains RPS trip capability.

SURVEILLANCE		FREQUENCY
SR 3.3.1.1.1	(Not used.)	
SR 3.3.1.1.2	Perform CHANNEL CHECK.	In accordance with the Surveillance Frequency Control Program
SR 3.3.1.1.3	------------------------------NOTE------------------------------ Not required to be performed until 12 hours after THERMAL POWER ≥ 23% RTP. --- Adjust the average power range monitor (APRM) channels to conform to the calculated power while operating at ≥ 23% RTP.	In accordance with the Surveillance Frequency Control Program
SR 3.3.1.1.4	------------------------------NOTE------------------------------ Not required to be performed when entering MODE 2 from MODE 1 until 12 hours after entering MODE 2. --- Perform CHANNEL FUNCTIONAL TEST.	In accordance with the Surveillance Frequency Control Program

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.1.5	Perform a functional test of each automatic scram contactor.	In accordance with the Surveillance Frequency Control Program
SR 3.3.1.1.6	Verify the source range monitor (SRM) and intermediate range monitor (IRM) channels overlap.	Prior to withdrawing SRMs from the fully inserted position
SR 3.3.1.1.7	------------------------------NOTE------------------------------ Only required to be met during entry into MODE 2 from MODE 1. --- Verify the IRM and APRM channels overlap.	In accordance with the Surveillance Frequency Control Program
SR 3.3.1.1.8	Calibrate the local power range monitors.	In accordance with the Surveillance Frequency Control Program
SR 3.3.1.1.9	Perform CHANNEL FUNCTIONAL TEST.	In accordance with the Surveillance Frequency Control Program
SR 3.3.1.1.10	Calibrate the trip units.	In accordance with the Surveillance Frequency Control Program

(continued)

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
SR 3.3.1.1.11	-----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.	In accordance with the Surveillance Frequency Control Program
SR 3.3.1.1.12	Perform CHANNEL FUNCTIONAL TEST.	In accordance with the Surveillance Frequency Control Program
SR 3.3.1.1.13	-----NOTES----- 1. Neutron detectors are excluded. 2. For Function 1, not required to be performed when entering MODE 2 from MODE 1 until 12 hours after entering MODE 2. 3. For Functions 2.b and 2.f, the recirculation flow transmitters that feed the APRMs are included. ---------- Perform CHANNEL CALIBRATION.	In accordance with the Surveillance Frequency Control Program
SR 3.3.1.1.14	(Not used.)	
SR 3.3.1.1.15	Perform LOGIC SYSTEM FUNCTIONAL TEST.	In accordance with the Surveillance Frequency Control Program

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.1.16	Verify Turbine Stop Valve—Closure and Turbine Control Valve Fast Closure, Trip Oil Pressure—Low Functions are not bypassed when THERMAL POWER is ≥ 26% RTP.	In accordance with the Surveillance Frequency Control Program
SR 3.3.1.1.17	-------------------------------NOTES--------------------------- 1. Neutron detectors are excluded. 2. For Functions 3 and 4, the sensor response time may be assumed to be the design sensor response time. -- Verify the RPS RESPONSE TIME is within limits.	In accordance with the Surveillance Frequency Control Program
SR 3.3.1.1.18	Adjust the flow control trip reference card to conform to reactor flow.	Once within 7 days after reaching equilibrium conditions following refueling outage

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

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Intermediate Range Monitors					
a. Neutron Flux—High	2	3	G	SR 3.3.1.1.2 SR 3.3.1.1.4 SR 3.3.1.1.5 SR 3.3.1.1.6 SR 3.3.1.1.7 SR 3.3.1.1.13 SR 3.3.1.1.15	≤ 120/125 divisions of full scale
	5(a)	3	H	SR 3.3.1.1.2 SR 3.3.1.1.4 SR 3.3.1.1.5 SR 3.3.1.1.13 SR 3.3.1.1.15	≤ 120/125 divisions of full scale
b. Inop	2	3	G	SR 3.3.1.1.4 SR 3.3.1.1.5 SR 3.3.1.1.15	NA
	5(a)	3	H	SR 3.3.1.1.4 SR 3.3.1.1.5 SR 3.3.1.1.15	NA
2. Average Power Range Monitors					
a. Neutron Flux—High (Setdown)	2	3(c)	G	SR 3.3.1.1.2 SR 3.3.1.1.5 SR 3.3.1.1.7 SR 3.3.1.1.8 SR 3.3.1.1.11 SR 3.3.1.1.13	≤ 22.7% RTP
b. Simulated Thermal Power—High	1	3(c)	F	SR 3.3.1.1.2 SR 3.3.1.1.3 SR 3.3.1.1.5 SR 3.3.1.1.8 SR 3.3.1.1.11 SR 3.3.1.1.13 SR 3.3.1.1.18	≤ 0.61W + 65.2% RTP(b),(e) and ≤ 117.1% RTP

(continued)

(a) With any control rod withdrawn from a core cell containing one or more fuel assemblies.
(b) ≤ [0.55 (W - ΔW) + 62.6% RTP] when reset for single loop operation per LCO 3.4.1, "Recirculation Loops Operating." The value of ΔW is defined in plant procedures.
(c) Each APRM channel provides inputs to both trip systems.
(e) With OPRM Upscale (Function 2.f) inoperable, the Automated BSP Scram Region setpoints are implemented in accordance with Action I of this Specification.

Table 3.3.1.1-1 (page 2 of 3)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
2. Average Power Range Monitors (continued)					
c. Neutron Flux—High	1	3(c)	F	SR 3.3.1.1.2 SR 3.3.1.1.3 SR 3.3.1.1.5 SR 3.3.1.1.8 SR 3.3.1.1.11 SR 3.3.1.1.13	≤ 118.7% RTP
d. Inop	1,2	3(c)	G	SR 3.3.1.1.5 SR 3.3.1.1.11	NA
e. 2-Out-Of-4 Voter	1,2	2	G	SR 3.3.1.1.2 SR 3.3.1.1.5 SR 3.3.1.1.11 SR 3.3.1.1.15 SR 3.3.1.1.17	NA
f. OPRM Upscale	≥ 18% RTP(f)	3(c)	I	SR 3.3.1.1.2 SR 3.3.1.1.5 SR 3.3.1.1.8 SR 3.3.1.1.11 SR 3.3.1.1.13 SR 3.3.1.1.18	(d)
3. Reactor Vessel Steam Dome Pressure—High	1,2	2	G	SR 3.3.1.1.2 SR 3.3.1.1.5 SR 3.3.1.1.9 SR 3.3.1.1.10 SR 3.3.1.1.13 SR 3.3.1.1.15 SR 3.3.1.1.17	≤ 1077 psig
4. Reactor Vessel Water Level—Low Level 1	1,2	2	G	SR 3.3.1.1.2 SR 3.3.1.1.5 SR 3.3.1.1.9 SR 3.3.1.1.10 SR 3.3.1.1.13 SR 3.3.1.1.15 SR 3.3.1.1.17	≥ 153 inches
5. Main Steam Isolation Valve—Closure	1	8	F	SR 3.3.1.1.5 SR 3.3.1.1.9 SR 3.3.1.1.13 SR 3.3.1.1.15 SR 3.3.1.1.17	≤ 10% closed
6. Drywell Pressure—High	1,2	2	G	SR 3.3.1.1.2 SR 3.3.1.1.5 SR 3.3.1.1.9 SR 3.3.1.1.10 SR 3.3.1.1.13 SR 3.3.1.1.15	≤ 1.8 psig

(continued)

(c) Each APRM channel provides inputs to both trip systems.
(d) See COLR for OPRM Confirmation Density Algorithm (CDA) setpoints.
(f) Following DSS-CD implementation, DSS-CD is not required to be armed while in the DSS-CD Armed Region during the first reactor startup and during the first controlled shutdown that passes completely through the DSS-CD Armed Region. However, DSS-CD is considered OPERABLE and shall be maintained OPERABLE and capable of automatically arming for operation at recirculation drive flow rates above the DSS-CD Armed Region.

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.1 Recirculation Loops Operating

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

OR

One recirculation loop may be in operation provided the plant is not operating in the MELLLA+ operating domain, as 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;

c. LCO 3.2.3, "LINEAR HEAT GENERATION RATE (LHGR)," single loop operation limits specified in the COLR; and

d. LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation," Function 2.b (Average Power Range Monitors 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.	6 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. Operation in the MELLLA+ domain with a single recirculation loop in operation.	B.1	Initiate action to exit the MELLLA+ operating domain.	Immediately
C. Required Action and associated Completion Time of Condition A or B not met. OR No recirculation loops in operation.	C.1	Be in MODE 3.	12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.1.1	-----------------------------NOTE----------------------------- Not required to be performed until 24 hours after both recirculation loops are in operation --- Verify recirculation loop jet pump flow mismatch with both recirculation loops in operation: a. $\leq$ 10% of rated core flow when operating at < 75% of rated core flow; and b. $\leq$ 5% of rated core flow when operating at $\geq$ 75% of rated core flow.	In accordance with the Surveillance Frequency Control Program

5.6 Reporting Requirements (continued)

5.6.5 CORE OPERATING LIMITS REPORT (COLR)

a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:

1. The AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR) for Specification 3.2.1;

2. The MINIMUM CRITICAL POWER RATIO (MCPR) for Specification 3.2.2;

3. The LINEAR HEAT GENERATION RATE (LHGR) for Specification 3.2.3;

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

5. The Allowable Values and power range setpoints for Rod Block Monitor Upscale Functions for Specification 3.3.2.1.

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

1. NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel."

2. XN-NF-81-58(P)(A), RODEX2 Fuel Rod Thermal-Mechanical Response Evaluation Model.

3. XN-NF-85-67(P)(A), Generic Mechanical Design for Exxon Nuclear Jet Pump BWR Reload Fuel.

4. EMF-85-74(P) Supplement 1(P)(A) and Supplement 2(P)(A), RODEX2A (BWR) Fuel Rod Thermal-Mechanical Evaluation Model.

5. ANF-89-98(P)(A), Generic Mechanical Design Criteria for BWR Fuel Designs.

(continued)

5.6 Reporting Requirements (continued)

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

6. XN-NF-80-19(P)(A) Volume 1, Exxon Nuclear Methodology for Boiling Water Reactors – Neutronic Methods for Design and Analysis.

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

8. EMF-2158(P)(A), Siemens Power Corporation Methodology for Boiling Water Reactors: Evaluation and Validation of CASMO-4/MICROBURN-B2.

9. XN-NF-80-19(P)(A) Volume 3, Exxon Nuclear Methodology for Boiling Water Reactors, THERMEX: Thermal Limits Methodology Summary Description.

10. XN-NF-84-105(P)(A) Volume 1, XCOBRA-T: A Computer Code for BWR Transient Thermal-Hydraulic Core Analysis.

11. ANP-10307PA, AREVA MCPR Safety Limit Methodology for Boiling Water Reactors, Revision 0, June 2011.

12. ANF-913(P)(A) Volume 1, COTRANSA2: A Computer Program for Boiling Water Reactor Transient Analyses.

13. ANF-1358(P)(A), The Loss of Feedwater Heating Transient in Boiling Water Reactors.

14. EMF-2209(P)(A), SPCB Critical Power Correlation.

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

16. EMF-2361(P)(A), EXEM BWR-2000 ECCS Evaluation Model.

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

18. EMF-CC-074(P)(A) Volume 4, BWR Stability Analysis – Assessment of STAIF with Input from MICROBURN-B2.

19. NEDC-33075P-A, GE Hitachi Boiling Water Reactor, Detect and Suppress Solution - Confirmation Density, Revision 8, November 2013.

(continued)

5.6 Reporting Requirements (continued)

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

20. BAW-10247PA, Realistic Thermal-Mechanical Fuel Rod Methodology for Boiling Water Reactors, Revision 0, April 2008.

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

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.

5.6.6 Post Accident Monitoring (PAM) Instrumentation Report

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

5.6.7 Oscillation Power Range Monitor (OPRM) Report

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

Amendment Number	Additional Conditions	Implementation Date
310	During the extended EDG Completion Times authorized by Amendment No. 310, dedicated non-licensed operators (NLOs) shall be briefed, each shift, regarding cross tying the 4160 V emergency bus E2 to 4160 V emergency bus E4 per plant procedure 0AOP-36.1, *Loss of Any 4kV OR 480V Bus*.	Upon implementation of Amendment No. 310.
310	During the extended EDG Completion Times authorized by Amendment No. 310, dedicated NLOs will be briefed, each shift, regarding cross-tying 480 V E7 bus to the 480 V E8 bus per 0AOP-36.1, *Loss of Any 4kV OR 480V Bus*.	Upon implementation of Amendment No. 310.
310	During the extended EDG Completion Times authorized by Amendment No. 310, dedicated NLOs will be briefed, each shift, regarding starting and tying the SUPP-DG to 4160 V emergency bus E4 per plant procedure 0EOP-01-SBO-08, *Supplemental DG Alignment*.	Upon implementation of Amendment No. 310.
310	During the extended EDG Completion Times authorized by Amendment No. 310, designated NLOs will be briefed, each shift, regarding load shed procedures and alignment of the FLEX diesel generators.	Upon implementation of Amendment No. 310.
310	During the extended EDG Completion Times authorized by Amendment No. 310, a continuous fire watch shall be established for the Unit 2 Cable Spread Room and for the Balance of Plant busses in the Unit 2 Turbine Building 20 foot elevation.	Upon implementation of Amendment No. 310.
310	During the extended EDG Completion Times authorized by Amendment No. 310, the FLEX pump and FLEX Unit 2 hose trailer shall be staged at the south side of the Unit 2 Condensate Storage Tank to support rapid deployment in the event the FLEX pump is needed for Unit 2 inventory control.	Upon implementation of Amendment No. 310.
313	The licensee shall not operate the facility within the MELLLA+ operating domain with Feedwater Temperature Reduction (FWTR), as defined in the Core Operating Limits Report.	Upon implementation of Amendment No. 313.

ENCLOSURE 3
(NON-PROPRIETARY)

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NOS. 285 AND 313

TO RENEWED FACILITY OPERATING LICENSE NOS. DPR-71 AND DPR-62

DUKE ENERGY PROGRESS, LLC

BRUNSWICK STEAM ELECTRIC PLANT, UNITS 1 AND 2

DOCKET NOS. 50-325 AND 50-324

Proprietary information pursuant to Title 10 of the *Code of Federal Regulations* Section 2.390 has been redacted from this document.

Redacted information is identified by blank space enclosed within double brackets as shown here [[ ]].

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APPENDICES

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NOS. 285 AND 313

TO RENEWED FACILITY OPERATING LICENSE NOS. DPR-71 AND DPR-62

DUKE ENERGY PROGRESS, LLC

BRUNSWICK STEAM ELECTRIC PLANT, UNITS 1 AND 2

DOCKET NOS. 50-325 AND 50-324

1.0 INTRODUCTION

By application dated September 6, 2016 (Reference 1), as supplemented by letters dated November 9, 2016 (Reference 2), April 6, 2017 (Reference 33), November 1, 2017 (Reference 30 and 31), February 5, 2018 (Reference 7), February 14, 2018 (Reference 34), March 1, 2018 (Reference 29), March 14, 2018 (Reference 10), March 29, 2018 (Reference 35) and April 10, 2018 (Reference 11), Duke Energy Progress, LLC (Duke Energy, the licensee) submitted a license amendment request (LAR) for Brunswick Steam Electric Plant, Units 1 and 2 (BSEP or Brunswick). The proposed amendment would revise the Technical Specifications (TSs) and Facility Operating Licenses to allow operation in the expanded Maximum Extended Load Line Limit Analysis Plus (MELLLA+ or M+) domain boundary without changing the maximum licensed core power and maximum licensed core flow (CF).

The supplements dated April 6, 201,7 through April 10, 2018 above provided additional information that clarified the application, did not expand the scope of the application as originally noticed and did not change the Nuclear Regulatory Commission (NRC) staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on January 3, 2017 (82 FR 158).

2.0 REGULATORY EVALUATION

The following Title 10 of the *Code of Federal Regulations* (CFR) Requirements apply to this review:

- 10 CFR Part 20, "Standards for Protection against Radiation," which limits doses to an individual exposed to radioactive material and radiation sources.

- 10 CFR 50.36, "Technical specifications," which contains regulatory requirements related to the contents of the TS. Specifically, 10 CFR 50.36(a)(1) states, "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. A summary statement of the bases or reasons for such specifications, other than those covering administrative controls, shall also be included in the application, but shall not become part of the technical specifications." Section 50.36(c)(3) states that "[s]urveillance 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 [LCOs] will be met."

- 10 CFR 50.36a "Technical specifications on effluents from nuclear power reactors," which requires licensees to develop and follow operating procedures for the control of effluents, to keep average annual releases of radioactive material in effluents and their resultant committed effective dose equivalents at small percentages of the dose limits specified in 10 CFR 20.1301, and to establish TSs that require compliance with the public dose limits in 10 CFR 20.1301. In addition, 10 CFR 50.36a provides licensees the flexibility of operations that may temporarily result in effluent releases higher than such small percentages of the dose limits, and expects that the licensee will exert its best efforts to keep levels of radioactive effluent ALARA (i.e., within the numerical guides established in 10 CFR Part 50, Appendix I).

- 10 CFR 50.44, "Combustible gas control for nuclear power reactors," which requires that plants be provided with the capability of controlling combustible gas concentrations in the containment atmosphere.

- 10 CFR 50.46, which sets standards for the calculation of emergency core cooling system (ECCS) performance and acceptance criteria for that calculated performance.

- 10 CFR Part 50, Appendix I, "Numerical Guides for Design Objectives and Limiting Conditions for Operation to Meet the Criterion 'As Low As Reasonably Achievable [ALARA]' for Radioactive Material in Light-Water-Cooled Nuclear Power Reactor Effluents," which provides the numerical guidance on limiting conditions for operation sufficient to meet the ALARA requirement for light-water-cooled nuclear power reactors.

- 10 CFR Part 50, Appendix K, which sets required and acceptable features of evaluation models for heat removal by the ECCS after the blowdown phase of a loss-of-coolant accident (LOCA).

- 10 CFR 50.55a(h), which requires that the protection systems meet IEEE Standard 279. Section 4.2 of IEEE 279-1971 discusses the general functional requirement for protection systems to assure they satisfy the single failure criterion.

- 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 to provide the means to address an ATWS event, an Anticipated Operational Occurrence (AOO) defined in Appendix A of this part, followed by the failure of the reactor trip portion of the protection system specified in General Design Criteria (GDCs) 20 of Appendix A.

- 10 CFR 50.62(c)(4), which it requires that the SLC system be capable of reliably injecting a borated water solution into the reactor pressure vessel (RPV) at a Boron concentration, Boron enrichment, and flow rate that provides a set level of reactivity control.

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

- NUREG-0737, "Clarification of TMI [Three Mile Island] Action Plan Requirements," Item II.B.2, "Design Review of Plant Shielding and Environmental Qualification of Equipment for Spaces/Systems Which May Be Used in Postaccident Operations," which specifies the performance of radiation shielding design reviews to ensure that design permits adequate access to important areas and provides for protection of safety equipment from radiation, following an accident. NRC Order issued on March 14, 1983, confirmed the licensee's commitment to implement those post-TMI related items set forth in this NUREG.

The BSEP design was reviewed for construction under the *General Design Criteria for Nuclear Power Plant Construction*, issued for comment by the AEC in July 1967 and is committed to meet the intent of the General Design Criteria (GDCs), published in the *Federal Register* on May 21, 1971, as Appendix A to 10 CFR Part 50. The following GDCs are applicable to this review:

- GDC 1, *Quality standards and records*, which requires structures, systems, and components important to safety to be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed.

- GDC 4, *Environmental and dynamic effects design bases*, which requires structures, systems, and components (SSCs) important to safety be protected against dynamic effects associated with flow instabilities and loads.

- GDC 5, *Sharing of structures, systems, and components*, which requires 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.

- GDC 10, *Reactor design*, which requires the reactor protection system be capable of terminating any anticipated transients, including unstable power oscillations, without challenge to the fuel.

- GDC 11, *Reactor inherent protection*, which requires 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.

- GDC 12, *Suppression of reactor power oscillations*, which requires unstable oscillations with the potential of exceeding specified acceptable fuel design limits (SAFDLs) either be impossible or reliably and readily detected and suppressed.

- GDC 13, *Instrumentation and control*, which requires instrumentation and controls be provided to monitor variables and systems affecting the fission process over anticipated ranges for normal operation, Anticipated Operational Occurrences (AOOs) and accident conditions, and to maintain the variables and systems within prescribed operating ranges.

- GDC 14, *Reactor coolant pressure boundary*, requires that the reactor coolant pressure boundary be designed and constructed so as to have an exceedingly low probability of gross rupture or significant leakage throughout its design lifetime.

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

- GDC 19, *Control room*, which requires that adequate radiation protection be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent, to any part of the body, for the duration of the accident.

- GDC 20, *Protection system functions,* which requires the reactor 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.

- GDC 21, *Protection system reliability and testability,* which requires 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.

- GDC 22, *Protection system independence*, which requires 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, or be demonstrated acceptable on some other defined basis.

- GDC 23, *Protection system failure modes*, which requires that the system be designed to fail to a safe state or into a state demonstrated to be acceptable on some other defined basis, in the event of conditions such as disconnection, loss of energy, or postulated adverse environments.

- GDC 24, *Separation of protection and control systems*, which requires that interconnection of the protection and control systems be limited to assure safety in case of failure or removal from service of common components.

- GDC 25, *Protection system requirements for reactivity control malfunctions*, which requires that the reactor protection system be designed to assure that SAFDLs are not exceeded for any single malfunction of the reactivity control systems.

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

- GDC 28, *Reactivity limits*, which requires that the reactivity control systems be designed to assure that the effects of postulated reactivity accidents can neither result in damage to the reactor coolant pressure boundary (RCPB) greater than limited local yielding, nor disturb the core, its support structures, or other reactor vessel internals to significantly impair the capability to cool the core.

- GDC 29, *Protection against anticipated operational occurrences*, which requires that the protection and reactivity control systems be designed to assure an extremely high probability of accomplishing their safety functions in the event of anticipated operational occurrences.

- GDC 33, *Reactor coolant makeup*, which requires that a system to supply reactor coolant makeup for protection against small breaks in the RCPB be provided. The system safety function must assure that specified acceptable fuel design limits are not exceeded as a result of reactor coolant loss due to leakage from the RCPB and rupture of small piping or other small components that are part of the boundary. The system must be designed to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished using the piping, pumps, and valves used to maintain coolant inventory during normal reactor operation.

- GDC 34, *Residual heat removal*, which requires a system to remove residual heat be provided. The system safety function must transfer fission product decay heat and other residual heat from the reactor core at a rate such that specified acceptable fuel design limits and the design conditions of the reactor coolant pressure boundary are not exceeded.

- GDC 35, *Emergency core cooling*, which requires an emergency system to provide abundant emergency core cooling to transfer heat from the reactor core following any loss-of-coolant accident (LOCA).

- GDC 38, *Containment heat removal*, which requires that a containment heat removal system be provided and that its function must rapidly reduce the containment pressure and temperature following a LOCA and maintain them at acceptably low levels;

- GDC 41, *Containment atmosphere cleanup*, which requires systems to:

  (1) control fission products, hydrogen, oxygen and other substances, which may be released into the reactor containment be provided to reduce the concentration and quality of fission products released to the environment following postulated accidents; and
  (2) control the concentration of hydrogen or oxygen and other substances in the containment atmosphere following postulated accidents to assure that containment integrity is maintained.

- GDC 50, *Containment design basis*, which requires that the containment and its associated heat removal systems be designed so that the containment structure can accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated temperature and pressure conditions resulting from any LOCA.

- GDC 54, *Piping systems penetrating containment*, which requires piping systems penetrating containment be designed with the capability to periodically test the operability of the isolation valves to determine if valve leakage is within acceptable limits.

The following guidance documents were used in this review:

- USNRC, Office of Nuclear Reactor Regulation, "Review Standard for Extended Power Uprates," RS-001, Revision 0, dated December 2003 (Reference 3)

- NUREG-0800, "Standard Review Plan [10 CFR 50.46] for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition" (hereinafter referred to as the SRP, Reference 4), specifically:

    - SRP Section 4, in particular:
        - 4.2 "Fuel System Design"
        - 4.3 "Nuclear Design"
        - 4.4 "Thermal and Hydraulic Design"
        - 4.6 "Emergency Systems"

    - SRP Section 5, in particular:
        - 5.2.2 "Overpressure Protection"

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

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

    - SRP Section 18, Human Factors Engineering

    - SRP Section 19, in particular:
        - 19.2, Severe Accident Evaluation

    - Branch Technical Position 7-19, Guidance for Evaluation of Diversity and Defense-in-Depth in Digital Computer-Based Instrumentation and Control Systems

- Regulatory Guide (RG) 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," Revision 2, dated May 2011

- Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," dated July 2000

- NUREG-1764, "Guidance for the Review of Changes to Human Actions," Revision 1, dated September 30, 2007

- NUREG-0711, "Human Factors Engineering Program Review Model," Revision 3, dated November 2012

- The NRC Staff Requirements Memorandum (SRM) on SECY-93-087, dated July 21, 1993, 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." (Access No. ML18145A018)

- Generic Letter 94-02, "Long Term Solutions and Upgrade of Interim Operating Recommendations for Thermal-Hydraulic Instabilities in Boiling Water Reactors", September 12, 1994. (Accession No. ML031070189)

3.0 TECHNICAL EVALUATION

Licensees Approach

The proposed amendment reflects adoption of the generically approved Detect and Suppress Solution – Confirmation Density (DSS-CD) long-term reactor core thermal-hydraulic stability solution. In the BSEP MELLLA+ Safety Analysis Report (BSEP SAR - also called the M+ SAR, Enclosure 5, Reference 1), the licensee documents the results of all significant safety evaluations (SEs) performed to justify the expansion of the core flow operating domain for BSEP to the MELLLA+ domain. These analyses support operation of BSEP at the post-EPU current licensed thermal power (CLTP) of 2923 MWt with rated core flow as low as 85%. The post EPU CLTP is equivalent to 120% of the original licensed thermal power (OLTP).

The analyses in the LAR rely upon analyses using AREVA as well as General Electric-Hitachi Nuclear Energy Americas LLC (GEH) methods. AREVA recently changed its name to Framatome. For purposes herein, AREVA and Framatome should be considered synonymous. AREVA methods were applied to the reload fuel analyses, including fuel and core design, the American Society of Mechanical Engineers (ASME) and anticipated transient without scram (ATWS) overpressure evaluation, and establishing the thermal operating limits and backup stability regions. GEH methods were applied to the DSS-CD long-term stability solution (LTS) confirmatory analyses, containment response, long-term ATWS, and Anticipated Transient without Scram – Instability (ATWS-I) evaluations. These analyses are based on the methodology in:

- NEDC-33006P-A, Revision 3 "General Electric Boiling Water Reactor Maximum Extended Load Line Limit Analysis Plus" Licensing Topical Report (M+ LTR, Reference 5)

- NEDC-33173P, "Applicability of GE Methods to Expanded Operating Domains" (Methods LTR, Reference 6)

- ANP-3108P, Revision 1, "Applicability of AREVA BWR Methods to Brunswick Extended Power Flow Operating Domain" (Enclosure 12, Reference 1)

- NEDC-33075P-A, Revision 8, "General Electric Boiling Water Reactor Detect and Suppress Solution-Confirmation Density" (DSS-CD LTR, Reference 8).

All limitations and conditions (L&Cs) from the approved methodology have been addressed.

At the time of initial MELLLA+ implementation, both BSEP units will be operating with a full core of ATRIUM-10XM fuel. Implementation of the MELLLA+ operating domain extension does not require any changes to the fuel mechanical design, and BSEP will use a full core of ATRIUM-10XM fuel for future MELLLA+ cycles.

MELLLA+ is an extension of the reactor operating domain. Under MELLLA+, the operating power is maintained constant, but the recirculation core flow is allowed to operate within a wider window than under EPU. For BSEP, with the MELLLA+ flow window is between 85% and 104.5% flow. This operating flexibility reduces the need for frequent control rod motion to accommodate burnup. A secondary effect is increased fuel utilization by increased plutonium (Pu) production with increased void fraction levels, which hardens the neutron flux spectrum.

The BSEP SAR includes licensee's analyses to justify the following for BSEP:

- Implementing the MELLLA+ expanded operating domain;
- Changing the BSEP stability solution from Option III to DSS-CD;
- Applying the GEH TRACG04 analysis code to DSS-CD;
- The acceptability of AREVA ATRIUM-10XM fuel for MELLLA+ conditions; and
- The application of AREVA methods to the MELLLA+ operating domain.

Overview of the License Amendment Request

The BSEP SAR (Reference 1, Enclosure 5) is based on the M+ LTR (Reference 5), which outlines the process and scope of work required for expansion of the core flow operating range of GE boiling water reactor (BWR) plants. The BSEP SAR is divided into 11 sections:

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

The BSEP SAR also includes three appendices that evaluate the resolution of L&Cs of applicable safety evaluation reports (SEs) for:

A. Applicability of GEH Methods to Expanded Operating Domains (Methods SE, Reference 6),

B. Maximum Extended Load Line Limit Analysis Plus (M+ SE, Reference 5), and

C. GEH BWR Detect and Suppress Solution – Confirmation Density (DSS-CD SE, Reference 8).

A complete listing of the required L&Cs is presented in Appendices A, B, and C of the BSEP SAR.

Staff Method of Review

To evaluate the impact of operation in the expanded operating domain, the NRC staff performed this review using relevant sections of the review guidance in RS-001 (Reference 3), relevant sections of the SRP (Reference 4), and the findings of the NRC staff's evaluation of the M+ LTR (Reference 5).

The BSEP SAR (Reference 1, Enclosure 5) follows the same structure and content as the M+ LTR (Reference 5). The BSEP SAR resolutions evaluations topics as either "Generic" or "Plant-Specific."

- Generic assessments and plant-specific assessments are described in Section 1.1.1 and Section 1.1.2 of the BSEP SAR, respectively. The generic assessments, as defined in the M+ LTR, include generic bounding analyses, impacts that have a negligible effects, subjects where there is no change as a result of MELLLA+, and evaluations that are reload dependent. For the generic assessments, the NRC staff reviewed the assessment to ensure applicability to BSEP.

- For the plant-specific reviews, the NRC staff review is to determine whether the licensee proposal meets the regulatory criteria and, for evaluations where calculations were necessary, the appropriate input assumptions and methods were used.

The NRC staff performed this review, in part, by using relevant sections of the review guidance in NRC, Office of Nuclear Reactor Regulation, "Review Standard for Extended Power Uprates," RS-001, Revision 0 (Reference 3). Although the MELLLA+ LAR is not an EPU, and RS-001 guidance is not wholly applicable, the NRC staff determined that RS-001 provides a good framework for the review of certain portions of the LAR. The technical evaluation of the plant-specific studies was based on the guidance of review in RS-001 (Reference 3). In particular, the following reactor systems areas needed to be reviewed in detail:

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

These reactor system areas are the focus of the NRC staff's SE in the following sections.

3.1 <u>BSEP SAR Section 1.0, "Introduction"</u>

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

In the BSEP SAR, the licensee stated that it will provide fuel and cycle dependent analysis including the plant-specific thermal limits assessment. Because of the lead time required for MELLLA+ submittals, the fuel design and core loading pattern for the initial cycle of MELLLA+ were not established at the time of the MELLLA+ submittal. Therefore, the reload fuel design and core loading pattern dependent plant evaluations for MELLLA+ operation will be performed with the reload analysis as part of the standard reload licensing process. BSEP will submit the Reload Safety Analysis Report (RSAR), which is the AREVA equivalent to the GEH Supplemental Reload Licensing Report (SRLR), for the initial MELLLA+ implementation cycle for NRC staff confirmation. No plant can enter the MELLLA+ domain unless the appropriate reload core analysis is performed and all criteria and limits are satisfied, to avoid being in an unanalyzed condition. However, along with the BSEP SAR submittal, the licensee submitted reload calculations for a representative MELLLA+ cycle based on BSEP Unit 1 Cycle 19 for the NRC staff's examination, in ANP-3280P (Reference 1, Enclosure 15).

Tables 1-1 and 1-1a of the BSEP SAR list all the GEH and AREVA computer codes, respectively, used in the MELLLA+ analysis.

Figure 1-1 of the SAR (reproduced here as Figure 3.1-1) defines the MELLLA+ operating domain for BSEP. The upper boundary of the MELLLA+ domain is defined by the following relation between the percent core power (P), and the percent core flow (WT).

[[ ]]

Section 1.2.4 of the SAR describes the allowed operational enhancements, which are covered by the approved M+ SE. These operational enhancements are currently in effect in and their impact on MELLLA+ operation has been evaluated in the BSEP SAR. The following enhancements are allowed in MELLLA+ at BSEP:

- Increased core flow (ICF)
- Up to 1 safety relief valve out of service (SRVOOS)
- Turbine bypass valves out of service (TBVOOS)
- Main steam isolation valve (MSIV) out of service
- 1 Automatic depressurization system valve out of service
- 24 month cycle

The following enhancements are not allowed in the MELLLA+ domain

- Feedwater (FW) heater out of service (FWHOOS)
- Single-loop operation (SLO)

The licensee has proposed to include all allowed enhancements for BSEP MELLLA+, while excluding all non-allowed enhancements.

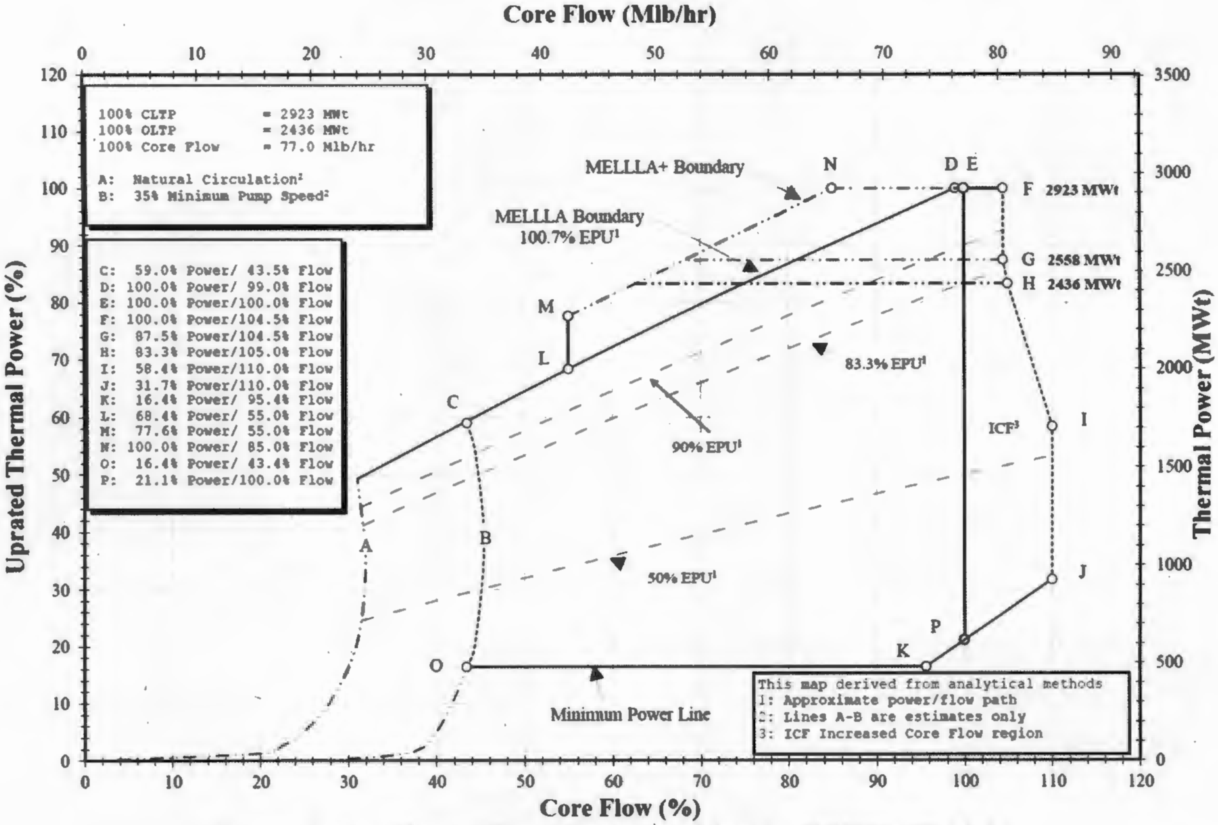

Figure 3.1-1 - Power/Flow Operating Map for BSEP MELLLA+

3.2 BSEP SAR Section 2.0, "Reactor Core and Fuel Performance"

3.2.1 BSEP SAR Section 2.1, "Fuel Design and Operation"

Regulatory Evaluation

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

(1) the fuel system is not damaged as a result of normal operation and AOOs,

(2) fuel system damage is never so severe as to prevent control rod insertion when it is required,

(3) the number of fuel rod failures is not underestimated for postulated accidents, and

(4) coolability is always maintained.

The NRC staff's review covered fuel system damage mechanisms, limiting values for important parameters, and performance of the fuel system during normal operation, AOOs, and postulated accidents. The NRC's acceptance criteria are based on 10 CFR 50.46, GDC 10, GDC 26, and GDC 35. Specific review criteria are contained in SRP Section 4.2 and other guidance provided in Matrix 8 of RS-001.

Limitations and Conditions

The Methods SE (Reference 6) and the M+ SE (Reference 5) contain L&Cs pertaining to the fuel system design. The licensee addressed these limitations in Appendix A of the BSEP SAR. The NRC staff evaluation for these limitations is given in Appendices A and B of this report.

Technical Evaluation

The NRC staff has reviewed the impact on the fuel system of the proposed MELLLA+ operating system domain based on the licensee-provided analyses results. Staff evaluation of these analyses and the results are documented in this section.

ATRIUM-10XM fuel was first introduced into both BSEP Unit 1 in Cycle 19 and Unit 2 in Cycle 20. The Unit 1 Cycle 19 core consisted of 234 fresh ATRIUM-10XM fuel assemblies and 326 irradiated ATRIUM-10 assemblies. At the time of initial MELLLA+ implementation, both BSEP units will be operating with a full core of ATRIUM-10XM fuel. Unit 1 Cycle 19 was used for the reference reload safety analyses given in ANP-3280P, Revision 1 (Enclosure 15, Reference 1). The reload safety analyses provided before each MELLLA+ cycle will reflect the appropriate core configuration including a full core of ATRIUM-10XM fuel. Implementation of the MELLLA+ operating domain extension does not require any changes to the fuel mechanical design, and BSEP will continue to use a full core of ATRIUM-10XM fuel for future MELLLA+ cycles.

The ATRIUM-10XM 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 [[ ]] full-length rods, and [[ ]] partial-length fuel rods (PLFRs). The active length of a PLFR is approximately one-half the length of a full-length rod. [[ ]] The AREVA ATRIUM-10XM fuel assembly consists of a lower tie plate, 91 fuel rods, [[ ]] spacer grids, a central water channel with [[ ]], and miscellaneous assembly hardware. [[ ]]

Mechanical design details of the AREVA ATRIUM-10XM fuel were evaluated by the NRC staff and are summarized in ANP-2948P, Revision 1 for the ATRIUM-10XM fuel transition that began in Unit 1 Cycle 19. This included the fuel rods, the fuel assembly and its components, and the fuel channel. The four objectives provided in SRP Section 4.2, which are listed in the regulatory evaluation of this section, assure the structural integrity of the ATRIUM-10XM fuel. The 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 the ATRIUM-10XM fuel design 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 MELLLA+ at BSEP.

The fuel design limits are established for all new fuel product lines as part of the fuel introduction, which allows the impact of MELLLA+ on the fuel product line to be addressed generically, as stated in the approved M+ LTR. However, the continued applicability of the thermal-mechanical fuel design limits is confirmed for each operating cycle during the reload licensing process. The establishment of cycle-specific core operating limits is addressed in BSEP TS 5.6.5; this includes cycle-specific confirmation that the ATRIUM-10XM fuel design limits for BSEP, established using the approved RODEX4 methodology, remains applicable for each reload cycle. Reload evaluations for MELLLA+ operating cycles will use MELLLA+ specific core configurations for the depletion calculations, consistent with M+ SE L&C 12.3.e.

The NRC staff reviewed the impact on the fuel system of the proposed MELLLA+ operating domain extension based on the applicant-provided analyses for normal operation, AOOs, infrequent and special events. The complete staff evaluation of these results is documented Section 3.9, "Reactor Safety Performance Evaluations." As stated in that evaluation, operation at the lower MELLLA+ flows has no impact on the response for anticipated transients because all AOOs analyzed are limiting at the 104.5% core flow condition. Furthermore, the applicant analyses demonstrate that, with the proposed BSEP MELLLA+ setpoints, fuel damage is not expected for any AOO or the analyzed infrequent or special events, and core coolability is always maintained. Thus, the NRC staff concludes that the impact on fuel of operation with the more restrictive setpoints at the lower MELLLA+ flows is minimal.

Conclusions

The NRC staff reviewed the licensee's analyses related to the effects of the proposed operating domain extension on the fuel system design of the fuel assemblies, control systems, and reactor core. The NRC staff concludes that the licensee has adequately accounted for the effects of the proposed operating domain extension on the fuel system and demonstrated that (1) the fuel system will not be 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 this, the NRC staff concludes that the fuel system and associated analyses will continue to meet the requirements of 10 CFR 50.46, GDC 10, GDC 26, and GDC 35 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.

3.2.2 BSEP SAR Section 2.2, "Thermal Limits Assessment"

Regulatory Evaluation

The NRC staff reviewed the thermal and hydraulic design of the core and the reactor coolant system (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 that 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 GDC 10 and GDC 12. Specific review criteria are contained in SRP, Section 4.4, and other guidance provided in Matrix 8 of RS-001.

Limitations and Conditions

The Methods SE (Reference 6) and the M+ SE (Reference 5) contain limitations and conditions pertaining to the fuel system design. The licensee addressed these limitations in Appendix A of the BSEP SAR. The details of the NRC staff evaluation for these limitations are discussed in the Appendices A and B of this SE.

Technical Evaluation

Section 4 of ANP-3280P, Revision 1 (Reference 1, Enclosure 15) documents the thermal hydraulic (TH) design analyses for BSEP, including the determination of the safety limit minimum critical power ration (SLMCPR), stability, and bypass voiding.

Safety Limit Minimum Critical Power Ratio (SLMCPR)

The SLMCPR is calculated based on the actual core loading pattern for each reload core, and the results will be reported in the RSAR for each reload core. In the event that the cycle-specific SLMCPR is not bounded by the current BSEP TS value, BSEP must implement a license amendment to change the TS.

Following the approved SAFLIM3D methodology in ANP-10307P, Revision 0 (Reference 13), the SLMCPR is calculated as the minimum critical power ratio (CPR) value that guarantees that <0.1 percent of the fuel rods experience boiling transition under normal operation and anticipated occurrences. 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 one traversing in-core probe (TIP) machine out-of-service (TIPOOS) or the equivalent number of TIP channels and a local power range monitor (LPRM) calibration interval of 2500 effective full-power hours average core exposure as documented in the B1C19 RSAR, ANP-3280P. 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.

Rod peaking factors and associated uncertainties are calculated by MICROBURN-B2 using the methodology given in ANP-10307P. Table 4.1 of ANP-3280 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% for TLO, ~6.0% for SLO).

Table 4.2 of ANP-3280P provides a summary of the SLMCPR calculations performed for BSEP Unit 1 Cycle 19 for MELLLA+. The conditions evaluated are statepoints F (100%CLTP/104.5%Flow), N (100%CLTP/85%Flow), and M (maximum allowed power at 55.0%Flow) of Figure 3.1-1, which bracket the operating flow conditions at high power.

Consistent with the restrictions in Section 2.2.1.1 of the M+ SE, the analysis also imposed the SLO flow uncertainties to points N and M to account for possible errors in the flow measurement at the higher void fraction conditions expected in the MELLLA+ region. In addition, an SLO condition is evaluated at 71.1%OLTP/58%Flow, which is the maximum allowed power and flow under SLO operation within the MELLLA domain. For all conditions evaluated, the number of rods in boiling transition is lower than 0.1%, with the highest number at point N (100%Power/85%Flow) with 0.097% of rods predicted to experience boiling transition, as presented in ANP-3280.

The NRC staff identified the calculations performed in ANP-3280P did not include an additional SLMCPR penalty for operating at higher power and lower flow statepoints associated with EPU and MELLLA+ that was included as part of precedent MELLLA+ applications as directed by the Methods SE L&Cs. The licensee stated that the 0.03 adder on the SLMCPR required by the Methods SE for operation in the MELLLA+ domain is not applicable to AREVA methods. The NRC staff requested additional information (RAI) in SRXB-RAI-12 for justification.

To respond to the RAI, the licensee provided seven cycles of 2D TIP uncertainty data for BSEP, with average 2D uncertainty of approximately **[[ ]]** and the highest uncertainty being approximately **[[ ]]**. This is less than the **[[ ]]** uncertainty conservatively assumed for the SLMCPR calculation for BSEP, which is taken from EMF-2158P (Reference 12). The BSEP measured uncertainties are lower primarily because of the use of gamma TIPs in BSEP, which tend to have lower uncertainties and less sensitivity to void fraction. In ANP-3108P, TIP data were presented for other plants as well. None of these TIP data show a discernible trend in uncertainty with respect to power-to-flow ratio, core average void fraction, or power; however, the Brunswick data only extend as high as approximately 39 MWt-hr/Mlb, with the majority of data below 38 MWt-hr/Mlb. Although the data for the other plants included power-to-flow ratios up to 52 MWt-hr/Mlb, few data were obtained above 42 MWt-hr/Mlb. For BSEP, 42 MWt-hr/Mlb encompasses a large portion of the MELLLA+ domain, with 52 MWt-hr/Mlb being exceeded in only a small corner of the MELLLA+ domain, which will not typically be entered during normal cycle operation.

[[

]] Therefore, the NRC staff finds it reasonable to conclude that the bundle power distribution uncertainties will not increase sufficiently at the higher MELLLA+ power-to-flow ratios to make the power distribution uncertainties in EMF-2158P inapplicable.

In Reference 10, the licensee described testing that will be performed at BSEP prior to the first cycle of MELLLA+ operation, including collection of TIP data on each unit near 100% power and 85% core flow, and near 77.6% power and 55% core flow. If the measured TIP uncertainties exceed a value representing the upper bound of the assumed TIP uncertainties at BSEP (which support the uncertainties assumed in EMF-2158P), this would be a condition adverse to quality, as defined in Appendix B to 10CFR Part 50, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants." The licensee would be required to, at a minimum, address the issue via its Quality Assurance Program.

[[

]], the low TIP uncertainties associated with the BSEP gamma TIP system, which was not credited in the BSEP SLMCPR analysis, and the planned testing the licensee will be performing during the first cycle of MELLLA+.

The result of the BSEP 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 MELLLA+ domain.

Operating Limit Minimum Critical Power Ratio (OLMCPR)

The OLMCPR is calculated by adding the change in Minimum Critical Power Ratio (MCPR) (i.e., delta-CPR) due to the limiting AOO event to the SLMCPR. The OLMCPR for BSEP is determined on a cycle-specific basis from the results of the reload transient analysis, which are documented in the RSAR.

Section 3.9.3 of this SE discusses the AOO analyses that were performed for BSEP MELLLA+, and the limiting delta-CPR value from these analyses. Demonstration that the methods are applicable to the MELLLA+ operating domain will be the basis of the approval for the OLMCPR. The limiting AOOs were load reject no bypass (LRNB) and turbine trip no bypass (TTNB), each with a delta-CPR of 0.33. This delta-CPR will be recalculated each cycle and added to the cycle-specific SLMCPR value to establish the cycle-specific OLMCPR values.

Critical Power Ratio (CPR) Correlations

For the steady state and transient analyses, the AREVA ATRIUM-10XM fuel is analyzed and monitored with the ACE critical power correlation, ANP-10298PA (Reference 14). The applicable critical power correlation for ATRIUM-10 fuel is the Siemens Power Corporation B (SPCB) critical power correlation, EMF-2209PA, Revision 3 (Reference 15). However, for the BSEP Unit 1 Cycle 19 MELLLA+ reference analyses in ANP-3280P, which included a mixed core of ATRIUM-10 and ATRIUM-10XM fuel, the critical power was only evaluated for the ATRIUM-10XM assemblies. The NRC staff finds this acceptable because these fuel types are similar and used only for the demonstration of steady state and transient analyses in MELLLA+.

The NRC staff has previously reviewed and approved the ACE critical power correlation for use with ATRIUM-10XM fuel in MELLLA+ applications. The ACE correlation has well-defined ranges of applicability that have been reviewed by the NRC staff, and include conservative actions to be applied in the event that these ranges are exceeded. The NRC staff reviewed the information and discussions provided on the AREVA methods in the expanded operating domain for BSEP MELLLA+ presented in ANP-3108P, Revision 1 (Enclosure 12, Reference 1) in Appendix E and concludes that the use of the ACE correlation is acceptable for the ATRIUM-10XM fuel in BSEP MELLLA+.

Void Fraction Correlations

The AREVA nuclear design, frequency domain stability, nuclear AOO transient, and accident analysis methods use the [[ ]] void-quality correlation, while the TH design, system AOO transient and accident analysis, and portions of the LOCA analysis use the Ohkawa-Lahey void-quality correlation. ANP-3108P, Revision 1 (Enclosure 12, Reference 1) contains an evaluation of both correlations using experimental void data from the KATHY facility, which uses a full-size AREVA ATRIUM-10XM electrically heated bundle to measure the in-channel void fraction using gamma densitometry. The calculated thermal-hydraulic conditions in the BSEP MELLLA+ core are bounded by the experimental data in terms of pressure, quality, and flow rate. Therefore, the NRC staff concludes that the use of the proposed void fraction methods is acceptable for BSEP MELLLA+ application.

3.2.3 BSEP SAR Section 2.3, "Reactivity Characteristics"

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 worth, criticality, burnup, and vessel irradiation.

The NRC's acceptance criteria are based on GDC 10, GDC 11, GDC 12, GDC 13, GDC 20, GDC 25, GDC 26 and GDC 28. Specific review criteria are contained in SRP Section 4.3 and other guidance provided in Matrix 8 of RS-001.

Limitations and Conditions

The Methods SE (Reference 6) and the M+ SE (Reference 5) contain L&Cs pertaining to the nuclear and fuel design. The licensee addressed these limitations in Appendix A of the BSEP SAR. The details of the NRC staff evaluation for these limitations are discussed in the Appendices A and B of this SE.

Technical Evaluation

Analysis for the representative core design is documented in ANP-3280P (Enclosure 15, Reference 1) for Unit 1 Cycle 19. This representative core design consists of 234 fresh ATRIUM-10XM fuel assemblies and 326 irradiated ATRIUM-10 assemblies. At the time of initial MELLLA+ implementation, both BSEP units will be operating with a full core of ATRIUM-10XM fuel.

The core design analysis has been performed using approved AREVA neutronics methodology as described in the LAR. 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 was used to model the core. Control rod patterns from Cycle 19 and the key operating parameters including thermal margins are shown in Appendix A of ANP-3013P (Reference 1, Enclosure 21). The cycle design calculations demonstrate adequate hot excess reactivity and cold shutdown margin throughout the cycle. ANP-3108P, Revision 1 (Reference 1, Enclosure 12), 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. TIP (neutron and gamma) and LPRM response models are included to compare calculated and measured instrument responses. Modern steady state THs models define the flow distribution among the assemblies. Models for the calculation of CPR, LHGR, and maximum average planar LHGR (MAPLHGR) 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 90 percent as opposed to the traditional 80 percent, which introduces a slightly larger interpolation error for intermediate void conditions. However, Figure A-14 of ANP-3108P shows good accuracy for the 0 percent, 40 percent, and 90 percent methodology for the majority of assemblies and is considered appropriate for the extended power/flow operating domain (MELLLA+) 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 12) data were reevaluated with deviations between measured and calculated TIP response at each axial level. The standard deviation of the error indicates 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 LTR EMF-2158(P)(A) (Reference 12) 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-3108P, Revision 1 (Enclosure 12, Reference 1), and determined that the methodology is capable of accurately predicting reactor conditions for fuel designs operated under current operating strategies and core conditions. Because the neutronic and TH conditions predicted for the MELLLA+ operation are bounded by the data provided in the LTR EMF-2158(P)(A), the NRC staff concludes that the isotopic validation continues to be applicable to MELLLA+ 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 MELLLA+ cycles, cycle-specific reload licensing calculations are performed using NRC- approved methodologies. The analysis presented in Appendix C.3 of ANP-3108P indicates that MELLLA+ operation in the standard power/flow map is within the range of the original methodology approval for assembly power and exit void fraction.

Fuel Assembly Design:

For Brunswick MELLLA+ operation, no fuel design modifications are necessary for both mechanical and TH characteristics. The maximum allowed enrichment level of any fuel pellet is 4.95 wt% U-235. Descriptions of fuel enrichments on both a lattice basis and an assembly basis for the first reload of ATRIUM 10XM fuel in Brunswick are listed in Table C-3 of ANP-3108P. For first reload batch, the maximum lattice enrichment is **[[** **]]** U-235 with Gd rods at **[[** **]]** Gd_2O_3. For the second reload batch, the maximum lattice enrichment is **[[** **]]** U-235 with Gd rods at **[[** **]]** Gd_2O_3.

Nuclear Design Conclusion

The NRC staff 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, thermal and hydraulic design, and transient and accident analyses, the NRC staff concludes that the nuclear design of the fuel assemblies, control systems, and reactor core will continue to meet the intent of GDC 10 and 12. Therefore, the NRC staff finds the proposed operating domain extension acceptable with respect to the nuclear design.

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 MELLLA+ operating domain 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 GDC 26 and 10 CFR 50.62(c)(4). Specific review criteria are contained in SRP, Section 9.3.5, and other guidance provided in Matrix 8 of RS-001.

Technical Evaluation:

The hot shutdown Boron weight (HSBW) is calculated on a generic basis for each fuel line (e.g., ATRIUM-10XM in the case of BSEP). The HSBW is confirmed to be effective on plant-specific and cycle-specific basis. Section 4.1 of ANP-3013P (Enclosure 21, Reference 1) documents the calculation of the HSBW, which determined a shutdown margin of 0.53 $\Delta k/k$, which is greater than the 0.4 $\Delta k/k$ limit.

The licensee requested that TS 3.1.7 be revised to increase the minimum Boron-10 enrichment from ≥47 to ≥ 92 atom percent in this LAR, consistent with the increased enrichment assumed in it safety analysis. Both the licensing bases and the best-estimate ATWS calculations, based on this increased Boron-10 enrichment, show that the generic HSBW is effective to shut down the BSEP core under MELLLA+ initial conditions.

The only change to the SLC system design is the increased Boron-10 enrichment. Since the reactor pressure was not modified and the SLC system Boron inventory shutdown margin has been evaluated for the initial core in the BSEP SAR and resulted in acceptable margin, the NRC staff finds the requirements of 10 CFR 50.62(c)(4), as well as the intent of GDC 26, continue to be satisfied.

Conclusion:

The NRC staff 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 NRC 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. Additionally, the revised TS 3.1.7 proposed to increase the minimum Boron-10 enrichment to ≥ 92 atom percent is sufficient to provide reactivity control independent of the control rod system following implementation of the proposed operating domain extension. Therefore, the NRC staff finds SLC system changes acceptable for the proposed operating domain extension.

3.2.4 BSEP SAR Section 2.4, "Stability"

Regulatory Basis

Acceptance criteria pertaining to BWR stability are based on GDC 10, GDC 12, GDC 13, GDC 20, and GDC 29 as well as Generic Letter (GL) 94-02, *Long-Term Solutions and Upgrade of Interim Operating Recommendations for Thermal-Hydraulic Instabilities in Boiling Water Reactor*s, which concerns the installation of LTS to satisfy GDC 10 and 12. SRP 15.9 (Reference 4) defines acceptance criteria acceptable to meet the relevant requirements of the above regulations. This includes guidance on acceptable decay ratios, LTS methodology, backup stability protection (BSP) implementation and other considerations relevant to protecting the SAFDLs during normal or anticipated conditions.

Applicable Limitations and Conditions

The M+ LTR (Reference 5) and associated SE approved the use of DSS-CD (Reference 8) as a long-term stability solution in the MELLLA+ operating domain. The relevant L&Cs in the M+ SE are:

- M+ SE L&C 12.2, which specifies the compliance with the L&Cs in the DSS-CD LTR SE (NEDC-33075PA, Revision 8, November 2013);

- M+ SE L&C 12.3f, which specifies the use of an approved stability method for MELLLA+ operation, and requires plant-specific demonstration that the analyses supporting the stability method are applicable to the non-GE fuel loaded in the core;

- M+ SE L&C 12.3g, which specifies the use of an approved stability protection method and approved backup stability method for MELLLA+ operation; and

- M+ SE L&C 12.7, which specifies a non-manual backup stability protection system for operation in the MELLLA+ domain.

The SE for the DSS-CD LTR identifies the following L&Cs related to DSS-CD implementation for BSEP operation in MELLLA+:

- DSS-CD SE L&C 5.1, which specifies the use of GEH Option III hardware, or a hardware review for non-GEH hardware;

- DSS-CD SE L&C 5.2, which specifies the use of the confirmation density algorithm (CDA) setpoint calculation formula and adjustable parameters as defined in the DSS-CD LTR; and

- DSS-CD SE L&C 5.3, which defines the plant-specific settings for eight FIXED parameters and three ADJUSTABLE parameters as licensing basis values, which must be addressed by the licensee.

- DSS-CD SE L&C 5.4 is not applicable because it applies to plants other than Brunswick Units 1 and 2. (Brunswick was used as the reference plant in the M+ LTR development and approval.

There is one L&C in the Methods SE pertaining to the nuclear and fuel design. The licensee addressed these limitations in Appendix A of the BSEP SAR.

- Methods SE L&C 9.18, which accounts for calibration errors due to bypass voiding when determining setpoints for any detect and suppress long-term stability methodology.

Technical Evaluation

Under some BWR operating conditions, the reactor may be susceptible to coupled neutronic and TH instabilities. These instabilities can lead to challenges to the acceptable fuel design limits and meeting the requirements of GDC 10 and GDC 12. Therefore, it is necessary for BWRs to implement a LTS solution that has the capability of automatically suppressing the instabilities. The LTS solution currently implemented in BSEP is Option III (Reference 17). Option III is not approved for use in the MELLLA+ operating domain, as stated in the M+ SE (Reference 5). Therefore, BSEP will implement the DSS-CD long-term stability solution (Reference 8), which was approved in the M+ SE (Reference 5) and DSS-CD SE (Reference 8) for operating domains up to and including MELLLA+.

The DSS-CD LTS solution is an NRC-approved LTS solution. To detect and suppress the instabilities, the DSS-CD solution uses a period based algorithm (PBA) to detect power oscillations that could potentially challenge the acceptable fuel design limits. If a significant number of oscillations is detected in a significant number of oscillation power range monitors (OPRMs) and the amplitude of the oscillations [[

]] (i.e., oscillations above the amplitude discriminator), a reactor trip signal is generated. [[

]] The DSS-CD LTR defines a nominal S_{AD} value of [[ ]] the DSS-CD LTR allows higher S_{AD} values to be selected [[

]]

The DSS-CD solution was generically demonstrated to be acceptable to automatically detect and suppress instabilities, with approved procedures specified to confirm that the DSS-CD solution remains applicable for plant and operating conditions outside of the conditions for which DSS-CD was originally approved. Therefore, is it not necessary for the NRC to review the DSS-CD design to determine if the design can suppress reactor instabilities. The NRC staff will focus its review on adequate implementation of the DSS-CD solution for the plant and operating conditions at BSEP in the MELLLA+ domain.

BSEP Implementation

As described in Section 4.0 of the DSS-CD LTR, [[

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The DSS-CD LTR generically demonstrated acceptable DSS-CD performance under the parameter ranges given in Tables 6-1 and 6-2 of the DSS-CD LTR. If a plant-specific application is outside the range of one or more of these parameters, additional calculations are required [[ ]] to confirm the acceptable performance of the DSS-CD solution for that application. [[

]]

The DSS-CD LTR has provisions for extending its applicability for fuel types that are beyond the generic licensing basis as described in Section 6.0 and Tables 6-3 and 6-5 of the DSS-CD LTR. Therefore, the licensee used the DSS-CD LTR to [[

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[[

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The extension of the applicability envelope, as discussed in Section 6.0 and Tables 6-3 and 6-5 of the DSS-CD LTR, [[

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[[

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In the RAI response, the licensee provided the justification [[

]]

The licensee proposed using plant-specific S_{AD} [[

]]

The licensee proposed a MCPR margin of greater than or equal to 0.107 for [[ ]] and 0.153 for [[ ]] as seen in Tables 2-4 and 2-5 of the BSEP SAR, respectively. [[

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[[

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[[

]] Since the DSS-CD LTR covers this process and the licensee is incorporating the DSS-CD LTR into its Section 5.6.5, "Core Operating Limits Report (COLR)," of the TSs, the NRC concludes that the licensee can sufficiently analyze the impact of future core designs with respect to LTS.

DSS-CD Methodology Change

In addition to the extension of the generic DSS-CD applicability envelope, [[

]] This parameter constitutes the minimum time period value, which may be used to generate successive confirmation counts in the PBA algorithm; if oscillations were to occur with a period less than the time period lower limit, no confirmation counts would be generated and the PBA algorithm would be unable to suppress these oscillations. In this case, the defense-in-depth algorithms included in the DSS-CD implementation (namely, the growth rate algorithm and the amplitude-based algorithm) may provide automatic trip capability; however, these algorithms are not part of the DSS-CD licensing basis and, therefore, cannot be credited to ensure that the SAFDLs are not violated.

[[

]] Such a justification would ensure that the DSS-CD licensing basis remains valid and that the SAFDLs will not be violated under any anticipated conditions in BSEP including MELLLA+ operation.

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The DSS-CD LTS is only capable of providing licensing basis SLMCPR protection if the oscillation period associated with any anticipated TH instability during plant operation remains above T_{min} and below T_{max} (the DSS-CD time period upper limit). [[

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[[

]] the NRC staff finds that a T_{min} of 1.0 sec [[
]] in providing a comparable margin [[ ]] to the expected minimum oscillation period.

Additionally, the NRC staff previously approved the position that, as stated in Section 2.6.2.3 of the Interim Methods LTR (Reference 6), "the existing GE thermal-hydraulic stability models reasonably and adequately model the magnitude and period of industry thermal-hydraulic instability events." The TH oscillation period is strongly tied to the coolant transit time through the BWR channel, which the NRC staff found in that review to be adequately represented in TRACG and which supports the NRC staff's conclusion that TRACG can reliably predict the oscillation period [[ ]]

Based on these considerations, the NRC staff concludes that a T_{min} of 1.0 sec, [[
]] provides adequate safety protection against anticipated TH oscillations in BSEP MELLLA+ and is an acceptable extension of the DSS-CD methodology for the plant-specific BSEP MELLLA+ application.

Technical Specification Updates Related to DSS-CD

The licensee provided proposed TS changes to implement the change from Option III to DSS-CD, which include updates to LCOs and updating the administrative controls section of the TSs to include a reference to the DSS-CD Methodology. The NRC staff reviewed these TS changes and found they support implementation of the DSS-CD methodology and, therefore, find the changes acceptable.

The proposed TS changes include implementation of Automated BSP (ABSP). In the event that the ABSP is not implemented per Action I of proposed TS 3.3.1.1, proposed Action J requires reduction of thermal power to below the BSP Boundary defined in the COLR, followed by restoration of the DSS-CD solution within 120 days. The licensee provided the BSP regions calculated for Unit 1 Cycle 19 of BSEP in the RSAR (Enclosure 15, Reference 1). The NRC staff reviewed these BSP regions and concluded that they were determined in accordance with the DSS-CD LTR and, therefore, finds them acceptable. Additionally, since the proposed backup stability method is an approved method and is a non-manual BSP system in the MELLLA+ domain, the proposed TS changes satisfy the M+ SE L&C 12.3g and 12.7.

The following Operating Limit and TS changes are proposed in this LAR related to instrumentation and control. The NRC staff evaluated the changes proposed to support the implementation of DSS-CD approach to automatically detect and suppress neutronic/thermal-hydraulic instabilities:

- TS 3.3.1.1, Reactor Protection System (RPS) Instrumentation

  Change Required Action I.1 from a single action to initiate an alternate method of detecting and suppressing thermal hydraulic instability to three separate actions as follows:

  I.1 Initiate action to implement the Manual BSP Regions defined in the Core Operating Limits Report (COLR). (Completion Time: Immediately)

  AND

  I.2 Implement the Automated BSP Scram Region using the modified APRM Simulated Thermal Power - High scram setpoints defined in the COLR. (Completion Time: 12 Hours)

  AND

  I.3 Initiate action in accordance with Specification 5.6.7. (Completion Time: Immediately)

  The NRC staff compared these required actions to those in the approved LTR, "GE Hitachi Boiling Water Reactor Detect and Suppress," (Reference 8) and finds these changes follow Section 8, "Effect on Technical Specifications," of the approved LTR. The required actions proposed are also the same actions provided in the Sample BWR-4 Technical Specifications in Appendix A of the Approved LTR. The sample LTR Appendix A TSs are applicable to BSEP Units 1 and 2 because both units are General Electric Type 4 BWR plants. The NRC staff finds these revisions of required actions are appropriate for safe operations in MELLLA+ domain, and therefore, acceptable.

- TS 3.3.1.1, Required Actions J.1, J.2, and J.3

  Change Required Action J.1 from one action to three, to address the situation where Required Action and associated Completion Time of Condition I is not being met. The NRC staff finds that these proposed required actions are also the same actions provided in the Sample BWR-4 Technical Specifications in Appendix A of the Approved LTR and follow Section 8, "Effect on Technical Specifications," of the approved LTR (Reference 8). The NRC staff finds these revisions of required actions are appropriate for safe operations in MELLLA+ domain, and therefore, acceptable.

  TS 3.3.1.1, Required Action K.1

  Add a new required action K.1 to address the situation where the Completion Time of Condition J is not met. The action is to reduce THERMAL POWER to less than 18% of

Rated Thermal Power and this action must be completed within 4 hours. The NRC staff finds that this proposed required action is consistent with the action provided in the Sample BWR-4 Technical Specifications in Appendix A of the approved LTR. Reducing power level to less than 18% RTP will place the plant into a condition to which LCO 3.3.1.1 does not apply for OPRM Upscale functions because the Function 2.f of Table 3.3.1-1 is only required at power levels greater than or equal to 18% RTP. This is consistent with Section 8, "Effect on Technical Specifications," of the approved LTR. The NRC staff finds this revision of required action is appropriate for safe operations in MELLLA+ domain, and therefore, acceptable.

TS 3.3.1.1, Surveillance Requirement (SR) 3.3.1.1.19

Delete this SR. The licensee stated that this requirement is no longer needed because the DSS-CD function is designed to automatically arm itself when plant conditions require it. The automatic arming functionality of the DSS-CD trip capability is described in Section 3.1 of the Approved LTR (Reference 8). This change is consistent with the Sample BWR-4 Technical Specifications in Appendix A, and with Section 8, "Effect on Technical Specifications" of the approved LTR (Reference 8). This deletion is also reflected in the proposed Table 3.3.1.1-1 Function 2.f, OPRM Upscale Surveillance Requirements. The NRC staff finds the deletion of this SR in both places is acceptable.

- TS 3.3.1.1, Table 3.3.1.1-1, Function 2.b

  Change the allowable value for Function 2.b in Table 3.3.1.1-1, "Simulated Thermal Power – High" from ≤ .55W+ 62.0% Reactor Thermal Power (RTP) to ≤ .61W+ 65.2% RTP. In addition, add a note (e) to address the OPRM Upscale function inoperable condition.

  The revised allowable value formula reflects the changed curve for determining the Simulated Thermal Power trip setpoint based on power level and core flow. These revised setpoints were calculated in accordance with the Safety Analysis Report for Brunswick Steam Electric Plant Units 1 and 2 Maximum Extended Load Line Limit Analysis + (MELLLA+ SAR), (Enclosure 5 of the BSEP MELLLA+ LAR) Section 5.3.1.

  The OPRM functions are described in the TS Basis Section 2.f, "Oscillation Power Range Monitor (OPRM) Upscale," Contained within Enclosure 4 of the LAR (Reference 1).

  The changes to the OPRM Upscale Function settings and the addition of note (e) follow the Sample BWR-4 Technical Specifications and TS Bases of Appendix A and with Section 8, "Effect on Technical Specifications," of the Approved LTR (Reference 8). The NRC staff finds that these changes are therefore acceptable.

- TS 3.3.1.1, Table 3.3.1.1-1, Function 2.f

  Change the specified condition associated with Function 2.f of Table 3.3.1.1-1 from ≥ 20% RTP to ≥ 18% RTP and add a new Footnote (f) to indicate an exception to the arming requirements of the DSS-CD function during the first reactor startup and first controlled shutdown that passes completely through the DSS-CD Armed region.

  Consistent with the deletion of Surveillance Requirement 3.3.1.19 discussed above, delete the reference in the table to SR 3.3.1.19 in the table. This Surveillance Requirement is no longer required because DSS-CD functions automatically arm when pre-determined conditions are met.

  Modify Footnote (d) to reflect the change from Period Based Detection algorithms to Confirmation Density Algorithms, which will be credited.

The change to the specified condition for Function 2.f meets the requirement that the DSS-CD must be operable above a power level 5% below the lower RTP boundary of the DSS-CD armed region. Since the lower boundary of the DSS-CD armed region is 23% as defined in TS 3.2.2, the NRC staff finds this revised condition of ≥ 18% RTP is acceptable. The NRC staff also finds the addition of Footnote (f) is consistent with the Sample BWR-4 Technical Specifications of Appendix A and with Section 8, "Effect on Technical Specifications" of the approved LTR (Reference 8) and is therefore acceptable.

- TS 3.4.1, Recirculation Loops Operating, LCO 3.4.1

  Revise this TS LCO to stipulate that single recirculating loop operation is not allowed when the reactor is in the MELLLA+ operating domain.

  The licensee's proposal is consistent with the M+ LTR, which does not allow SLO in the MELLLA+ operating domain. SLO in MELLLA+ could result in a higher power-to-flow ratio if there was a reactor pump trip which would adversely impact reactor stability. This revised LCO reinforces the TS requirement that two recirculation loops with matched flows must be in operation when the reactor is operating in the MELLLA+ region. Thus, the NRC staff finds this revised LCO acceptable.

- LCO 3.4.1, Conditions B and C

  Redesignate the current Condition B and Required Action B.1 of LCO 3.4-1 as Condition C with Required action C.1

  Add a new Condition B with Action B.1 to LCO 3.4-1, which states that Operation in the MELLLA+ domain with a single recirculation loop in operation will require an immediate action to exit the MELLLA+ operating domain.

  This revised action condition specifies that single recirculating loop operation is not allowed when the reactor is in the MELLLA+ operating domain. The NRC staff finds that this new action condition is acceptable for the same reason stated as to LCO 3.4.1 above.

- TS 5.6.5, Core Operating Limits Report (COLR), Item a.4 and Item b.19

  Replace current Item a.4 (period based detection algorithm setpoint for oscillation power range monitor) of this TS with updated documentation requirement to reflect new COLR setpoint requirements associated with the DSS-CD Long-Term Stability Solution. The NRC staff finds that is change is consistent with the Sample BWR-4 Technical Specifications of Appendix A and with Section 8, "Effect on Technical Specifications" of the approved LTR (Reference 8) and will ensure that core parameters are established consistent with the methodology. Therefore, the NRC staff concludes this replacement to be acceptable.

  Similarly, replace Item b.19 with the approved analytical DSS-CD methodology (Reference 8) which is used for the plant's long term stability solution associated with Item a.4. The NRC staff finds that the reference to the approved methodology will ensure that core parameters are established consistent with the methodology. Therefore, the NRC staff finds this replacement to be acceptable.

- TS 5.6.7, Oscillation Power Range Monitor (OPRM) Report

Add a new Technical Specification to specify when a report required by Condition I of LCO 3.3.1.1, RPS Instrumentation, shall be submitted and the required contents.

The NRC staff finds that this new Technical Specification follows the Sample BWR-4 Technical Specifications of Appendix A and with Section 8, "Effect on Technical Specifications" of the approved LTR (Reference 8). Thus, the NRC staff finds the report timing and contents appropriate for safe operations in MELLLA+ domain, and therefore, the change is acceptable.

3.2.5 BSEP SAR Section 2.5, "Reactivity Control"

"Reactivity Control" is addressed generically following the approach in the M+ LTR. The NRC staff reviewed the licensee's justification for use of the generic resolution to ensure BSEP MELLLA+ plant conditions and associated analysis falls within the bounds of generic resolution. Given that the BSEP dome pressure is unchanged for MELLLA+, the NRC staff finds that the use of the generic resolution was acceptable.

3.2.6 BSEP SAR Section 2.6, "Additional Limitations and Conditions Related to Reactor Core and Fuel Performance"

These L&Cs are addressed in Appendices A and B of this SE.

3.3 <u>BSEP SAR Section 3.0, "Reactor Coolant and Connected Systems"</u>

3.3.1 BSEP SAR Section 3.1, "Nuclear System Pressure Relief and Overpressure Protection"

The pressure relief system prevents overpressurization of the nuclear system during AOOs, the plant ASME upset overpressure protection event, and postulated ATWS events. BSEP stated that the limiting overpressure event is the Main Steam Isolation Valve Closure with Scram on High Flux. The peak RPV bottom head pressure is unchanged and remains less than the ASME limit of 1,375 pound per square inch gauge (psig). The peak RPV dome pressure is unchanged and remains less than the ASME limit of 1,325 psig.

BSEP determined that the ASME overpressure event met the acceptance criteria. Additionally, the ATWS analysis discussed in Section 9.3.1 of the SAR concludes that no increase in the number of safety relief valves (SRVs) credited in the analysis is required to demonstrate acceptable results. The ASME overpressure event continues to be analyzed each reload analysis and this requirement is unchanged by MELLLA+ operation. The NRC staff reviewed the analysis and finds that there is no change in overpressure relief capacity needed for MELLLA+ operation and finds that the licensee can adequately evaluate the event in future reload.

3.3.2 BSEP SAR Section 3.2, "Reactor Vessel"

As discussed in BSEP SAR Section 3.2.2, the licensee confirmed that the generic M+ LTR treatment of the reactor vessel structural evaluation topic is applicable to BSEP. Specifically, MELLLA+ operation does not change the reactor operating pressure, maximum FW flow, or maximum steam flow rates. As such, there is no change to the stress or fatigue for reactor vessel components.

The NRC staff concludes that the generic M+ LTR resolution is applicable to BSEP because the stress and fatigue of reactor vessel components is unaffected by operation in the MELLLA+ operating domain. Because the vessel is still in compliance with the regulatory requirements, operation with MELLLA+ does not have an adverse effect on the reactor vessel fracture toughness. These analyses show compliance with the M+ LTR SE L&C 12.8.

3.3.3 BSEP SAR Section 3.3, "Reactor Internals"

3.3.3.1 Reactor Internal Pressure Differences

3.3.3.1.1 Fuel Assembly Lift Forces

The licensee confirmed that the generic M+ LTR treatment of the fuel assembly lift forces topic is applicable to BSEP. Specifically, there are no significant changes in the core exit steam flow, reactor operating pressure, FW flow rates, or steam flow rates for MELLLA+ operation. The only variable affecting forces on the fuel assemblies in the MELLLA+ operating domain for normal, upset, emergency, and faulted conditions is the core flow. Maximum core flow is reduced in the MELLLA+ operating domain. As such, **[[**

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The NRC staff concludes that the generic M+ LTR resolution is applicable to BSEP because operation in the MELLLA+ domain is bounded by current plant operation with respect to fuel assembly lift forces.

3.3.3.1.2 Reactor Internal Pressure Differences for Normal, Upset, Emergency and Faulted Conditions

The licensee confirmed that the generic treatment in the M+ LTR for the reactor internal pressure differences (RIPDs) topic is applicable to BSEP. Specifically, there are no significant changes in the core exit steam flow, reactor operating pressure, FW flow rates, or steam flow rates for MELLLA+ operation. The only variable affecting RIPDs in the MELLLA+ operating domain for normal, upset, emergency, and faulted conditions is the core flow. Maximum core flow is reduced in the MELLLA+ operating domain. As such, **[[**

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The NRC staff concludes that the generic M+ LTR treatment is applicable to BSEP because operation in the MELLLA+ domain is bounded by current plant operation with respect to RIPDs.

3.3.3.1.3 Reactor Internals Pressure Differences (Acoustic and Flow-Induced Loads) for Faulted Conditions

As part of RIPDs, the faulted acoustic and flow-induced loads in the RPV annulus on jet pump, core shroud and core shroud support resulting from the recirculation line break LOCA have been considered in the BSEP evaluation. **[[**

]] The NRC staff concludes that the results of the BSEP specific evaluation for the RIPDs during faulted conditions are acceptable.

3.3.3.2 BSEP SAR 3.3.2, "Reactor Internals Structural Evaluation"

Structural integrity evaluations for MELLLA+ operating domain expansion are performed consistent with the existing design basis of the components. The NRC staff finds that the tabular summary discussed in Section 3.3.2 of the SAR containing Dead Weight, Seismic, RIPDS, Fuel Assembly Lift Forces, Hydrodynamic Containment Dynamic Loads - (LOCA and SRV), Annulus Pressurization, Jet Reaction and Thermal Effects is bounded by CLTP or only resulted in small increases. The NRC staff concludes that the results of the BSEP specific evaluation for the RIPDs are acceptable for the reactor internals during normal, upset, emergency, and faulted conditions as summarized in table in Section 3.3.2 of the BSEP SAR.

3.3.3.3 BSEP SAR 3.3.3, "Steam Separator and Dryer Performance"

The main purpose of steam separator and dryer is to reduce the moisture carry over (MCO) in order to protect safety related downsteam components from erosion damage. The performance of the BSEP steam separator-dryer was evaluated by the licensee to determine the moisture content of the steam leaving the RPV. Compared to the current licensed operating domain, 100% of RCF statepoint, the average separator inlet flow decreases and the average separator inlet quality increases at MELLLA+ conditions. These factors, in addition to the core radial power distribution, affect the steam separator-dryer performance. Steam separator-dryer performance was evaluated at equilibrium cycle limiting conditions of high radial power peaking and 85% of RCF to assess their capability to provide the quality of steam necessary to meet operational criteria at MELLLA+ operating conditions.

The evaluation of steam separator and dryer performance at MELLLA+ conditions indicates an increase in MCO to < 0.20 wt% where the original MCO performance specification was 0.10 wt%. BSEP SAR Section 3.3.4 identifies a plant-specific moisture performance specification under MELLLA+ operating conditions. As discussed in more detail in Appendix F of this SE on EMIB-RAI-1, the licensee provided additional justification the lack of significant impact of the higher moisture concentration on downstream components in the main steam (MS) lines, including MSIVs, SRVs, flow elements. The NRC staff concludes that the performance of the steam separator and dryer with the higher moisture concentration in MELLLA+ is acceptable because its impact on the each of the affected downstream components is discussed in Section 3.3.3.4 and found to be acceptable for operation in MELLLA+ domain.

3.3.3.4 BSEP SAR 3.3.4, "Steam Line Moisture Performance Specification"

The effect of increased MCO on plant operation has been analyzed by the licensee to verify acceptable steam separator-dryer performance under MELLLA+ operating conditions for an allowed maximum moisture content of 0.20 wt%. The moisture content of the steam leaving the RPV increases in the MELLLA+ operating domain. The effect of increasing steam moisture content has been analyzed in the tasks that use the MCO value from Sections 3.3.3 and 3.3.4. The effects of increased moisture are discussed in the following sections:

a. Flow Induced Vibration (FIV) Influence on Piping - Safety Related

The licensee stated in the LAR that:

> Because there are no safety-related MS line thermowells or sample probes, no safety-related FW line sample probes, and no safety-related RRS line sample probes, no FIV evaluations were needed to be performed for these components.

The NRC staff finds this conclusion to be acceptable because there is no safety related piping affected by operating in the MELLLA+ domain.

b. Reactor Coolant Pressure Boundary Piping

The licensee stated:

> Plant-specific evaluations of reactor coolant pressure boundary (RCPB) components and operational considerations have determined the potential increase in MCO up to 0.20 wt% is acceptable because it does not result in unacceptable design or operating margins, and the resulting increase in steam density and system pressure drop are considered to be negligible.

In the response to Mechanical Engineering and Inservice Testing (EMIB)-RAI-1, the licensee also stated that the plant's flow accelerated corrosion (FAC) program, which includes the main steam line piping, monitors susceptible areas for corrosion and factors the results into the piping replacement program at the plant site so that any adverse impact on the MSL piping will be monitored by the plant.

The NRC staff finds that the generic M+ LTR is applicable and the plant specific MCO in MELLLA+ is addressed by licensee's existing FAC program. Thus, the licensee's conclusion is acceptable.

c. Main Steam Flow - FW Flow Mismatch

The licensee stated:

> Operation at the higher MCO performance specification is acceptable. With a dryer moisture performance specification of 0.20 wt%, the additional coolant removed from the RPV must be returned to the reactor in order to maintain correct water level. The FW system will be required to provide a slightly higher flow rate. The effect of the increased MS line MCO is to cause a slight imbalance (approximately 0.4%) in the feedwater control system (FWCS) control point, which will not have a significant effect on the normal reactor water level.

The NRC staff finds the licensee's conclusion is acceptable because the compensation for this slight imbalance is within the control range of the FWCS to maintain the optimal reactor water level.

d. Liquid and Solid Waste Management

The licensee stated:

> Although the volume of waste generated is not expected to increase, potentially higher MCO in the reactor steam could result in slightly higher loading on the condensate demineralizers. Because the higher moisture content will occur infrequently, the MELLLA+ operating domain expansion will

> not cause the condensate demineralizer or the reactor water cleanup (RWCU) filter demineralizer backwash frequency to be changed significantly.

As discussed in Section 3.8.1, the NRC staff concludes that the waste volumes will not be significantly affected by the operation in the MELLLA+ operating domain. Thus, the NRC staff finds the licensee's determination acceptable.

e. Fission and Activation Corrosion Products

The licensee stated:

> Steam separator and dryer performance for MELLLA+ operation is discussed in Section 3.3.3. The moisture content of the MS leaving the vessel may increase up to 0.20 wt% at times while operating near the minimum CF [core flow] in the MELLLA+ operating domain. The distribution of the fission and activated corrosion product activity between the reactor water and steam is affected by the increased moisture content. With increased MCO, additional activity is carried over from the reactor water with the steam. The BSEP plant-specific results for the concentration of total fission products and total activated corrosion products in reactor water are bounded by the design basis concentrations.

As discussed in Section 3.8.4, the NRC staff concludes that the concentration of radiation sources in the reactor water is not expected to be significantly impacted by operating in the MELLLA+ operating domain. Thus, the NRC staff finds the licensee's determination acceptable.

f. Radiation Levels

The licensee stated:

> As discussed in Section 8.4, the moisture carry over (MCO) of the MS leaving the vessel may increase for brief periods while operating in the MELLLA+ operating domain near 100%P/85%F. However, the BSEP cycle average value will be monitored and controlled within the analytical assumption of 0.2 wt% used in the determination of normal operation radiation levels. The overall radiological effect of the increased moisture content is a function of the plant water radiochemistry and the levels of activated corrosion products.

As discussed in Section 3.8.5, the NRC staff concludes that the increase in radiation sources associated with operations in the MELLLA+ operating domain will not adversely impact the licensee's ability to maintain occupational and public radiation doses within the applicable limits in 10 CFR Part 20 and ALARA. Thus, the NRC staff finds the licensee's determination acceptable.

g. Flow Accelerated Corrosion (FAC)

The licensee stated:

> As discussed in Section 3.3.3, there is a small increase in average moisture content during short periods of the cycle. This small increase in moisture content has no significant effect on FAC parameters.

In the response to EMIB-RAI-1, the licensee also stated that the plant's flow accelerated corrosion (FAC) program, which includes the main steam line piping, monitors susceptible areas for corrosion and factors the results into the piping replacement program at the plant site so that any adverse impact on the MSL piping will be monitored by the plant.

As discussed in Section 3.10.7.2, the NRC staff finds that the generic M+ LTR is applicable and the plant specific MCO in MELLLA+ is addressed by licensee's existing FAC program. Thus, the NRC staff finds the licensee's determination is acceptable.

3.3.4 BSEP SAR Section 3.4, "Flow-Induced Vibration"

3.3.4.1 BSEP SAR Section 3.4.1, "FIV Influence on Piping"

The licensee confirmed that the generic treatment of the M+ LTR for the FIV influence on piping topic is applicable to BSEP. [[

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The NRC staff concludes that the generic M+ LTR resolution is applicable to BSEP. Because there is no increase in the flow rates in the affected piping, operation in the MELLLA+ domain is bounded by current plant operation with respect to FIV influence on piping. Thus, the NRC staff finds the licensee's determination acceptable.

3.3.4.2 BSEP SAR Section 3.4.2, "Flow Induced Vibration Influence on Reactor Internals"

The licensee confirmed that the generic treatment of the M+ LTR for the FIV influence on reactor internals topic is applicable to BSEP. Specifically, the MELLLA+ operating domain results in decreased core and recirculation flow and no increase in MS or FW flow rates. As such, there is no increase in FIV for the reactor internal components.

The NRC staff concludes that the generic M+ LTR treatment is applicable to BSEP. Because there is no increase in the flow rates in the affected piping, operation in the MELLLA+ domain is bounded by current plant operation with respect FIV influence on reactor internals.

3.3.5 BSEP SAR Section 3.5, "Piping Evaluation"

3.3.5.1 BSEP SAR Section 3.5.1, "Reactor Coolant Pressure Boundary Piping"

3.3.5.1.1 BSEP SAR Section 3.5.1.1, "Main Steam and Feedwater Piping Inside Containment"

The licensee confirmed that the generic M+ LTR treatment of the MS and FW piping inside containment topic is applicable to BSEP. Specifically, MS and FW system temperatures, flows, and pressures at MELLLA+ operating conditions are bounded by the current plant operation temperatures, flows, and pressures. As such, the parameters are within the values used in the design of the piping and supports for worst case conditions.

The NRC staff concludes that this generic M+ LTR resolution is applicable to BSEP because operation in the MELLLA+ domain is bounded by current plant operation with respect to MS and FW piping inside containment.

3.3.5.1.2 BSEP SAR Section 3.5.1.2, "Reactor Recirculation and Control Rod Drive Systems"

The licensee confirmed that the generic M+ LTR treatment of the reactor recirculation and control rod drive systems topic is applicable to BSEP. Specifically, the reactor recirculation and control rod drive system temperatures, flows, and pressures are bounded by the current plant operation temperatures, flows, and pressures. As such, the parameters are within the values used in the design of the piping and supports for worst case conditions.

The NRC staff concludes that this generic M+ LTR resolution is applicable to BSEP because operation in the MELLLA+ domain is bounded by current plant operation with respect to the reactor recirculation and control rod drive systems.

3.3.5.1.3 BSEP SAR Section 3.5.1.3, "Other RCPB Piping Systems"

As discussed in BSEP SAR Sections 3.5.1.3.1 through 3.5.1.3.4, the licensee confirmed that the generic M+ LTR treatment of the other reactor coolant pressure boundary (RCPB) piping systems topic is applicable to BSEP. Specifically, the temperatures, flows and pressures for these systems at MELLLA+ operating conditions are bounded by current plant operation temperatures, flows and pressures. As such, the parameters are within the values used in the design of the piping and supports for worst-case conditions. In addition, the susceptibility of these systems to erosion/corrosion does not change.

The NRC staff concludes that this generic M+ LTR resolution is applicable to BSEP because operation in the MELLLA+ domain is bounded by current plant operation with respect to the other RCPB piping systems.

3.3.5.1.4 BSEP SAR Section 3.5.1.4, "Other Than Category "A" RCPB Material"

The Category "A" is assumed to mean intergranular stress corrosion cracking (IGSCC) Category "A," which is a resistant material to IGSCC for BWR piping weldments in accordance with NUREG-0313 (Reference 3). The other than Category "A" material means non-resistant or cracked material to IGSCC for BWR piping weldments in accordance with NUREG-0313

(Reference 3) (IGSCC Categories B through G). As required by the M+ LTR SE Limitation and Condition 12.9, the licensee provided a discussion regarding other than Category "A" material that exist in the RCPB piping. The licensee confirmed that the generic M+ LTR treatment of other than Category "A" RCPB material topic is applicable to BSEP. The following are the key elements of licensee's discussion:

- The BSEP in-service inspection augmented inspection program is in full conformance with the ASME Section XI, Subsection IWB, IWC and IWD program for the detection and characterization of service-induced, surface connected planar discontinuities, such as IGSCC.
- The inspection ensures identification of any degradation of RCPB components during refuel outage inspections that may have initiated during MELLLA+ operation.
- BSEP has implemented stress corrosion cracking (SCC) mitigation processes, which includes component replacement and preventive measures to mitigate SCC, and inspections to monitor SCC and its effects.
- Component replacement methodologies include piping replacement with SCC-resistant stainless steel.
- Preventive measures include heat sink welding, induction heating, mechanical stress improvement, and water chemistry control in accordance with industry recognized guidelines.
- BSEP has implemented chemical mitigation technologies to address IGSCC, which are hydrogen water chemistry and Online NobleChem™ for reducing the potential for IGSCC initiation and to lower the crack growth rates of existing unrepaired relevant indications of RCPB components.
- BSEP augmented inspection program addresses the concerns related to other than Category "A" materials in the RCPB

The NRC staff concludes that the generic M+ LTR resolution is applicable to BSEP because BSEP meets all M+ LTR resolutions for Other Than Category "A" materials in the RCPB.

3.3.5.2 BSEP SAR Section 3.5.2, "Balance-of-Plant Piping"

3.3.5.2.1 BSEP SAR Section 3.5.2.1, "Main Steam and Feedwater (Outside Containment)"

The licensee confirmed that the generic M+ LTR treatment of the MS and FW outside containment topic is applicable to BSEP. Specifically, MS and FW system temperatures, flows, pressures, and mechanical loads at MELLLA+ operating conditions are bounded by the current plant operation temperatures, flows, pressures, and mechanical loads. As such, the parameters are within the values used in the design of the piping and supports for worst-case conditions. In addition, the MS and FW piping outside containment susceptibility to erosion/corrosion does not increase since their flows does not increase.

The NRC staff concludes that this generic M+ LTR resolution is applicable to BSEP because operation in the MELLLA+ domain is bounded by current plant operation with respect to MS and FW piping outside containment.

3.3.5.2.2 BSEP SAR Section 3.5.2.2.1, "Other BOP Piping Systems – RCIC, HPCI, CS, and RHR"

The licensee confirmed that the generic M+ LTR treatment of other balance of plant (BOP) piping systems (reactor core isolation cooling (RCIC), high-pressure coolant injection (HPCI), core spray (CS) and residual heat removal (RHR) systems) topic is applicable to BSEP. Specifically, RCIC, HPCI, CS, and RHR system temperatures, flows and pressures at MELLLA+ operating conditions are bounded by the current plant operation temperatures, flows and pressures. As such, the parameters are within the values used in the design of the piping and supports for worst case conditions. In addition, for each of these BSEP systems, the loads and temperatures used in the analyses continue to be bounded by the loads and temperatures performed for the current licensed operating domain.

The NRC staff concludes that this generic M+ LTR resolution is applicable to BSEP because operation in the MELLLA+ domain is bounded by current plant operation with respect to these BOP piping systems.

3.3.5.2.3 BSEP SAR Section 3.5.2.2.2, "Other BOP Piping Systems - Offgas System, Containment Air Monitoring, and Neutron Monitoring System"

The licensee confirmed that the generic M+ LTR treatment of other BOP piping systems (offgas system, containment air monitoring, and neutron monitoring system) topic is applicable to BSEP. Specifically, there is no change in the BSEP reactor operating pressure or power level at MELLLA+ operating conditions.

The NRC staff concludes that this generic M+ LTR resolution is applicable to BSEP because these BOP piping systems are unaffected by operation in the MELLLA+ operating domain.

3.3.6 BSEP SAR Section 3.6, "Reactor Recirculation System"

The licensee confirmed that the generic M+ LTR treatment of the "Reactor Recirculation System" is applicable to BSEP because the use of the generic resolution to ensure BSEP MELLLA+ plant conditions and associated analysis falls within the bounds of generic resolution. The NRC staff concludes that, since there is no expected change in MS flow, feedwater flow, core flow or operating pressure for BSEP MELLLA+, this generic resolution is acceptable and the Section 3.6 of the M+ SE evaluation is applicable to this application.

3.3.7 BSEP SAR Section 3.7, "Main Steam Line Flow Restrictors"

The licensee confirmed that the generic M+ LTR treatment of the "Main Steam Line Flow Restrictors" is applicable to BSEP because for use of the generic resolution to ensure BSEP MELLLA+ plant conditions and associated analysis falls within the bounds of generic resolution. The NRC staff concludes that, since there is no expected change in MS flow, feedwater flow, core flow or operating pressure for BSEP MELLLA+, this generic resolution is acceptable and the Section 3.7 of the M+ SE evaluation is applicable to this application.

3.3.8 BSEP SAR Section 3.8, "Main Steam Isolation Valves"

The licensee confirmed that the generic M+ LTR treatment of the "Main Steam Isolation Valves" is applicable to BSEP because for use of the generic resolution to ensure BSEP MELLLA+ plant conditions and associated analysis falls within the bounds of generic resolution. The NRC staff concludes that, since there is no expected change in MS flow, FW flow, core flow or operating pressure for BSEP MELLLA+, this generic resolution is acceptable and the Section 3.8 of the M+ SE evaluation is applicable to this application.

3.3.9 BSEP SAR Section 3.9, "Reactor Core Isolation Cooling"

The RCIC system serves as a standby source of cooling water to provide a limited decay heat removal capability whenever the main FW 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 reviewed the effect of the proposed MELLLA+ operating domain on the functional capability of the system. 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. Thus, the NRC staff concludes that the requirements of 10 CFR 50.63 and GDC 4, 5, 33, and 54 continue to be satisfied.

3.3.10 BSEP SAR Section 3.10, "Residual Heat Removal System"

The RHR system is used to cool down the RCS following shutdown. The RHR system is typically a low pressure system that takes over the shutdown cooling function when the RCS temperature is reduced. The NRC staff reviewed the effect of the proposed MELLLA+ operating domain on the functional capability of the RHR system to cool the RCS following shutdown and provide decay heat removal. 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 NRC staff concludes that GDC 4 and 5 continue to be satisfied.

3.3.11 BSEP SAR Section 3.11, "Reactor Water Cleanup System"

The licensee confirmed that the generic treatment of the M+ LTR for "Reactor Water Cleanup System" is applicable to BSEP because for use of the generic resolution to ensure BSEP MELLLA+ plant conditions and associated analysis falls within the bounds of generic resolution. The NRC staff reviewed the licensee's justification for use of the generic resolution to ensure BSEP MELLLA+ plant conditions and associated analysis falls within the bounds of generic resolution. The NRC staff concludes that, since there is no expected change in MS flow, FW flow, core flow or operating pressure for BSEP MELLLA+, the generic resolution is acceptable and the Section 3.11 of the M+ SE evaluation is applicable to this application.

3.4 <u>BSEP SAR Section 4.0, "Engineering Safety Features"</u>

Summary

All of Section 4.3, "Emergency Core Cooling System Performance," was evaluated on a plant-specific basis.

Section 4.2, "Emergency Core Cooling Systems," is addressed generically following the approach in the M+ LTR. The NRC staff reviewed the licensee's justification for use of the generic resolution to ensure BSEP MELLLA+ plant conditions and associated analysis falls within the bounds of generic resolution. The NRC staff concludes that, since there is no expected change in operating pressure, decay heat, and SRV setpoints for BSEP MELLLA+, the generic resolution is acceptable and the Section 4.2 M+ SE evaluation is applicable to this application. Note that the ECCS performance is also demonstrated in the evaluation of the design-basis events.

3.4.1 BSEP SAR Section 4.1, "Containment System Performance"

3.4.1.1 BSEP SAR Section 4.1.1, "Short-Term Containment Pressure and Temperature Response"

The purpose of the short-term analysis is to confirm that containment peak pressure and temperature does not exceed their design limits with the proposed change. The analysis is affected by any change in the mass flow rate and/or enthalpy of the break fluid. In BSEP SAR (Reference 2) Section 4.1.1, the licensee stated;

> [[
>
>]]

In SRXB-C-RAI 2, referring to Section 4.1.1 of the BSEP SAR, the licensee was requested to provide the analyzed cases for the recirculation and Main Steam Line Break (MSLB) LOCAs that formed the basis for the limiting primary containment response due to a postulated LOCA as initiated from 102% power / 85% core flow) (Figure 1-1 in BSEP SAR), MELLLA+ statepoint N). The licensee was also requested to include the calculated primary containment pressure and temperature results corresponding to the cases analyzed, and to justify that no further cases are necessary to be analyzed. In response to SRXB-C-RAI 2 (Reference 30), the licensee stated that for all BWRs with Mark I containment, the limiting break for a DBA for short-term containment pressure and temperature is the double-ended guillotine Recirculation Suction Line Break (RSLB). Therefore, [[ ]] To determine the limiting power/flow point in the MELLLA+ domain, referring to Figure 1-1 in BSEP SAR, the licensee performed sensitivity analysis for [[ ]] Table 1 below shows the sensitivity results for points N and E.

[[

]]

Table 1: Sensitivity Analysis Results for Short Term Containment Response

MELLLA+ State Point in Figure 1-1 in BSEP SAR (Reference 2)	Power (%)	Flow (%)	RSLB Short-Term Containment Pressure (psia)	RSLB Short-Term Containment Temperature (°F)
N	[[			
E				]]
[[ ]]				

In its MELLLA+ analysis, the licensee calculated the short-term peak drywell pressure to be 45.8 psig and the peak drywell temperature to be 292.3°F, both limiting for the RSLB LOCA case. The MELLLA+ short-term peak drywell pressure and temperature are bounded by their current values 46.4 psig and 293.0°F and remain below the design limits of 62 psig and 340°F respectively.

Based on the above evaluation under the MELLLA+ operating domain, the NRC staff finds that the containment continues to meet the following requirements:

(a) GDC 16 because the LOCA containment pressure and temperature transients in the MELLLA+ operating domain are bounded by the current containment pressure and temperature transients and therefore the containment will be maintained as a leak-tight barrier to a release of radioactivity to the environment; and

(b) GDC 50 because the LOCA containment pressure and temperature transients in the MELLLA+ operating domain are bounded by the current pressure and temperature transients.

The NRC staff concludes the design containment leakage rate will not be exceeded and therefore acceptable.

3.4.1.2 BSEP SAR Section 4.1.2, "Long-Term Suppression Pool Cooling Temperature Response"

The licensee provided evaluation of the long-term suppression pool temperature under CLTP conditions operating in MELLLA+ operating domain. The licensee stated:

[[

]]

Based on the above evaluation, under the MELLLA+ operating domain the containment system continues to meet the requirements of GDC 38 because the containment heat removal system rapidly reduces the containment pressure and temperature following a LOCA and maintains them at acceptable levels.

The NRC staff concludes the licensee's evaluation acceptable, because the generic resolution in the M+ LTR is applicable to BSEP.

3.4.1.3 BSEP SAR Section 4.1.3, "Containment Dynamic Loads"

3.4.1.3.1 BSEP SAR Section 4.1.3.1, "Loss-of-Coolant Accident Loads"

The M+ LTR requires plant-specific evaluation to determine the effect of MELLLA+ operating domain expansion on the LOCA containment dynamic loads. These loads include vent thrust, pool swell, condensation oscillation (CO), and chugging loads as defined in the generic Load Definition Report (LDR) NEDO-21888 Revision 2 (Reference 41) for Mark I containments approved by NRC in NUREG-0661 (Reference 8). The BSEP Units 1 and 2 plant-specific loads are defined in the Plant Unique Load Definition report NEDO-24582, Revision 1 (Reference 42).

Vent Thrust Loads

[[**]]** Using the methodology in the LDR (Reference 41), the licensee **[[** **]]** in the MELLLA+ operating conditions using the NRC accepted M3CPT (Reference 45) computer code for the short-term response. **[[** **]]** The NRC staff concludes that the current LOCA vent thrust load definitions remain applicable because they bound the calculated loads in the MELLLA+ operating domain.

Pool Swell Loads

The BSEP Units 1 and 2 plant-specific pool swell loads are defined in NEDE-21944-P Volume 1 (Reference 43), which are based on a Quarter Scale Test Facility plant unique test. These loads depend on the **[[** **]]**, and the load definition is based on a DBA-LOCA which is the **[[** **]]**. Therefore, the NRC staff finds that the current pool swell load definition would be applicable in the MELLLA+ operating domain because the **[[** **]]**

Condensation Oscillation Loads

The DBA-LOCA Condensation Oscillation (CO loads occur due to the oscillation of the steam-water interface that forms at the vent exit, during the period of high steam mass flow rate. These loads occur after the pool swell phenomena. The basis for the Mark I CO load definition is the LDR (Reference 41). In Section 4.1.1 of BSEP SAR under heading "Condensation Oscillation Loads" the licensee stated:

> The Mark I containment CO load definition was developed from test data from Full Scale Test Facility (FSTF) tests (Reference 44) to simulate LOCA thermal-hydraulic conditions (i.e., **[[**
> **]]**). The tests are bounding for all US Mark I plants, including the BSEP, considering MELLLA+ conditions.

In SRXB-C-RAI 3, referring to the above statement, the licensee was requested to explain why the FSTF tests (Reference 44) results are bounding for the BSEP, Units 1 and 2 CO loads in the MELLLA+ operating domain. In response to SRXB-C-RAI 3 (Reference 30), the licensee explained that the **[[**

]]

[[

]] The NRC staff reviewed licensee's explanation and results and finds the condensation oscillation loads acceptable because **[[**
]].

Chugging Loads

Chugging begins after the CO phenomena (i.e., **[[**

]] The current chugging loads are in accordance with LDR (Reference 6) and the FSTF tests (Reference 44). The licensee stated that these load definitions remain applicable under the MELLLA+ conditions because the thermal-hydraulic conditions for these tests (i.e., **[[** **]]** were selected to produce maximum chugging amplitudes so that it bound all Mark I containment plants.

In SRXB-C-RAI 4, the licensee was requested to explain why the FSTF tests (Reference 44) results are bounding for the BSEP Units 1 and 2 chugging loads in the MELLLA+ operating domain. In response to SRXB-C-RAI 4 (Reference 30) the licensee explained that **[[**

]] Regarding the chugging test program the licensee stated that the Mark I containment test program **[[**

]]

The NRC staff finds the licensee's response for the LOCA chugging loads acceptable because the [[

]]

The results of the plant-specific LOCA containment dynamic loads evaluation demonstrate that existing vent thrust, pool swell, CO, and chugging load definitions remain bounding for operation in the MELLLA+ operating domain. Therefore, the NRC staff finds it acceptable that the LOCA containment dynamic loads defined for BSEP Units 1 and 2 are not affected by the MELLLA+ operating domain.

Based on the above evaluation, the containment under the MELLLA+ operating domain continues to meet the requirements of GDC 4 because the LOCA dynamic loads on safety-related containment structures and components, are bounded by the current LOCA dynamic loads on the safety-related containment structures and components.

3.4.1.3.2 BSEP SAR Section 4.1.3.2, "Subcompartment (Annulus) Pressurization"

BSEP UFSAR Section 6.2.1.2.3 states that the annular region between the RPV and the sacrificial shield is the only subcompartment where pressurization can take place due to a LOCA within that region. The sacrificial shield is a welded, cylindrical, structural steel frame consisting of H-shaped and box beams and columns. Steel plates line the inside and outside surfaces of the structure and act as a form for fill concrete that is provided for shielding purposes. The breaks postulated to be analyzed in the annular region are the recirculation suction line and FW line breaks. The licensee stated that guard pipes installed (in 1980) on the recirculation suction piping would force the released mass from the break directly into the drywell. With this modification in place, all safety related components of the RPV and its supports, and reactor internals were evaluated to withstand a combination of design basis earthquake and annulus pressurization events resulting from recirculation or FW piping breaks, and normal loads without exceeding allowable stresses. All subsequent plant changes including FW heaters out-of-service, stretch power uprate, and EPU also confirmed acceptable loads for the analyzed breaks and met the original analysis basis. In the MELLLA+ operating domain, the FW and recirculation line breaks does not change the annulus pressurization loads because the maximum reactor pressure, steam flow, and FW flow are not increased. The licensee stated that even though the MELLLA+ annulus pressurization loads are bounded by the loads in the MELLLA operating domain with reduced FW temperature, the reduced FW temperature operation is not allowed in the MELLLA+ operating domain.

Based on the above evaluation of the recirculation and FW line break LOCA annulus pressurization loads, the NRC staff finds that the containment continues to meet the requirements of GDC 4 under the MELLLA+ operating domain, because the dynamic loads due to LOCA subcompartment pressurization are bounded by the current loads. Thus, the NRC staff concludes the subcompartment (annulus) pressurization is acceptable.

3.4.1.3.3 BSEP SAR Section 4.1.3.3, "SRV Piping - Containment Dynamic Loads"

The Safety/Relief Valve (SRV) and its piping loads depend on the SRV setpoints, reactor sensible heat, and decay heat. For the BSEP, these parameters do not change from the current operating domain to the MELLLA+ operating domain. Therefore, the NRC staff finds that the SRV and SRV piping loads are not affected in the MELLLA+ operating domain.

Based on the above evaluation, the NRC staff concludes that under the MELLLA+ operating domain the containment continues to meet the requirements of GDC 4 because the SRV piping loads are bounded by the current SRV piping loads and therefore acceptable.

3.4.1.3.4 BSEP SAR Section 4.1.3.4, "SRV Containment Dynamic Loads"

The generic resolution in the M+ LTR (Reference 4) Section 4.1 states:

> **[[**
>
> **]]**

For BSEP Units 1 and 2, the generic resolution is applicable because the reactor thermal power, dome pressure, and SRV setpoints do not change from the MELLLA to MELLLA+ operating domain.

Based on the above evaluation, the NRC staff concludes that under the MELLLA+ operating domain the containment continues to meet the requirements of GDC 4 because the generic resolution in the M+ LTR is applicable, which states that operation in MELLLA+ domain does not affect the SRV discharge loads on the containment.

3.4.1.4 BSEP SAR Section 4.1.4, "Containment Isolation"

The generic resolution in the M+ LTR (Reference 4), Section 4.1.4 states;

> **[[**
>
> **]]** a plant-specific evaluation is required to demonstrate the adequacy of the containment isolation system.

As stated in Section 4.1.4 of the BSEP SAR, the MELLLA+ **[[**
]]
therefore further evaluation of the containment isolation systems is not required; the NRC staff concludes that the BSEP SAR evaluation is acceptable in the MELLLA+ operating domain.

3.4.1.5 BSEP SAR Section 4.1.5, "Generic Letter 89-10"

The evaluation under Generic Letter (GL) 89-10 Supplement 3, "Consideration of the Results of NRC-Sponsored Tests of Motor-Operated Valves" (Accession No. ML031200576), would be affected by the containment pressure and temperature under DBA conditions. The MELLLA+ operating domain does not impact the current evaluation under GL 89-10 **[[**

]] The licensee also confirmed that other parameters such as environment temperature during normal conditions and under high energy line break conditions, that could potentially affect the safety-related motor-operated valves (MOVs), are not changed in the MELLLA+ operating domain. Therefore, NRC concludes that the current evaluation under the GL 89-10 is acceptable in the MELLLA+ operating domain.

3.4.1.6 BSEP SAR Section 4.1.6, "Generic Letter 89-16"

In response to GL 89-16 "Installation of a Hardened Wetwell Vent" (Accession No. ML031140220), a hardened wetwell vent system is installed in BSEP Units 1 and 2. The requirement for the hardened wetwell vent is the ability to exhaust energy from the containment equivalent to 1-percent of the CLTP. Since the reactor thermal power does not change in the MELLLA+ operating domain, therefore, the NRC staff concludes that the current response to GL 89-16 is acceptable.

3.4.1.7 BSEP SAR Section 4.1.7, "Generic Letter 95-07"

The evaluation under the GL 95-07, "Pressure Locking and Thermal Binding of Safety-Related Power-Operated Gate Valves" (Accession No. ML003674456), would be affected by the containment pressure and temperature under DBA conditions. The MELLLA+ operating domain does not impact the current evaluation under GL 95-07 [[

]] Therefore, the NRC staff concludes that the current evaluation under GL 95-07 is acceptable in the MELLLA+ operating domain.

3.4.1.8 BSEP SAR Section 4.1.7, "Generic Letter 96-06"

The evaluation under the GL 96-06, "Assurance of Equipment Operability and Containment Integrity during Design-Basis Accident Conditions" (Accession No. ML022550116), would be affected by the containment pressure and temperature under DBA conditions. The MELLLA+ operating domain does not impact the current evaluation under GL 96-06 [[

]] Therefore, the NRC staff concludes that the current evaluation under GL 96-06 is acceptable in the MELLLA+ operating domain.

3.4.2 BSEP SAR Section 4.2, "Emergency Core Cooling System"

Section 4.2, "Emergency Core Cooling Systems," is addressed generically following the approach in the M+ LTR. The NRC staff reviewed the licensee's justification for use of the generic resolution to ensure BSEP MELLLA+ plant conditions and associated analysis falls within the bounds of generic resolution. The NRC staff concludes that, since there is no expected change in operating pressure, decay heat, and SRV setpoints for BSEP MELLLA+, the generic resolution is acceptable and the Section 4.2 M+ SE evaluation is applicable to this application. Also note, that the ECCS performance is discussed in Section 3.4.3 of this SE.

Section 4.2.1 through Section 4.2.5, "Emergency Core Cooling Systems," are addressed generically following the approach in the M+ LTR. The NRC staff reviewed the licensee's justification for use of the generic resolution to ensure BSEP MELLLA+ plant conditions and associated analysis falls within the bounds of generic resolution. The NRC staff concludes that, since there is no expected change in operating pressure, decay heat, and SRV setpoints for

BSEP MELLLA+, the generic resolution is acceptable and the Sections 4.2.1 through 4.2.5 M+ SE evaluation is applicable to this application. The ECCS performance is also discussed in Section 3.4.3 of this SE.

3.4.2.1 BSEP SAR Section 4.2.6, "ECCS Net Positive Suction Head"

The ECCS and containment heat removal pumps are the Residual Heat Removal (RHR) and Core Spray (CS) pumps. MELLLA+ does not result in an increase in core power and decay heat or heat addition to the suppression pool during a LOCA, Station Blackout (SBO), or National Fire Protection Association (NFPA)-805 Fire event that would affect the its temperature response. There are no physical changes in the RHR and CS system piping or arrangement that could impact the net positive suction head (NPSH) analysis for their pumps. Also, there is no change in the operator actions for throttling the RHR and CS flows during a LOCA and special events. In the current licensing basis, BSEP Units 1 and 2 credit Containment Accident Pressure (CAP) for calculating the available NPSH for these pumps. As mentioned in the SE (Reference 38) for the EPU, the NRC approved a CAP credit of up to 5.0 psig for the long term (i.e., greater than 600 sec) NPSH analysis from the available post-LOCA CAP of 11.3 psig. No CAP credit was requested or approved by the NRC for the short term (i.e., 0 to 600 sec) post-LOCA NPSH analysis in the current licensing basis.

For the mitigation of ATWS event in the MELLLA+ operating domain, the licensee increased the SLC system Boron-10 enrichment from ≥47 atom percent to ≥92 atom percent. The licensee stated this modification would shut down the reactor faster during an ATWS event, and increase the time for the suppression pool to reach its heat capacity temperature limit (HCTL) because of decrease in its heat load. Therefore, the suppression pool temperature response will be reduced, which would increase the NPSH margin of the RHR pump used for cooling of the suppression pool.

The NRC staff finds that the NPSH margin of the RHR and CS pumps are not adversely affected during a LOCA, SBO, and NFPA-805 Fire events. The NRC staff therefore finds it acceptable that operating in the MELLLA+ domain does not impact the ECCS pumps NPSH analysis.

Based on the above evaluation, the NRC staff concludes that the ECCS and the containment heat removal system pumps continue to meet the requirements of GDC 34, 35, and 38 under the MELLLA+ operating domain, because the RHR pumps will have adequate NPSH to perform their safety function for core cooling and containment heat removal during a LOCA, SBO, ATWS and NFPA-805 Fire events. The NRC staff also concludes that the CS pump NPSH analysis continues to meet the GDC 35 requirement because the plant will have adequate NPSH for core cooling during a LOCA.

3.4.3 BSEP SAR Section 4.3, "Emergency Core Cooling System Performance"

The BSEP ECCS is designed to provide protection against postulated LOCAs caused by ruptures in the primary system piping. Successful ECCS performance is documented in Section 4.3 of the BSEP SAR. The evaluation model used for LOCA analysis is the EXEM BWR-2000 Evaluation Model, which is a collection of the following codes and methods:

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

- XN-CC-33(A) Revision 1, "HUXY: A Generalized Multirod Heatup Code with 10 CFR 50 Appendix K Heatup Option User's Manual"

- XN-NF-82-07(P)(A) Revision 1, "Exxon Nuclear Company ECCS Cladding Swelling and Rupture Model"

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

The methodology used is unchanged from the analysis of record methods. The NRC staff reviewed Enclosures 18 and 24 of Reference 1 to ensure that the methodology was appropriately applied for the BSEP MELLLA+ application. Additionally, the NRC staff reviewed the application to ensure compliance with the following L&Cs:

- 9.7 of the Methods LTR
- 9.8 of the Methods LTR
- 12.3.a of the M+ LTR
- 12.10.a of the M+ LTR
- 12.10.b of the M+ LTR
- 12.10.c of the M+ LTR
- 12.10.d of the M+ LTR
- 12.11 of the M+ LTR
- 12.12.a of the M+ LTR
- 12.12.b of the M+ LTR
- 12.13 of the M+ LTR
- 12.14 of the M+ LTR

These L&Cs are addressed in Appendices A and B of this SE.

The NRC staff reviewed the LOCA calculations and determined that the methodology was adequately applied, the break spectrum was sufficient, and the appropriate setpoints were used in the analysis. The results are listed in the BSEP SAR in Section 4.3.1 for peak cladding temperature (PCT), Section 4.3.2 for local cladding oxidation, Section 4.3.3 for core wide oxidation, and Section 4.3.4 for coolable geometry demonstrate that the 10 CFR 50.46 requirements are achieved. Thus, the NRC staff finds the analysis acceptable.

The licensee analyzed the impact of fuel thermal conductivity degradation (TCD) in Appendix F of Enclosure 12 of Reference 1. The RODEX2 code used in the LOCA analysis does not account for the effects of TCD. To assess the TCD impact, the licensee first compared the results of RODEX2 and more recent RODEX4 code, which accounts for TCD. The licensee then adjusted the RODEX2 input, based on the differences between the code results, and reran the LOCA calculations. The results of TCD are shown in Table F-1 of Enclosure 12 of Reference 1 and show that there is a negligible TCD impact on the limiting case. This approach to address TCD was previously reviewed by the NRC staff and the licensee has not taken a deviation from the previous analysis. Therefore, the NRC staff concludes that the effects of TCD on the LOCA analysis are acceptable for MELLLA+ operations.

3.4.4 BSEP SAR Section 4.4, "Main Control Room Atmosphere Control System"

The main control room atmosphere control system under MELLLA+ operating domain would be affected by the increase in the radiation source term. The licensee stated that the MELLLA+ operating domain expansion does not result in a change in the source term or the release rate. [[ ]]

Based on the above evaluation, the requirement of GDC 19 continues to be met because the main control room atmosphere control system that is unaffected will provide adequate radiation protection to the personnel accessing and occupying the main control room under accident conditions in the MELLLA+ operating domain. Therefore, the NRC staff concludes that the main control room atmosphere control system remains acceptable for the MELLLA+ operating domain.

3.4.5 BSEP SAR Section 4.5, "Standby Gas Treatment System"

The Standby Gas Treatment System (SGTS) maintains the secondary containment at a negative pressure and filters the exhaust air by removing fission products present during abnormal conditions. The parameters that could be affected for operation in the MELLLA+ operating domain are the SGTS flow capacity and its iodine removal capability.

In the MELLLA+ operating domain, the [[ ]]

In the MELLLA+ operating domain, the SGTS iodine removal capacity is not affected because the core fission product inventory is not changed. Also there is no change in the adsorber iodine loading, decay heat, or iodine removal efficiency. [[ ]]

Based on the above evaluation, the GDC 16 requirements for the secondary containment functional design are met because the capability of the SGTS for depressurizing the secondary containment, maintaining it at the required negative pressure, and its ability to remove fission products is unaffected in the MELLLA+ operating domain under post-LOCA conditions. The NRC staff concludes that the GDC 41 requirement to reduce the concentration and quality of fission products released to the environment following postulated accidents is met because the SGTS is unaffected in the MELLLA+ operating domain and is therefore acceptable.

3.4.6 BSEP SAR Section 4.6, "Main Steam Isolation Valve Leakage Control System"

BSEP does not use a MSIV leakage control system.

3.4.7 BSEP SAR Section 4.7, "Post-LOCA Combustible Gas Control System"

The NRC revised 10 CFR 50.44, "Combustible gas control for nuclear power reactors," rule in September 2003. The revised rule eliminated the requirements for maintaining hydrogen and oxygen control equipment associated with a design-basis accident, and relaxed the requirements for hydrogen and oxygen monitoring in containment. License Amendment Nos. 234 and 261 for BSEP Units 1 and 2, respectively (Reference 39) eliminated the requirements for the hydrogen and oxygen monitors. The requirements for the Containment Atmospheric Dilution (CAD) system were eliminated by License Amendment Nos. 252 and 280 for BSEP Units 1 and 2, respectively (Reference 40). The design purpose of the CAD system was to maintain combustible gas concentrations in the primary containment at or below the flammability limits following a postulated LOCA by diluting hydrogen and oxygen with the addition of nitrogen. As a result of the revised rule, the CAD system was no longer required to be maintained as combustible gas control system. Since the combustible gas control is not required in the MELLLA operating domain, and there is no change in the core power, decay heat or fuel design, therefore there is no impact on the combustible gas control in the MELLLA+ operating domain.

Based on the above evaluation, the NRC staff concludes that the combustible gas control system meets the requirements of 10 CFR 50.44 because the containment is inerted during normal operation in the MELLLA+ operating domain, therefore an uncontrolled combination of hydrogen and oxygen will not take place in the containment during post-LOCA conditions. Thus, the GDC 41 requirement to the control of hydrogen or oxygen concentration in the containment following postulated accidents to assure that containment integrity is maintained.

3.5 BSEP SAR Section 5.0, "Instrumentation and Control"

3.5.0. *Overall Diversity and Defense-In-Depth Evaluation*

BSEP will be implementing DSS-CD core monitoring functions using the installed Nuclear Measurement Analysis and Control (NUMAC) Power Range Neutron Monitoring (PRNM) system. The NUMAC PRNMS system is a digital safety system that is potentially susceptible to software common cause failures (CCFs) that could adversely affect the system's ability to perform core stability monitoring and associated reactor scram functions. The licensee performed a D3 analysis to verify vulnerabilities to a CCF of the NUMAC system have been adequately addressed to demonstrate compliance with 10 CFR 50.55a and with GDC 23, 25, and 29. The NRC staff evaluated the D3 analysis provided by the licensee in accordance with SRP BTP 7-19.

The NUMAC PRNM system includes the OPRM function, which uses the MELLLA-Plus domain with the DSS-CD stability solution. The DSS-CD performs three algorithms for detecting thermal-hydraulic instability related neutron flux oscillations. The OPRM performs a Period Based Algorithm (PBA) reactor trip safety function, which is credited in the BSEP UFSAR Chapter 7.6.1.1.4 "Average Power Range Monitor Subsystem" for mitigation of a plant instability event. It states:

> The design of the APRM subsystem shall be such that for the worst permitted input LPRM bypass and failure conditions, the APRM shall be capable of generating a scram trip signal in response to local neutron flux oscillations resulting from a thermal-hydraulic instability in time to prevent fuel damage.

The other two OPRM algorithms are Amplitude Based Algorithm and Growth Based Algorithm. These algorithms are not credited safety functions, but are included as defense-in-depth features of the system. The PBA function is used to demonstrate protection of MCPR safety limit for anticipated reactor instabilities. A failure of the NUMAC OPRM or APRM could disable the automatic safety trip function performed by the DSS-CD algorithms since both of these subsystems are relied upon to ensure operability of the DSS-CD algorithms. The BSEP NUMAC system includes a means of providing Automatic Backup Stability Protection (ABSP) in the event that the primary means of stability protection (DSS-CD) becomes inoperable. However, the NRC staff notes that both primary (DSS-CD) and backup (ABSP) stability protection use common software, which can lead to a condition where both of these automatic functions would become disabled. A postulated software defect could be triggered and result in a CCF of the OPRM reactor trip safety function (in the PRNM system).

If the OPRM system is inoperable and the ABSP function performed by the APRM either cannot be implemented or is also inoperable, manual Backup Stability Protection (BSP) becomes the licensed stability solution. The BSEP power – core flow graph contains regions of operation that are defined by a BSP boundary (Region I as outlined in BSEP TS 3.3.1.1 and defined in the COLR). With the BSP boundary being the credited stability solution, the reactor power is reduced below the BSP line so that two recirculation pump trips will not result in immediate operation inside the exclusion region. When plant conditions exceed this BSP scram region boundary, administrative actions require initiation of a manual reactor scram. This is described in Section 7 and in the Technical Specification (TS) changes documented in the approved DSS CD Licensing Topical Report (Reference 8) and in BSEP TS 3.3.1.1 Condition I.

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 CCF of the NUMAC Power Range Neutron Monitoring (PRNM) (APRM/OPRM) system in conjunction with the plant instability events described in the BSEP UFSAR. The results of this analysis were provided in Section 2.4.1.1 of Enclosure 5 of the LAR (Reference 1). This analysis identified Manual Operator Actions as diverse means of maintaining plant safety if the automatic trip functions performed by the DSS-CD algorithms and the ABSP become unavailable due to a postulated common-mode failure of the NUMAC PRNM system.

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

In its previous evaluation of BSP protection (Reference 8), the NRC staff concluded the proposed BSP methodology is an acceptable solution, because it provides sufficient protection against plant Safety Limit Minimum Critical Power Ratio (SLMCPR) violations commensurate with the probability of an instability event in the short period of time they are active. The NRC staff evaluation further concluded the manual control measures needed to support BSP protection are sufficiently diverse from the digital PRNMS NUMAC systems and therefore provide an acceptable means of diverse protection for the DSS-CD safety function.

The BSEP D3 analysis identifies **[[**

]], BSEP operators are procedurally required to reduce reactor power level to mitigate high growth rate power oscillations. Operators will be aware of this situation because flow information is available from the recirculation flow system, and power level information is available from either the electrical power output or a core thermal power calculation. These indications are independent from the PRNM system instruments and would not be affected by the CCF failure of the PRNM system. This immediate action is uncomplicated and instrumentation necessary for completion of this action are not affected by the CCF. Control room operators can also confirm that manual actions are successful based on control panel indications that are independent from the PRNM system. Further confirmation is provided by information available on the plant process computer.

The licensee's evaluation provided in Section 2.4.1.1 of Reference 1 confirmed that Recirculation Flow, Rod Position Indication, Reactor Manual Control, and Manual Scram systems used for initiation of the power reduction, and for confirmation that the power reduction was successful do not rely on NUMAC based technology. Thus, the NRC staff finds that systems needed for mitigation of oscillations would not be affected by a postulated software CCF that renders the automatic protection functions inoperable. **[[** **]]**, the credited diverse manual operator actions are operations that control **[[** **]]** on the power to flow map. These include adjustment of recirculation flow and control rod positions. The D3 analysis identified multiple diverse control room indications **[[** **]]** that are independent from the effects of the postulated PRNM system CCF.

Therefore, the NRC staff finds the systems to be relied upon to maintain plant safety would not be affected by a postulated software CCF of the PRNMS that renders the automatic protection functions inoperable, and that acceptable means of diverse protection are provided in accordance with the guidance of BTP 7-19.

The following is a brief summary of the licensee's generic MELLLA+ resolution for the BSEP SAR sub-topics:

3.5.1 BSEP SAR Section 5.1, "NSSS [Nuclear Steam Supply System] Monitoring and Control"

Section 5.1 of the BSEP SAR describes changes to process parameters resulting from the MELLLA+ operating domain expansion and their effects on instrument performance. These change evaluations include; Average Power Range, Intermediate Range, and Source Range monitors, Local Power Range Monitors, Rod Block Monitor, Rod Worth Minimizer and Traversing Incore Probes.

The licensee stated that there is no change in BSEP core power as a result of MELLLA+ operating domain expansion. The NRC staff reviewed the licensee's justification for this position and determined that, since MELLLA+ operating domain expansion does not change BSEP core power, the generic resolution is acceptable and Sections 5.1.1 through 5.1.5 MELLLA+ LTR evaluation are applicable.

3.5.1.1 BSEP SAR Section 5.1.1, "Average Power Range, Intermediate Range, and Source Range Monitors"

The licensee confirmed that the generic M+ LTR treatment of the APRMs, intermediate range monitors (IRMs), and source range monitors (SRMs) topic is applicable to BSEP. The APRM output signals are calibrated to read 100 percent at the CLTP. **[[** **]]** The IRMs may be adjusted to ensure adequate overlap with the SRMs and APRMs. [[

]]

The NRC staff concludes that the generic M+ LTR resolution is applicable to BSEP because there is no change in BSEP core power as a result of MELLLA+ operating domain expansion.

3.5.1.2 BSEP SAR Section 5.1.2, "Local Power Range Monitors"

The licensee confirmed that the generic M+ LTR treatment of the local power range monitors (LPRMs) topic is applicable to BSEP. There is no change in the neutron flux experienced by the LPRMs resulting from operating in the MELLLA+ domain. As such, [[

]]

The NRC staff concludes that the generic M+ LTR resolution is applicable to BSEP because [[ ]]

3.5.1.3 BSEP SAR Section 5.1.3, "Rod Block Monitors"

The licensee confirmed that the generic M+ LTR treatment of the rod block monitors (RBM) topic is applicable to BSEP. The RBM uses LPRM instrumentation inputs that are combined and referenced to an APRM channel. [[
]]

The NRC staff concludes that the generic M+ LTR resolution is applicable to BSEP because [[ ]]

3.5.1.4 BSEP SAR Section 5.1.4, "Rod Worth Minimizer"

The licensee confirmed that the generic M+ LTR treatment of the rod worth minimizer (RWM) topic is applicable to BSEP. The BSEP RWM supports the operator by enforcing rod patterns until reactor power has reached appropriate levels. [[
]]

The NRC staff concludes that the generic M+ LTR resolution is applicable to BSEP because [[ ]]

3.5.1.5 BSEP SAR Section 5.1.5, Traversing Incore Probes

The licensee confirmed that the generic M+ LTR treatment of the traversing incore probes (TIPs) topic is applicable to BSEP. There is no change in neutron flux experienced by the TIPs by MELLLA+ operation.

The NRC staff concludes that the generic M+ LTR resolution is applicable to BSEP because the TIPs are unaffected by operation in the MELLLA+ operating domain.

3.5.2 BSEP SAR Section 5.2, "BOP Monitoring and Control"

As discussed in BSEP SAR Sections 5.2.1 through 5.2.6, the licensee confirmed that the generic M+ LTR treatment of the balance-of-plant (BOP) monitoring and control topic is applicable to BSEP. Operation of the plant in the MELLLA+ domain has no effect on the BOP instrumentation and control devices because [[ ]]

The NRC staff concludes that the generic M+ LTR resolution is applicable to BSEP because BOP monitoring and control devices are unaffected by operation in the MELLLA+ operating domain.

3.5.3 BSEP SAR Section 5.3, "Technical Specification Instrumentation Setpoints"

Section 5.3 of the BSEP SAR describes changes to instrumentation setpoints resulting from the MELLLA+ operating domain expansion. The instrumentation setpoints evaluated are associated with the APRM Flow-Biased Scram and Rod Block Monitor functions.

3.5.3.1 BSEP SAR Section 5.3.1, "APRM Flow-Biased Scram"

The licensee confirmed that the generic M+ LTR treatment of the APRM Flow-Biased Scram topic is applicable to BSEP. [[ ]]

The NRC staff concludes that the generic M+ LTR resolution is applicable to BSEP because [[ ]] The NRC staff also notes that Single Loop Operation is not applicable to the MELLLA+ operating domain and the associated SLO setpoints are not affected by operation of the plant in the MELLLA+ region.

3.5.3.2 BSEP SAR Section 5.3.2, "Rod Block Monitor"

The licensee confirmed that the generic the M+ LTR treatment of the rod block monitor (RBM) topic is applicable to BSEP. [[ ]]

The NRC staff concludes that the generic M+ LTR resolution is applicable to BSEP because [[ ]]

3.6 <u>BSEP SAR Section 6.0, "Electrical Power and Auxiliary Systems"</u>

3.6.1 BSEP SAR Section 6.1, "AC Power"

The licensee confirmed that the generic M+ LTR treatment of the alternating current (AC) power topic is applicable to BSEP. Specifically, MELLLA+ operation does not change the BSEP reactor thermal power or the electrical output from the station. In addition, [[ ]]

The NRC staff concludes that the generic M+ LTR resolution is applicable to BSEP because the licensee finds that for the AC power system, [[

]]

3.6.2 BSEP SAR Section 6.2, "Direct Current (DC) Power"

The licensee confirmed that the generic M+ LTR treatment of the DC power topic is applicable to BSEP. Specifically, MELLLA+ operation does not change system requirements for control or motive power loads. As such, As such, [[
]]

The NRC staff concludes that the generic M+ LTR resolution is applicable to BSEP because the licensee finds that [[ ]] as a result of MELLLA+ operating domain expansion.

3.6.3 BSEP SAR Section 6.3, "Fuel Pool"

3.6.3.1 SAR Section 6.3.1, "Fuel Pool Cooling"

The licensee confirmed that the generic M+ LTR treatment of the spent fuel pool (SFP) cooling topic is applicable to BSEP. Specifically, reactor power does not increase as a result of MELLLA+ operation. [[

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The NRC staff concludes that the generic M+ LTR resolution is applicable to BSEP because fuel pool cooling [[ ]]

3.6.3.2 SAR Section 6.3.2, "Crud Activity and Corrosion Products"

The licensee confirmed that the generic M+ LTR treatment of the SFP crud activity and corrosion products topic is applicable to BSEP. Specifically, [[

]]

The NRC staff concludes that the generic M+ LTR resolution is applicable to BSEP because crud activity and corrosion products [[
]]

3.6.3.3 SAR Section 6.3.3, "Radiation Levels"

The licensee confirmed that the generic M+ LTR treatment of the SFP radiation levels topic is applicable to BSEP. Specifically, [[
]]

The NRC staff concludes that the generic M+ LTR resolution is applicable to BSEP because SFP radiation levels [[ ]]

3.6.3.4 SAR Section 6.3.4, "Fuel Racks"

The licensee confirmed that the generic M+ LTR treatment of the fuel racks topic is applicable to BSEP. Specifically, reactor power does not increase as a result of MELLLA+

operation. [[
]]

The NRC staff concludes that the generic M+ LTR resolution is applicable to BSEP because the fuel racks [[ ]]

3.6.4 BSEP SAR Section 6.4, "Water Systems"

The licensee confirmed that the generic M+ LTR treatment of the water systems topic is applicable to BSEP. Specifically, MELLLA+ operation does not affect the performance of the safety-related service water system or the RHR service water system during and following the most limiting design basis event (i.e., LOCA). In addition, [[
]]

The NRC staff concludes that the generic M+ LTR resolution is applicable to BSEP because the performance of water systems are unaffected by operation in the MELLLA+ operating domain.

3.6.5 BSEP SAR Section 6.5, "Standby Liquid Control System"

See discussion in Section 3.2.3 for SLC evaluation in MELLLA+.

3.6.6 BSEP SAR Section 6.6, "Heating, Ventilation and Air Conditioning"

The licensee provided an evaluation of the heating, ventilation and air conditioning (HVAC) systems that consists mainly of heating, cooling supply, exhaust and recirculation units in the turbine building, reactor building, containment building and the drywell, auxiliary building, fuel-handling building, control building, and the radwaste building. The licensee confirmed that the generic M+ LTR treatment of the HVAC topic is applicable to BSEP. Specifically, for BSEP HVAC systems, the process temperatures and heat loads from motors and cables in the CLTP MELLLA+ operating conditions are bounded by the CLTP process temperatures and heat loads. Thus, the licensee maintained that the HVAC systems in the MELLLA+ operating domain are within their current design for the worst case conditions.

The NRC staff concludes that the generic M+ LTR resolution is applicable to BSEP because operation in the MELLLA+ domain is bounded by current plant operation with respect to HVAC systems.

3.6.7 BSEP SAR Section 6.7, "Fire Protection"

The licensee confirmed that the generic M+ LTR treatment of the fire protection topic is applicable to BSEP. Specifically [[

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The NRC staff concludes that the generic M+ LTR resolution is applicable to BSEP because fire protection [[ ]]

3.6.8 BSEP SAR Section 6.8, "Other Systems Affected"

The licensee confirmed that the generic M+ LTR treatment of the other systems affected topic is applicable to BSEP. Specifically, the licensee performed a review to assure that the SAR included all systems that may be affected by the implementation on MELLLA+. The licensee has confirmed that those systems that are significantly affected by the operating in the MELLLA+ domain are addressed in the BSEP SAR.

The NRC staff concludes that the generic M+ LTR resolution is applicable to BSEP because other systems not addressed in the BSEP SAR are not affected by operation in the MELLLA+ operating domain with significant impact.

3.7 <u>BSEP SAR Section 7.0, "Power Conversion Systems"</u>

3.7.1 BSEP SAR Section 7.1, "Turbine-Generator"

The licensee confirmed that the generic M+ LTR treatment of the turbine-generator topic is applicable to BSEP. Specifically, there is no change in the BSEP reactor power level, reactor operating pressure, MS flow rates, or electrical output of the generator as a result of MELLLA+ operation. Thus, the licensee maintained that there is no change to the BSEP missile avoidance and protection analysis.

The NRC staff concludes that the generic M+ LTR resolution is applicable to BSEP because the turbine-generator is unaffected by operation in the MELLLA+ operating domain.

3.7.2 BSEP SAR Section 7.2, "Condenser and Steam Jet Air Ejectors"

The licensee confirmed that the generic M+ LTR treatment of the condenser and steam jet air ejectors topic is applicable to BSEP. Specifically, the licensee stated that there is no change in the BSEP reactor power level, reactor operating pressure, or MS flow rates as a result of MELLLA+ operation. The licensee stated **[[**

]]

The NRC staff concludes that the generic M+ LTR resolution is applicable to BSEP because the condenser and steam jet air ejectors are unaffected by operation in the MELLLA+ operating domain.

3.7.3 BSEP SAR Section 7.3, "Turbine Steam Bypass"

The licensee confirmed that the generic M+ LTR treatment of the turbine steam bypass topic is applicable to BSEP. Specifically, there is no change in the BSEP reactor power level, reactor operating pressure, or MS flow rates as a result of MELLLA+ operation.

The NRC staff concludes that the generic M+ LTR resolution is applicable to BSEP because the turbine steam bypass system is unaffected by operation in the MELLLA+ operating domain.

3.7.4 BSEP SAR Section 7.4, "Feedwater and Condensate Systems"

The licensee confirmed that the generic M+ LTR treatment of the FW and condensate topic is applicable to BSEP. Specifically, there is no change in the BSEP FW pressure, temperature, and flow rates. The moisture carryover (MCO) for MELLLA+ conditions increases from 0.10 to

0.20 percent by weight. The impact of higher moisture content is negligible on the reactor feed pump turbines and casing drains.

The NRC staff concludes that the generic M+ LTR resolution is applicable to BSEP because the FW and condensate systems are unaffected by operation in the MELLLA+ operating domain.

3.8 BSEP SAR Section 8.0, "Radwaste systems and radiation sources"

3.8.1 BSEP SAR Section 8.1, "Liquid and Solid Waste Management"

The largest source of liquid and wet solid waste is a result of the backwash of the condensate demineralizer. The overall volume of liquid radioactive waste and the coolant concentration of fission and corrosion products will be unchanged since the power level, FW flow, and steam flow do not change over the MELLLA+ operating domain expansion. Although the volume of waste generated is not expected to increase, there is potential that the MCO from the reactor steam could result in higher loading on the condensate demineralizers. The increase in MCO will be small and occur infrequently, which means that the condensate demineralizer and the reactor water clean-up (RWCU) filter demineralizer backwash frequency will not change significantly. Therefore, the NRC staff concludes that the waste volumes will not be significantly affected by the operation in the MELLLA+ operating domain.

3.8.2 BSEP SAR Section 8.2, "Gaseous Waste Management"

The operation of the offgas system helps to process and control the release of gaseous radioactive effluents to the environment. The system is operated and administratively controlled to ensure that the total radiation exposure of members of the public in the off-site environment is ALARA. The gaseous release rate is dependent on the fuel cladding performance, main condenser air inleakage, charcoal absorber inlet dew point, and charcoal absorber temperature. **[[ ]]** In addition, the radiolytic hydrogen gas flow rate does not change as a result of operations in the MELLLA+ operating domain. Thus, the performance of the recombiner will be unaffected. Therefore, the NRC staff concludes that the operation of the gaseous waste management system will not be affected by operations in the MELLLA+ operating domain.

3.8.3 BSEP SAR Section 8.3, "Radiation Sources in the Reactor Core"

The radiation sources in the reactor core are directly related to the power level during normal plant operations. These radiation sources include radiation from the fission process, accumulated fission products, and neutron activation of materials. BSEP operating in the MELLLA+ operating domain expansion does not involve a change in the current licensed maximum reactor thermal power. As a result, the NRC staff concludes that there is no impact on overall activity of the accumulated fission products or neutron activation of materials in the reactor core.

3.8.4 BSEP SAR Section 8.4, "Radiation Sources in Reactor Coolant"

In addition to the radioactive materials in the core, normal plant operations result in radioactive materials in the reactor coolant. These sources include small concentrations of fission products released from the fuel into the reactor coolant, activation of the reactor coolant itself producing nitrogren-16 (N-16), and activation of impurities in the coolant. Much like the radiation sources in the reactor core the production of the radiation sources in the reactor coolant is also directly related to the power level that did not change.

When conducting operations in the MELLLA+ operating domain the moisture content of the steam leaving the reactor vessel may increase up to 0.2 wt% at times when operating at the minimum core flow in the MELLLA+ operating domain. The moisture content that is carried in the steam is called MCO. With this increase in the MCO from the reactor vessel steam additional radioactivity will be carried over resulting in fission and activated corrosion product levels in the plant being affected when operating in the MELLLA+ operating domain. The concentrations of the fission and activated corrosion products in the reactor coolant and steam will remain bounded by the current design basis radionuclide concentrations. Therefore, the NRC staff concludes that the concentration of radiation sources in the reactor water is not expected to be significantly impacted by operating in the MELLLA+ operating domain.

3.8.5 BSEP SAR Section 8.5, "Radiation Levels"

Radiation levels in the plant during normal and post-shutdown operations are dependent on radiation levels and radionuclides present in the reactor coolant (water and steam). The post-shutdown radiation levels are dominated by the accumulated contamination of some fission and activated corrosion products. The BSEP reactor power and steam flow rate does not change as a result of the MELLLA+ operating domain expansion, so the radiation levels from the activation of the coolant will not vary significantly, unless the MCO from the reactor vessel increases. The MCO may increase at certain times while operating in the MELLLA+ operating domain. The licensee stated that BSEP will monitor the MCO to ensure it is controlled within the 0.2 wt% limit. The overall radiological effects of the increase in MCO is a function of the coolant radiochemistry and the levels of activated corrosion products. To address any increases in the radiological effects of the MCO, BSEP maintains appropriate health physics and ALARA controls in accordance with the regulations in 10 CFR Part 20.

The in-plant post-accident radiation levels depend primarily upon the post-accident source term. The post-accident source term consist of the core inventory of fission products and radionuclides in the coolant available for release during a postulated accident. The post-accident source term is also dependent on the maximum licensed power. Operating in the MELLLA+ operating domain does not change the maximum licensed power so there is no impact on the in-plant radiological hazard expected during an accident or on the licensee's assessment of vital area access per the Three Mile Island Lessons Learned Action Plan in NUREG-0737, Item II.B.2. Therefore, the NRC staff concludes that the increase in radiation sources associated with operations in the MELLLA+ operating domain will not adversely impact the licensee's ability to maintain occupational and public radiation doses within the applicable limits in 10 CFR Part 20 and ALARA.

3.8.6 BSEP SAR Section 8.6, "Normal Operation Off-Site Doses"

Airborne releases from the offgas system and gamma shine from the plant turbines are the primary sources of off-site doses to members of the public. As a result of operations in the MELLLA+ operating domain the reactor power and steam flow rate do not change. The MCO in the MS can increase in the MELLLA+ operating domain for short periods of time during the operating cycle. The gamma shine from the plant turbines during normal operations is dominated by the short-lived radionuclide N-16. Since the maximum power level is not increasing the amount of N-16 gamma shine will not increase. In addition, the potentially higher fission products in the steam due to the higher MCO are expected to have a negligible effect on the normal radiation levels. The increase in the MCO will also result in a negligible effect on plant gaseous emissions and gamma shine. Therefore, the NRC staff concludes that the contribution to off-site doses will be negligible and doses to the public will remain a small percentage of the dose limits in 10 CFR Part 20.

3.9 BSEP SAR Section 9.0, "Reactor Safety Performance Evaluations"

Section 9.0, "Reactor Safety Performance Evaluations" of the BSEP SAR evaluates the following topics, and their associated subjections, on a plant-specific basis:

9.1 "Anticipated Operational Occurrences"
9.2 "Design Basis Accidents and Events of Radiological Consequences"
9.3 (except 9.3.2 "Station Blackout") "Special Events"

3.9.1 BSEP SAR Section 9.1, "Anticipated Operational Occurrences"

The plant-specific updated final safety analysis report (UFSAR) for Brunswick contains the design basis analyses to evaluate the effects of a wide range of AOOs that might occur at the plant. The licensee has reviewed the UFSAR to identify the potentially limiting AOOs, in terms of thermal margin (i.e., ΔCPR), under the proposed MELLLA+ conditions at BSEP. The potentially limiting events include:

- Generator Load Rejection Without Bypass (LRNB)
- Turbine Trip Without Bypass (TTNB)
- Feedwater Controller Failure (Maximum Demand) (FWCF)
- Loss of Feedwater Heater (LFWH)
- Rod Withdrawal Error (RWE)

This list of limiting events is consistent with the M+ LTR and approving SE, which require that these events be analyzed on a cycle- and core-configuration-specific basis during the standard reload analyses. For each event, the M+ SE specifies that the analyses be performed at 100% CLTP and at both the minimum MELLLA+ flow condition and the ICF condition, to ensure that the extension to the MELLLA+ operating domain results in acceptable ΔCPR values that do not compromise the integrity of the fuel (i.e. do not violate the SAFDLs, as required by GDC 10, 15, 17, and 20).

Per the condition in the M+ SE, the licensee will submit the limiting AOO analyses as part of the RSAR for the initial BSEP MELLLA+ cycle to the NRC staff for confirmation, satisfying L&C 12.4. Non-limiting events in BSEP are treated via the generic resolution of these events in the M+ SE.

BSEP intends to implement the proposed MELLLA+ amendment during Cycle 22 in Unit 1 and prior to the end of Cycle 23 in Unit 2. These cycles will contain full cores of ATRIUM-10XM fuel, and an RSAR will be submitted for the first MELLLA+ cycle. In addition, the licensee has submitted, along with the SAR, analyses for the limiting AOOs for a representative Brunswick MELLLA+ cycle (based on Brunswick Unit 1 Cycle 19). The Unit 1 Cycle 19 core consisted of 234 fresh ATRIUM-10XM fuel assemblies and 326 irradiated ATRIUM-10 assemblies. The NRC staff issued SRXB-RAI-15 to clarify how ΔCPR values were reported for this reference cycle. In the RAI response, the licensee clarified that the ΔCPR values reported for the reference cycle were the limiting ΔCPR values taking into account only the ATRIUM-10XM fuel assemblies. The licensee stated that any limiting CPR values that may have occurred in ATRIUM-10 assemblies were ignored for the purposes of the reference cycle analyses. The NRC staff finds this acceptable because reload analyses, after MELLLA+ implementation, will be submitted by the licensee before each cycle based on the actual core configuration of that cycle (which will include only ATRIUM-10XM fuel under MELLLA+). Additionally the reference calculations are provided to demonstrate the licensee's ability to calculate the ΔCPR at MELLLA+ conditions.

Results of the licensee's analyses are shown in Table 3.9.1-1 (reproduced from Table 9-1 of the BSEP SAR). The limiting events are TTNB and LRNB, which both show a ΔCPR of 0.33 for the ICF/CLTP case and 0.30 for the 85% flow/CLTP case. For these two cases, as well as for FWCF, the ΔCPR response was more limiting at ICF than at 85% flow. Therefore, the licensee determined that operating in the MELLLA+ domain does not result in an increased OLMCPR.

Table 3.9.1-1: AOO Event Results Summary

Event	Parameter	Unit	CLTP ICF (104.5% Rated Core Flow)	CLTP 85% Rated Core Flow
TTNB	Peak Neutron Flux	% Initial	377	337
	Peak Heat Flux	% Initial	130	128
	Peak Vessel Pressure	psia	1302	1290
	ΔCPR (TSSS)	N/A	0.33	0.30
LRNB	Peak Neutron Flux	% Initial	377	333
	Peak Heat Flux	% Initial	130	128
	Peak Vessel Pressure	psia	1300	1289
	ΔCPR (TSSS)	N/A	0.33	0.30
FWCF	Peak Neutron Flux	% Initial	314	270
	Peak Heat Flux	% Initial	128	126
	Peak Vessel Pressure	psia	1274	1259
	ΔCPR (TSSS)	N/A	0.30	0.27
LFWH	ΔCPR	N/A	0.10	--
CRWE	ΔCPR	N/A	0.19	--

The licensee's analysis shows that there is acceptable margin to the fuel design limits for these AOOs. The licensee demonstrated that the methodology used to analyze these AOOs is acceptable in the MELLLA+ operating domain (see the NRC staff's evaluation in Appendix E of this SE for BSEP plant-specific use). Thus, the NRC staff finds that the licensee can adequately evaluate AOOs in the MELLLA+ operating domain.

3.9.2 BSEP SAR Section 9.2, "Design Basis Accidents and Events of Radiological Consequence"

The control rod drop accident (CRDA) is evaluated in this section.

Enclosure 15 of Reference 1 (ANP-3280P, *Brunswick Unit 1 Cycle 19 MELLLA+ Reload Analysis)* reports that a CRDA evaluation was performed for both A and B sequence startups consistent with the withdrawal sequence specified by Duke Energy using an AREVA CRDA legacy methodology (Reference 28). This CRDA analysis demonstrated that the maximum deposited fuel rod enthalpy is less than 280 cal/g and the estimated number of fuel rods that exceed the fuel damage threshold of 170 cal/g is less than the number of failed rods supported by the Brunswick CRDA alternate source term (AST) analysis based on the ATRIUM 10XM fuel. However, the NRC staff requested the licensee to reevaluate the CRDA analysis based on either the SRP Section 4.2 or draft regulatory guide (DG-1327) since the NRC staff determined that the licensee's initial CRDA analysis was found inadequate to ensure fuel rod geometry and long term core coolability. In response, the licensee submitted an assessment of the CRDA analysis with the revised criteria per DG-1327 (Reference 29). This assessment included the pellet-cladding mechanical interaction (PCMI) criteria that addresses fuel failures due to pellet clad mechanical interaction, high temperature failure threshold, and rod failure assessment. The results indicate that an approximate margin of **[[ ]]** is maintained to the PCMI failure thresholds. For the high temperature failure threshold assessment, the maximum total enthalpy to the rod drops is **[[ ]]** cal/g. The assessment also supports the conclusion that fuel melting will not occur for the rod drops occurring in the startup range. The NRC staff reviewed the submitted report and finds that the results from the new CRDA assessment are within the acceptance criteria established in the Draft Regulatory Guide (RG) DG-1327 and are thus acceptable.

Moreover, the CRDA, MS line break accident (outside containment), LOCA (inside containment), and fuel-handling accident were resolved generically in the M+ LTR and approved in the corresponding NRC SE. The licensee confirmed in the BSEP SAR and the NRC staff agrees that the CRDA, MS line break accident (outside containment), LOCA (inside containment), and fuel-handling accident generic evaluations were consistent with the generic resolution in the M+ LTR and conservative with respect to the BSEP alternative source term. The M+ LTR also resolutions the instrument line break accident, large line break (FW or RWCU), offgas system failure, and cask drop accident on a generic basis. The applicant determined that these accidents were not applicable to BSEP. The NRC staff determined that these accidents were not BSEP DBAs in the BSEP UFSAR or evaluated in the BSEP alternate source term evaluation.

The M+ LTR further specifies that a plant-specific evaluation be performed for the liquid radwaste tank failure because it was not resolutioned generically. The liquid radwaste tank failure is not listed in the BSEP UFSAR as a DBA. However, catastrophic failures of the radwaste system tanks due to seismic forces are addressed in Chapter 11 of the BSEP UFSAR. The current licensing basis specifies that:

> The radwaste building has the capacity to contain simultaneous rupture of all radwaste tanks. Process containers associated with the radwaste processing area are contained within a separate facility adjacent to the radwaste building loading dock. This facility is designed to contain leakage similar to the radwaste building. Therefore, because leaks or spills from the liquid radwaste system are retained on site, a small or major leak has no effect on the dose rates at the plant boundary. Furthermore, the system is monitored for inadvertent discharge of high level waste (see Section 11.5). Thus the liquid radwaste system fulfills the safety design basis by limiting the discharge of radioactive liquids through the

site boundary well within the applicable requirements of 10 CFR 20 and Appendix I of 10 CFR 50.

Furthermore, BSEP indicated "Catastrophic failure of the radwaste system by seismic forces has been safeguarded by designing tanks containing high liquid activity concentrations and the radwaste building to seismic Class I standards. All catastrophic spills, if they occur, will be contained within the radwaste building or the radwaste processing area."

The NRC staff confirmed that the LAR contained no changes to the relevant BSEP licensing basis (licensed core power, decay heat, pressure, or steam flow as a result of the MELLLA+ operating range expansion). Furthermore, the magnitude of the potential radiological consequences depends on the quantity of fission products released to the environment, the atmospheric dispersion factors, and the dose exposure pathways. The currently licensed quantity of fission products, atmospheric dispersion factors, and the dose exposure pathways do not change as a result of operating in the MELLLA+ operating domain.

The NRC staff reviewed the dose consequences of the licensee's proposed changes. Since there are no major modifications to plant equipment, no increases in the design basis operating pressure, power, core inventory source terms, steam flow rate, and FW flow rate, the NRC staff finds that BSEP's DBA dose consequence evaluation is reasonable. Furthermore, all dose consequences relating to the proposed expansion of the power/flow map to MELLLA+ is bounded by the currently licensed DBAs.

Since BSEP MELLLA+ operation is (1) bounded by the existing analyses in the NRC approved M+ LTR and the plant-specific design basis information documented in the BSEP UFSAR, and (2) the radiological dose consequences for all accidents remain below the design criteria specified in 10 CFR 50.67, "Accident Source Term," and (3) the accident specific design criteria outlined in RG 1.183, the NRC staff concludes that the implementation of MELLLA+ at BSEP is acceptable.

3.9.3 BSEP SAR Section 9.3, "Special Events"

3.9.3.1 BSEP SAR Section 9.3.1, "Anticipated Transient Without Scram"

Regulatory Basis

The NRC staff's evaluation of BSEP ATWS is based on 10 CFR 50.62, "Requirements for reduction of risk from ATWS events for light-water-cooled nuclear power plants," which defines an ATWS as an AOO followed by the failure of the reactor trip portion of the protection system specified in GDC 20.

GDC 35 requires that fuel and clad damage that could interfere with continued core cooling must be prevented (the "core coolability" requirement) and that clad metal-water reaction be limited to negligible amounts. Section 46 of 10 CFR Part 50 defines three specific core coolability criteria: (1) Peak clad temperature shall not exceed 2200°F, (2) Maximum cladding oxidation shall not exceed 17% the total cladding thickness before oxidation, and (3) Maximum hydrogen generation shall not exceed 1% of the maximum hypothetical amount if all the fuel clad had reacted to produce hydrogen. Satisfying the 10 CFR 50.46 criteria is a way to demonstrate the core coolability requirement of GDC 35.

The following 10 CFR 50.62 requirements are relevant to BSEP:

- Each BWR must have an alternate rod insertion (ARI) system that is diverse (from the reactor trip system) from sensor output to the final actuation device. The ARI system must be designed to perform its function in a reliable manner and be independent (from the existing reactor trip system) from sensor output to the final actuation device.
- Each BWR must have a SLC system capable of injecting borated water solution into the reactor pressure vessel (RPV) at such a flow rate, Boron concentration, and Boron-10 enrichment, and accounting for RPV volume, that the resulting reactivity control is at least equivalent to that resulting from injecting 86 gallons per minute of 13 weight percent sodium pentaborate decahydrate solution at the natural Boron-10 abundance into a 251-inch inside diameter RPV for a given core design. The SLC system and its injection location must be designed to perform its function in a reliable manner.
- Each BWR must have equipment to trip the reactor coolant recirculation pumps automatically under conditions indicative of an ATWS. This equipment must be designed to perform its function in a reliable manner.

The NRC staff's review was conducted to determine if that:

- the above requirements are met,
- sufficient margin is available in the setpoint for the SLC system pump discharge relief valve such that SLCS operability is not affected by the proposed operating domain expansion, and
- operator actions specified in the plant's emergency operating procedures (EOPs) are consistent with the generic emergency procedure guidelines (EPGs) and severe accident guidelines (SAGs), insofar as they apply to the plant design.

In addition, the NRC staff reviewed the licensee's ATWS analysis to determine if the following ATWS acceptance criteria were met:

- the peak vessel bottom pressure is less than the ASME Service Level C limit of 1500 psig;
- GDC 35 is satisfied;
- the peak suppression pool temperature is less than the design limit; and
- the peak containment pressure is less than the containment design pressure.

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

Applicable Limitations and Conditions

There are 17 L&Cs, or parts of L&Cs, in the M+ SE pertaining to the ATWS analysis. The licensee addressed these limitations in Appendix B of the BSEP SAR.

1. M+ SE L&C 12.17, which specifies that at least two plant-specific ATWS calculations (MSIVC and PRFO) be performed, with an additional loss of offsite power (LOOP) calculation required if the RHR capability is affected by LOOP.

2. M+ SE L&C 12.18.a, which specifies that plant-specific TRACG calculations in the event that the HCTL is exceeded in the ODYN ATWS calculations.

3. M+ SE L&C 12.18.b, which specifies that TRACG ATWS calculations are not required if the plant increases the Boron-10 concentration/enrichment so that the peak suppression pool temperature does not change with respect to a reference OLTP/75% flow ODYN calculation.

4. M+ SE L&C 12.18.c, which specifies that PCT for both the initial overpressure and emergency depressurization phases of the transient must be evaluated on a plant-specific basis with TRACG.

5. M+ SE L&C 12.18.d, which specifies that operation in the MELLLA+ domain is to be consistent with the plant-specific ATWS analyses, including equipment out of service conditions (as specified in the Supplemental Reload Licensing Report (SRLR)). Additionally, the condition requires that the plant-specific ATWS analyses are consistent with the input parameters and engineering safety features as defined in the TSs and with the allowed plant configuration.

6. M+ SE L&C 12.18.e, which specifies that nominal input parameter values and treatment of their uncertainties may be used consistent with the original GE ATWS analyses in NEDE-24222, or may differ from the original analyses in a manner yielding more conservative results.

7. M+ SE L&C 12.18.f, which specifies that the licensee tabulate and discuss the key input parameters and uncertainty treatment.

8. M+ SE L&C 12.23.1, which is included as part of L&C 12.18.d.

9. M+ SE L&C 12.23.2, which requires that all plant-specific ODYN and TRACG key calculation parameters be provided for staff verification.

10. M+ SE L&C 12.23.3, which defines requirements for SRV upper pressure tolerances used in the ATWS analyses based on plant-specific performance and consideration of uncertainty and valve drift.

11. M+ SE L&C 12.23.4, which specifies review of the EPG/SAG parameters for applicability to the MELLLA+ domain and requires confirmation that the ATWS analyses are consistent with the EOP operator actions.

12. M+ SE L&C 12.23.5, which specifies that a power/flow ratio of less than 52.5 MWt/Mlbm/hr at minimum allowable core flow rate at 120 percent OLTP.

13. M+ SE L&C 12.23.8, which specifies that the plant-specific ATWS analyses are to account for all plant- and fuel-design-specific features, such as the debris filters.

14. M+ SE L&C 12.23.9, which specifies that a review of the safety system specifications be done to ensure that all assumptions used for the ATWS analyses apply to the plant-specific conditions, particularly for crucial safety systems such as high pressure coolant injection (HPCI) and physical limitations such as net positive suction head (NPSH). It also requires discussion and evaluation of NPSH and system performance throughout the ATWS event.

15. M+ SE L&C 12.23.10, which states that plant-specific applications must ensure that an ATWS-related containment pressure increase under MELLLA+ operation does not adversely affect the operation of safety-grade equipment.

16. M+ SE L&C 12.23.11, which specifies that 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 heat capacity temperature limit (HCTL) limit for emergency depressurization.

17. M+ SE L&C 12.24.4, which is included as part of L&C 12.18.d.

Technical Evaluation

An ATWS is an AOO, as defined in Appendix A to 10 CFR Part 50, followed by the failure of the reactor trip portion of the protection system specified in GDC 20. Since protection systems (e.g., the reactor trip system) must satisfy the single-failure criterion, multiple failures or a common mode failure must cause the assumed failure of the reactor trip. The probability of an AOO, in coincidence with multiple failures or a common mode failure, is much lower than the probability of any of the other events that are evaluated under SRP Chapter 15 (Reference 4). Therefore, an ATWS event cannot be classified as either an AOO or a design-basis accident.

The failure of the reactor to shut down during certain transients can lead to unacceptable reactor coolant system pressures, fuel conditions, and/or containment conditions. For a BWR, AOOs with failure to scram that could lead to unacceptable conditions include closure of main steam line isolation valves, or turbine trip with bypass available if unmitigated unstable power oscillations are allowed to grow.

Safety issues associated with an ATWS have been evaluated since the early 1970s. During NRC evaluations of vendor models and analyses addressing ATWS events, the NRC formally identified the ATWS as Unresolved Safety Issue (USI) A-9, "Anticipated Transients Without Scram." The NRC presents the staff's studies and findings regarding USI A-9 in NUREG-0460. In 1986, the NRC resolved USI A-9 through publication of 10 CFR 50.62, the ATWS rule (the rule). Although the rule does not require ATWS analyses, SECY 83-293 and the *Federal Register* notice of the final rule in 49 FR 26036 present the bases for current regulatory requirements related to ATWS events, including the associated regulatory evaluation.

The rule requires that certain light-water-cooled plants have prescribed systems and equipment that have been determined to reduce the risks attributable to ATWS events, for each of the nuclear steam supply system (NSSS) vendor's designs, to an acceptably low level. The rule also requires applicants to demonstrate the adequacy of their plants' prescribed systems and equipment.

Effect of MELLLA+ Operation on ATWS

A large number of ATWS events are possible. However, only a small number of events are expected to be limiting in terms of the ATWS acceptance criteria described in the Regulatory Evaluation section above. L&C 12.17 of the M+ SE specifies that the following ATWS events must be considered as the limiting events for operation up to and including the M+ domain:

- Main steam isolation valve closure (MSIVC)
- Pressure regulator failure open (PRFO)

In the event of a failure-open of the pressure regulator, the turbine control and turbine bypass valves open. This increases the steam flow rate until the low-pressure setpoint is reached, resulting in MSIV closure. In both the MSIVC and PRFO events, the closure of the MSIV results in a pressurization wave that decreases the core void fraction that leads to an increase in core power due to coolant density reactivity feedback. In addition, during these events, a recirculation pump trip is performed. This reduces the power level, which reduces the heat load to the suppression pool by reducing the rate of steam generation in the core (thereby reducing the mass flow rate of steam being vented to the suppression pool). Containment integrity is assured as long as the suppression pool temperature and containment pressure remain within the ATWS acceptance limits. If depressurization is required to avoid exceeding the containment pressure limit, the radiological consequences of the released gases to the environment must be determined to be within acceptable limits. Additionally, analyses must be performed for both events to ensure that the maximum vessel pressure does not exceed the ASME Service Level C limit of 1,500 psig.

Additionally, M+ SE L&C 12.17 states that if the RHR heat exchanger effectiveness is affected by LOOP, then the LOOP event must be analyzed as well.

There is no change in core power, decay heat, pressure, or steam flow as a result of the MELLLA+ operating domain expansion. However, operation at the highest-power, minimum-flow MELLLA+ operating condition results in a less effective power reduction following a recirculation pump trip, compared to operation in the MELLLA operating domain. This leads to increased suppression pool temperatures when starting from MELLLA+ conditions relative to MELLLA conditions. The less-effective power reduction in M+ after RPT also results in higher short-term peak vessel overpressure, possibly challenging the 1500 psig peak vessel bottom pressure ATWS acceptance limit. Therefore, extension of the operating domain to M+ requires an analysis of the limiting ATWS events to ensure that the ATWS acceptance criteria continue to be met with the increased post-RPT power level.

ATWS Calculations for MELLLA+

The AREVA transient code COTRANSA2 (Reference 19) and the GEH transient code ODYN (Reference 20) have been approved as licensing basis codes for ATWS analyses in the M+ domain, per the M+ SE and justification of COTRANSA2 in the MELLLA+ domain is discussed in Appendix E herein. COTRANSA2 and ODYN are suitable for modeling the transient behavior of the system during limiting ATWS events, to determine whether the ATWS acceptance criteria are satisfied. For BSEP MELLLA+, COTRANSA2 is used exclusively for the short-term ATWS vessel overpressure analyses, which must be confirmed on a cycle-specific basis. For BSEP MELLLA+, the long-term ATWS analyses for suppression pool temperature, containment pressure, and cladding PCT are performed using ODYN.

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Additionally, confirmatory calculations performed by the NRC staff in the M+ SE indicate that ODYN is conservative in terms of suppression pool temperature at the end of the transient, but not conservative throughout the ATWS scenario timeline up to that point.

Due to the limitations of the ODYN code, the M+ SE requires that best-estimate TRACG calculations be additionally provided, if the licensing basis ODYN calculations predict that the HCTL is reached before hot shutdown is achieved. However, the ODYN calculations are not required if the plant increases the Boron-10 concentration/enrichment such that the integrated heat load to the containment calculated by ODYN does not change relative to a reference OLTP/75 percent flow ODYN calculation, in which case the reference OLTP/75 percent flow condition is found to be more limiting. In addition to suppression pool temperature and containment pressure, the best-estimate TRACG calculations must also evaluate PCT on a plant-specific basis for both the initial overpressure and emergency depressurization phases of the transient. Until TRACG is approved as the licensing code for ATWS analyses, a condition of the M+ SE specifies that the licensing ODYN calculations are to be supplemented by these higher-fidelity best-estimate TRACG calculations that include all operator actions and water level strategies specific to each plant.

Application to BSEP MELLLA+

The licensee performed plant-specific analyses for the MSIVC and PRFO ATWS events, as required by L&C 12.17 in the M+ SE. If the RHR heat exchanger effectiveness is affected by LOOP, then the LOOP event may be limiting in terms of containment response, and the M+ SE (L&C 12.17) requires that an analysis for the LOOP event be provided in this case. However, in the BSEP SAR, the licensee affirms that the RHR heat exchanger effectiveness in BSEP is not affected by LOOP, and therefore an evaluation of the LOOP event is not required for BSEP.

The long-term ATWS ODYN calculations for MSIVC and PRFO were initiated from 100% CLTP and 85% reactor core flow (RCF), which is the lowest allowed operating flow rate in the MELLLA+ domain corresponding to the highest allowed power level. This statepoint provides the most limiting initial condition in terms of suppression pool temperature and containment pressure because the low-flow operating point corresponds to the highest power level after the RPT. For both events, calculations were performed at beginning of cycle (BOC) and end of cycle (EOC) conditions, consistent with the approved M+ LTR. BOC is expected to be the most limiting exposure for peak vessel pressure, and EOC is expected to be the most limiting exposure for suppression pool temperature. This selection of initial operating point and exposure conditions is consistent with the approach used in the approved M+ LTR, and the NRC staff finds these conditions acceptable because they satisfy the requirements for ATWS analyses by providing reasonably limiting assumptions. There are no changes to the assumed operator actions (which includes water level reduction, SLC system Boron injection, and RHR suppression pool cooling) as defined in the BSEP EOPs, for the MELLLA+ ATWS analysis, relative to the MELLLA analysis.

Table 3.9.3.1-1 shows key results for the ODYN ATWS analysis. The calculated peak vessel pressure, peak suppression pool temperature, peak containment pressure, and peak cladding temperature remain within the design limits for all cases analyzed. The vessel pressure limit of 1,500 psig is the ASME Service Level C limit. The BSEP design limit for peak suppression pool temperature is 207.7 F. This value is above the HCTL value, which is determined from the BSEP EOPs as a function of reactor pressure and suppression pool level and is conservatively assumed to be 158 F (based on a pressure near the SRV opening pressure) for the ATWS analyses. M+ SE L&C 12.23.11 states that a licensee is to justify the use of a suppression pool temperature limit higher than the HCTL for emergency depressurization. The licensee used the containment design limit which is higher than HCTL for emergency depressurization. The NRC staff notes that the suppression pool temperature limit is unchanged relative to pre-M+ operation; additionally, the peak suppression pool temperature from the ATWS analyses for M+ (crediting the increased Boron enrichment included with the MELLLA+ extension) is lower than for pre-M+ conditions, so the original justification remains applicable and the NRC staff finds this resolution acceptable.

Results from using nominal fuel parameter values as well as the bounding fuel parameter sensitivities are described in Appendix D of this SE. **[[**

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These results are shown in Table 3.9.3.1-1. Although not all results were presented for the nominal fuel case, the licensee has selected the limiting fuel parameter case as the licensing basis, to conservatively bound the performance of ATRIUM-10XM fuel for BSEP MELLLA+.
[[
]] the ODYN analyses demonstrate that the ATWS acceptance criteria are satisfied for BSEP MELLLA+.

ODYN calculated a peak suppression pool temperature of 174.0°F, which is acceptable because it is below the BSEP suppression pool temperature design limit. The calculated value of 174.0°F exceeded the HCTL (conservatively assumed to be 158°F), requiring emergency depressurization that ODYN is not capable of modeling. However, the licensee elected to increase the Boron-10 enrichment to support MELLLA+ operation. This increased Boron shutdown capability led to a calculated peak suppression pool temperature of 174.0°F for CLTP and minimum flow conditions under MELLLA+. This was a reduction of 15.4°F relative to the pre-MELLLA+ calculated peak suppression pool temperature at OLTP and 75% flow. Because the pre-MELLLA+ conditions and original Boron-10 concentration and enrichment led to higher calculated heat load to the containment, no TRACG calculation was required for BSEP for MELLLA+ conditions, in accordance with Part b of M+ SE L&C 12.18.

Table 3.9.3.1-1. Key Results for Licensing Basis ODYN ATWS Analysis

ATWS Acceptance Criterion	MELLLA+ (nominal fuel parameters)	MELLLA+ (limiting fuel sensitivity parameters)	Design Limit
Peak Vessel Pressure (psig)	[[ ]]	1496	1500
Peak Suppression Pool Temperature (°F)	[[ ]]	174.0	207.7
Peak Containment Pressure (psig)	[[ ]]	8.4	62
Peak Cladding Temperature (°F)	Not given	1215	2200
Peak Local Cladding Oxidation (%)	<17	<17	17

The NRC staff reviewed the COTRANSA2 ATWS overpressure analyses provided for the representative MELLLA+ reload cycle analyses for Unit 1 Cycle 19. Analyses were presented for both 100%P/85%F and 100%P/104.5%F conditions. The maximum vessel pressure was calculated to be [[ ]] which occurred for the 100%P/85%F case for PRFO. These ATWS overpressure analyses will be repeated using the specific core configuration for each MELLLA+ cycle to confirm that the ATWS vessel pressure criterion continues to be met. Therefore, the NRC staff finds the licensee's analyses acceptable.

3.9.3.2 BSEP SAR Section 9.3.2, "Station Blackout"

Section 9.3.2 "Station Blackout" is addressed generically following the approach in the M+ LTR. The NRC staff reviewed the licensee's justification for use of the generic treatment to ensure BSEP MELLLA+ plant conditions and associated analysis falls within the bounds of generic treatment. The NRC staff finds that, since there is no expected change in core power, operating pressure, decay heat, and steam flow for BSEP MELLLA+, the generic resolution is acceptable and the Section 9.3.2 M+ SE evaluation is acceptable for this BSEP application.

3.9.3.3 BSEP SAR Section 9.3.3, "ATWS with Core Instability"

Regulatory Basis

The NRC staff's evaluation of BSEP ATWS is based on 10 CFR 50.62, "Requirements for reduction of risk from anticipated transients without scram (ATWS) events for light-water-cooled nuclear power plants," which defines an ATWS as an AOO followed by the failure of the reactor trip portion of the protection system specified in GDC 20.

GDC 35 requires that fuel and clad damage that could interfere with continued core cooling must be prevented (the "core coolability" requirement) and that clad metal-water reaction be limited to negligible amounts. 10 CFR 50.46 defines three specific core coolability criteria: (1) Peak clad temperature shall not to exceed 2200°F, (2) Maximum cladding oxidation shall not to exceed 17% the total cladding thickness before oxidation, and (3) Maximum hydrogen generation shall not to exceed 1% of the maximum hypothetical amount if all the fuel clad had reacted to produce hydrogen. Satisfying the 10 CFR 50.46 criteria is a way to demonstrate that the core coolability requirement of GDC 35.

However, the 10 CFR 50.46 criteria were developed for LOCA events in which the limiting factor is the availability of coolant; in these events, the absence of coolant is expected to result in gross core heating. By contrast, the issue during ATWS-I events is not one of inadequate cooling but of high local energy deposition and cladding heat flux. In the event that cladding temperatures exceed 2200°F in only a limited length of the hottest fuel rods in a few assemblies in the core, no significant core distortion, loss of core coolability, or impaired ability to safely shut down the core is expected to occur. Therefore, the NRC staff considers the ATWS acceptance criterion to be satisfied under these circumstances. This position has been previously stated by the NRC staff in its evaluation of NEDO-32047A (Reference 21).

Applicable Limitations and Conditions

The M+ LTR (Reference 5) and associated SE provide the following L&Cs relevant to BSEP analyses for ATWS-I:

- M+ SE L&C 12.3.d, which specifies that new plant-specific analyses be performed to demonstrate ATWS-I performance for cores with non-GE fuel;
- M+ SE L&C 12.19, which specifies that plant-specific ATWS-I analyses that satisfy the ATWS acceptance criteria listed in SRP Section 15.8, and gives requirements for ATWS-I calculations that must be based on approved NRC neutronic/TH codes;
- M+ SE L&C 12.23.6, which specifies that that bounding ATWS-I analyses be provided for M+ applications involving non-GE fuel.
- M+ SE L&C 12.23.7, which also specifies plant specific ATWS-I analysis to performed for fuel other than GE14. It is included in L&C 12.23.6.
- M+ SE L&C 12.24.1, which specifies that plant-specific applications that use TRACG are to use the actual flow configuration, including in-channel water rod flow.

Technical Evaluation

Under certain core conditions, BWRs may become susceptible to growing oscillations in power and flow rate, due to the time-dependent feedback between channel inlet flow rates, channel pressure drop, and local neutronic power levels. These coupled density wave oscillations become increasingly unstable with decreasing core flow rate and increasing core power, which requires that this region of the power-flow operating map be avoided during normal operation and AOOs. A long-term stability solution (LTS) is required in order to detect growing oscillations and suppress oscillations via reactor scram before the safety limits defined in GDC 10 and 12 are violated.

In the event of a reactor scram failure, the LTS is unable to suppress oscillations, and ATWS mitigation actions are required to suppress the oscillations in a timely manner to prevent loss of core coolability. As discussed in the ATWS section, operation in the M+ domain leads to a higher power level following a recirculation pump trip, relative to operation in pre-M+ domains. This increases the degree of instability of the core and causes oscillations to grow faster, increasing the likelihood of violating the coolability criterion before oscillation suppression can occur via the ATWS mitigation actions. The most limiting event for ATWS-I is the **[[**

]] This leads to the highest

power level at the lowest flow rate (i.e., natural circulation flow) among all ATWS events under consideration, and, therefore, this is the most limiting ATWS event in terms of stability.

Once the oscillations grow sufficiently large, local dryout may occur on one or more fuel rods during the low-flow phase of the oscillations, causing a dramatic reduction in the local cladding-to-coolant heat transfer coefficient and a corresponding increase in local cladding temperature. As the flow rate increases during the same oscillation period, rewetting of the cladding surface may occur depending on the flow conditions and cladding temperature near the dryout location. However, a 'ratcheting' effect will often occur, in which the cladding temperature does not have sufficient time to return to the cladding temperature from the previous oscillation period. This, in addition to increasing oscillation amplitude, may lead to conditions where the cladding does not rewet during the entire oscillation period. This failure to rewet leads to a much larger increase in cladding temperature which, if not mitigated quickly enough, may lead to an uncoolable geometry.

ATWS-I Calculations for BSEP MELLLA+

Calculations for ATWS-I in the BSEP SAR were performed with TRACG04. TRACG04 is not approved for long-term ATWS calculations, including ATWS with depressurization and ATWS with core instability. ODYN is the approved licensing basis code for ATWS, consistent with the NRC SE for NEDC-33006P (Reference 5). However, TRACG04 is used as a best-estimate code for the ATWS analysis, which is consistent with the NRC SE for NEDC-33006P. In the SE for NEDO-32047 and NEDO-32164, the NRC staff concluded that TRACG04 is an adequate tool to estimate the behavior of operating reactors during transients that may result in large power oscillations. Therefore, the NRC staff concludes that the licensee use of TRACG04 for best-estimate ATWS-I calculations is acceptable, as this code provides the most suitable capabilities and modeling features to adequately model the complex TH and coupled TH/neutronic phenomena associated with BWR instability.

TRACG04 defines a cladding temperature, known as the minimum stable film boiling temperature (T_{min}), above which the cladding surface heat transfer is forced to remain in film boiling even though the TH conditions indicate that transition to nucleate boiling is possible. The Modified Shumway correlation was used to determine T_{min} for ATWS-I calculations in the BSEP SAR. Additionally, the modified form of the correlation was used, which ignores void fraction dependence and assumes zirconium thermophysical properties for the cladding (which is the default given by Shumway).

This correlation was based primarily on experimental data involving reflooding and/or quench fronts characteristic of LOCA conditions. However, recent NRC experiments performed at the KATHY facility involving full-length BWR assemblies under realistic ATWS-I oscillatory conditions has indicated that the modified Shumway correlation may not adequately capture Tmin under ATWS-I conditions (Reference 16). As described in the Reference 16, the NRC staff determined that a T_{min} based on the homogeneous nucleation temperature gives acceptable agreement with the experimental data and is a reasonable model for ATWS-I analyses.

The licensee presented TRACG04 results for a simulated TTWBP event in BSEP initiated from minimum allowable core flow at EPU power levels, which is the most limiting point because it results in the highest post-RPT power level and therefore the largest oscillation growth rate. This is consistent with the calculations performed in the approved M+ LTR (NEDO-330060A, Revision 3, June 2009). The licensee performed calculations using an equilibrium MELLLA+

cycle at several exposure conditions (BOC, peak reactivity, and EOC) and found BOC to be the limiting exposure for the BSEP cycle that was analyzed. Two different TRACG channel grouping (i.e., nodalization) schemes were used in the analyses: one for regional mode oscillations and one for core-wide mode oscillations. M+ SE L&C 12.19 only specifies that regional mode analyses be performed, as this mode is expected to be limiting. To confirm that appropriate modeling was used, the NRC staff requested additional information regarding channel grouping for regional, as well as core-wide oscillations and local assembly behavior for regional mode oscillations in SRXB-RAI-5.

In its RAI response to SRXB-RAI-5, the licensee provided the requested information on TRACG TTWBP channel grouping for the core wide and regional oscillation modes and confirmed that the regional mode analyses were more limiting than the core wide mode analyses in terms of maximum PCT. Therefore, the licensee justified that the regional mode (and not core wide mode) TTWBP results in the SAR were calculated appropriately. The licensee also provided the requested results for the limiting channel TH and neutronic behavior, along with corresponding results for the radially symmetric channel in the core, during the TTWBP event. The information made the behavior of the core during regional oscillations more apparent compared to the full-specified channels that qualitatively exhibit the expected flow, power, and cladding temperature behavior.

Nominal or reasonably conservative values were used for most parameters for the ATWS-I calculations. The licensee stated this is consistent with the historical approach for ATWS (Reference 21 and 22), in which best-estimate calculations are performed. For this calculation, a 120-sec delay was assumed after failure to scram before manual water level reduction occurred, consistent with the BSEP EOPs. A nominal time-dependent FW temperature boundary condition was originally applied, based on a 1.3°F/sec FW temperature reduction rate, as presented in the BSEP SAR. The licensee refined this FW temperature boundary condition to 0.5°F/sec as a result of the sensitivities requested in the RAI responses. The licensee concluded that this reduction rate continues to conservatively bound plant data under similar turbine trip conditions.

The original ATWS-I results are shown in Figures 9-10 and 9-11 in the BSEP SAR (Enclosure 5, Reference 1). The NRC staff requested additional information regarding the approach used to ensure that the maximum steady-state linear heat generation rate was less than **[[ ]]** of the maximum linear heat generation rate limit and as clarification on the maximum allowed timestep in SRXB-RAI-4. The licensee subsequently amended the response to address an issue with the maximum allowable timestep size setting in TRACG, as well as a change in the initial control rod pattern for the TRACG analyses to be consistent with AREVA's equilibrium cycle specifications. The NRC staff reviewed these changes and concludes that they are acceptable for use in the TRACG ATWS-I analyses for the reasons provided in Appendix F of this SE. For the purposes of this section, "nominal ATWS-I results" refers to the updated results presented in SRXB-RAI-4 rather than the original results in the BSEP SAR.

The nominal ATWS-I results predict the occurrence of large-amplitude oscillations for roughly 60 sec before the oscillations are suppressed by downcomer water level reduction. During these oscillations, the limiting channel was predicted to repeatedly undergo dryout but was predicted to successfully rewet after each oscillation peak. This kept the peak clad temperature to a maximum of **[[ ]]** during the transient.

The licensee provided an additional set of TTWBP results that accounts for the uncertainties of using TRACG04 with ATRIUM-10XM fuel (see discussion in Appendix D herein). A fuel parameter sensitivity study was performed by varying relevant modeling parameters within appropriate ranges of uncertainty. The process of selecting parameters, determining sensitivity ranges, and determining limiting values for the TTWBP analyses is documented in the response to RAI-SRXB-6, which the NRC staff evaluates in Appendix F of this SE. Based on this evaluation, the NRC staff finds that the fuel parameter sensitivities acceptably account for ATRIUM-10XM performance during the ATWS-I analyses. The limiting fuel parameter sensitivities results in a maximum PCT of **[[ ]]** during the TTWBP event, which is a **[[ ]]** increase over the nominal ATWS-I results. The sensitivity results are discussed further in the next section.

ATWS-I T_{min} Sensitivity Analyses

Previous M+ reviews (e.g., Peach Bottom MELLLA+, Reference 23) have shown that, should T_{min} be exceeded and should failure-to-rewet occur, the PCT would be expected to increase at least several hundred degrees, and in some cases may exceed a PCT of 2200°F in one or more fuel rods. Furthermore, based on recent NRC test experiments at the KATHY facility, the NRC staff has determined that a more realistic estimation of T_{min} under ATWS-I oscillatory conditions is the homogenous nucleation temperature (Reference 16). Therefore, additional sensitivity calculations were requested by the NRC staff to examine the calculated behavior using the homogeneous nucleation temperature plus contact temperature (HN+CT) model for T_{min} as proposed by Bjornard and Griffith (Reference 24), which provides a lower estimation of T_{min} and, therefore, allows failure-to-rewet to occur at lower cladding temperatures.

The licensee provided the requested sensitivity results in the response to SRXB-RAI-7. To ensure that the 2200°F PCT criterion was not exceeded when using nominal fuel parameter values, the licensee included sensitivity results that assumed a "best-estimate" FW temperature reduction rate of 0.5°F/sec as opposed to 1.3°F/sec assumed in the nominal ATWS-I case. The NRC staff requested additional justification for this reduction rate in SRXB-RAI-8.

In its RAI response, the licensee presented plant data for four turbine trip events at BSEP in which the majority, if not all, of the coolant inventory following the trip was supplied by the FW system. The measured FW temperature in each event exceeded the FW temperature that would occur assuming a constant 0.5 F/sec rate of decrease initiated at the time of the trip or first significant turbine load decrease.

In these events, after plant scram, the FW demand after the trip was low. By contrast, during a high power ATWS, the FW demand remains relatively high as the operators follow the EOP reactor water level strategy. This means that the inventory of heated water in the FW piping after the trip will enter the vessel sooner, leading to a more rapid decrease in FW temperature for the ATWS. To account for this, in Figure SRXB-8-5 of the RAI response (and included as Figure 3.9.3.3-1 below), the licensee presented the same measured FW temperature data versus the integrated FW mass that has entered the RPV since the turbine trip. In each case, the FW temperature versus integrated FW mass is conservatively bounded by the case of a 0.5 F/sec FW temperature reduction rate.

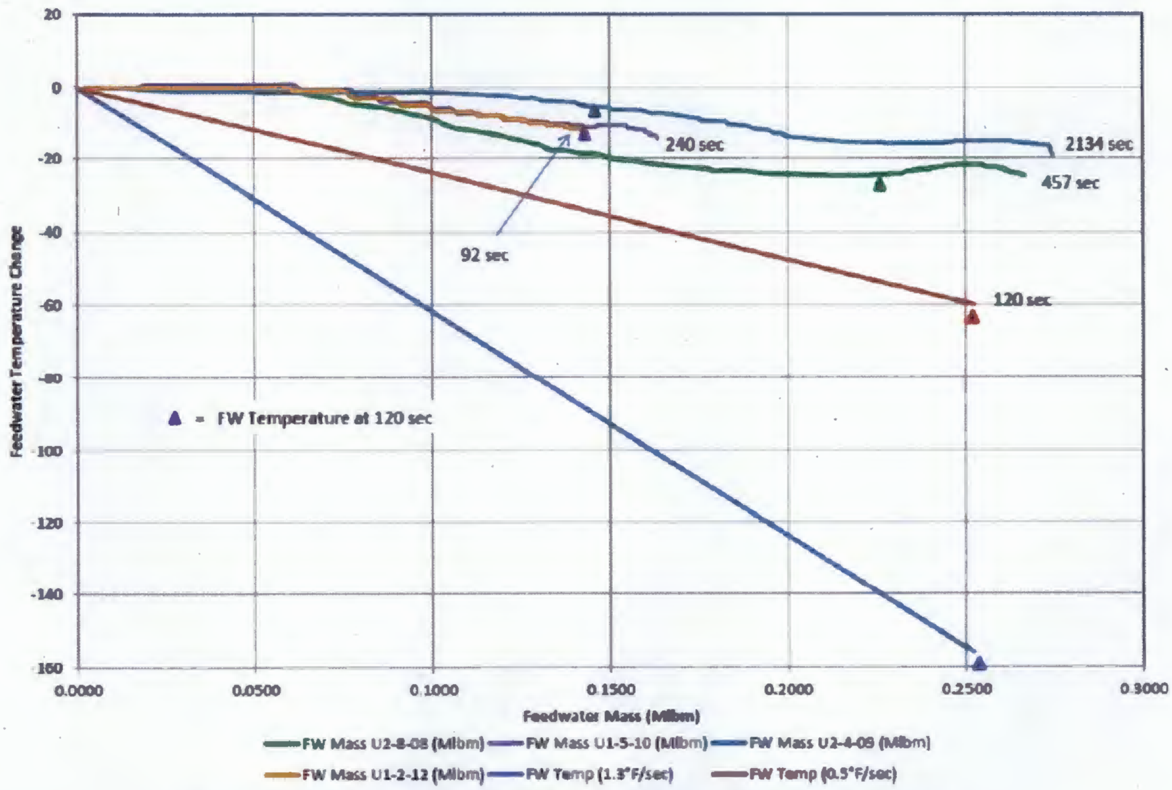

Figure 3.9.3.3-1: Feedwater Temperature Drop vs Mass

After the turbine trip, essentially no heat is added to or lost from the FW inventory between the condenser and RPV. Therefore, the NRC staff finds that the results in Figure SRXB-8-5 adequately represent the expected FW temperature response during an ATWS, based on consideration of measured plant data and fundamental principles of heat transfer. For this reason, the NRC staff finds the use of a 0.5°F/sec FW temperature reduction rate acceptable for the ATWS-I TTWBP analyses for BSEP MELLLA+.

Using the 0.5°F/sec FW temperature reduction rate along with the HN+CT T_{min} model, the maximum PCT was calculated to be 1733°F. Failure-to-rewet occurred in the simulation, but the slower decrease in FW temperature meant that the operator actions at 120 sec were successful in preventing the PCT from exceeding 2200°F.

The NRC staff examined an additional sensitivity case that used the 0.5°F/sec FW temperature reduction rate, the HN+CT T_{min} model, and the limiting fuel parameter sensitivity values. In this case, the increased oscillation growth rate meant that a cladding temperature of 2200°F was exceeded in at least one rod in 18 fuel bundles. Based on its evaluation in Appendix D of this SE, the NRC staff concludes that the limiting fuel parameter study constitutes a conservatively high PCT representation of ATRIUM-10XM fuel performance for ATWS-I, relative to the more realistic nominal fuel parameter values. Under the postulated conservative limiting fuel parameter assumptions, the NRC staff concludes that core coolability can be reasonably expected to be maintained given the relatively small number of fuel rods predicted to exceed

2200°F. This conclusion is consistent with previous staff evaluations for NEDO-32047A (Reference 21), in which the NRC staff concludes that no significant distortion of the core, impediment of core cooling, or prevention of safe shutdown was expected to occur for an ATWS TTWBP case in which localized cladding temperatures above 2200°F were predicted in as many as 88 bundles. For BSEP MELLLA+, the NRC staff concludes that core coolability is maintained during ATWS based primarily on the nominal fuel parameter case in which the PCT did not exceed 2200°F, with additional confidence provided by the limiting fuel parameter case in which core coolability is expected to be maintained even under conservative fuel parameter assumptions. As discussed earlier, the 2200°F limit in 10 CFR 50.46 was developed for LOCA events in which the limiting factor is the availability of coolant; in these events, the absence of coolant is expected to result in gross core heating. By contrast, the issue during ATWS-I events is not one of inadequate cooling but of high local energy deposition and cladding heat flux. In the event that cladding temperatures exceed 2200°F in only a limited length of the hottest fuel rods in a few assemblies in the core, no significant core distortion, loss of core coolability, or impaired ability to safely shut down the core is expected to occur. Therefore, the NRC staff considers the ATWS acceptance criterion to be satisfied under these circumstances. This position has been previously stated by the NRC staff in its evaluation of NEDO-32047A (Reference 21).

ATWS-I Two Reactor Pump Trip Consideration

The 2RPT ATWS events are similar to TTWBP ATWS events, with a primary difference being that the FW preheaters remain active during a 2RPT ATWS event. Therefore, the core inlet temperature in a 2RPT ATWS event remains significantly higher than in a TTWBP ATWS event, resulting in less severe limit cycle oscillations in the 2RPT ATWS event. In past MELLLA+ applications, the NRC staff has requested 2RPT ATWS analyses to be performed when the EOP mitigation actions in the TTWBP ATWS analyses occur sufficiently early to prevent large cladding temperature excursions associated with failure to rewet. Under these conditions, the 2RPT ATWS event may become the limiting ATWS-instability event because the operator action to reduce water level is assumed to occur longer after the initiating event in the 2RPT case than in the TTWBP case, which gives additional time for the oscillations to reach failure-to-rewet conditions. However, for BSEP MELLLA+, all TTWBP ATWS analyses with the HN+CT T_{min} model evaluated by the NRC staff exhibited failure-to-rewet in multiple bundles with sufficient time to reach high cladding temperatures before the EOP actions became effective. Therefore, the NRC staff concludes that the TTWBP ATWS analyses are bounding for BSEP MELLLA+.

NRC TRACE/PARCS Sensitivity/Confirmatory Modelling Results

To support the basis of the SE in this section, the NRC staff performed a plant-specific confirmatory analysis using TRAC/RELAP Advanced Computing Engine (TRACE) / Purdue Advanced Reactor Core Simulator (PARCS). The confirmatory study is summarized below. For full discussion of the confirmatory studies, see Reference 37.

The confirmatory analysis included a case matrix of sensitivity studies. This case matrix includes sensitivity studies to address differences in operator action timing, plant performance, and certain analysis inputs that may be subject to increased uncertainty owing to the hybrid methodology employed by the licensee in the LAR submittal. Specifically, the LAR includes analyses that are performed using a methodology developed by GEH but relies on core design parameters developed by AREVA for the core loading of ATRIUM-10XM fuel. Because the analysis vendor and fuel vendor are different in the LAR, the licensee relied on an analysis methodology that uses several sensitivity calculations to address the potential impact of fuel design related parameters. The NRC staff designed the case matrix to closely match the licensee's sensitivity calculations. Table 3.9.3.3-1 below is the case matrix for the confirmatory study.

Table 3.9.3.3-1: Case Matrix Description

Case	Description
1	This is the base case analysis, which assumed best estimate values for fuel related parameters and assumes licensing basis operator action times and plant performance parameters. Specifically, operators manually control water level at 120 seconds and feedwater temperature decreases at a rate of 1.3 °F/sec.
2-1	This case explores the sensitivity of the analysis results in "best" plant performance. It assumes very rapid operator action response time (96 seconds) to initiate manual level control and that the feedwater temperature following turbine trip is slow (-0.5 °F/sec).
2-2	This case explores the sensitivity of the analysis results to feedwater temperature response by assuming a slower feedwater temperature response following turbine trip (-0.5 °F/sec)

The NRC staff performed calculations 10 times for each case with varied core average gap conductance values that range from 3000 kW/m^2-K to 30000 kW/m^2-K in 3000 kW/m^2-K increments. The difference in gap conductance effectively adjusts the fuel thermal time constant, which impacts the timing of instability onset and transient cladding heat flux that can ultimately impact the PCT.

A summary of the most limiting results is found in Table 3.9.3.3-2. Plots of the PCTs are found in Figures 3.9.3.3-2, 3.9.3.3-3, and 3.9.3.3-4. These figures contain PCT trajectories for the candidate hot assemblies, CHAN-299 and CHAN-599, but for average rods within those assemblies. In the calculation it can be seen that these candidate hot assemblies show heat-up following the onset of large amplitude power/flow oscillations. However, since these assemblies do not include the core PCT, it highlights the difficulty in determining the core hot-spot a priori in these types of calculations. Additionally, the core PCT is also plotted in these figures.

The TRACE/PARCS calculations indicate that there is no fuel damage in Case 1 with a PCT of 2109 °F. As seen in Figure 3.9.3.3-2, the PCT increases early in the transient in response to the depressurization and 2RPT. This is a result of dryout followed by a mild fuel heat-up. PCT remains close to 900 °F until the onset of large amplitude power oscillations after 50 sec. The core PCT and CHAN-299 PCT responses show significant heat-up around the same time (~70 sec). Core PCT reaches ~2100 °F around 100 sec. Core PCT and candidate assembly PCTs begin to drop subsequently as manual operator actions begin to take effect and reduce core power and oscillation amplitude.

Figure 3.9.3.3-3 shows the PCT for the core and candidate hot assemblies for Case 2-1. For reference, the core PCT from Case 1 is also shown. While there are some differences that lead to an earlier rise in core PCT in Case 2-1 around 60 sec, the Case 2-1 PCT is ultimately lower. The differences in the calculations include the different FW temperature transient and the assumed gap conductance, a combination of these two effects would explain the difference in the initial PCT rise. The core PCT in Case 1 continues to increase through 100 sec and remains higher longer compared to Case 2-1. The effectiveness of level reduction in Case 2-1 is clearly shown in the hot assembly PCT curves that show decreasing PCT following level reduction around 100 sec.

PCT in the TRACE calculation is higher than the reference results provided in RAI SRXB-7, which the licensee reports a PCT of **[[** **]]** (Reference 2). However, these differences are most likely due to the prediction of early dryout and heatup in TRACE during the initial transient response. Since the TRACG results for dryout during a pressurization transient are reliable because TRACG methodology has been previously approved by the staff to conduct analyses of AOOs and certain ATWS scenarios and use approved CPR correlations, the lower temperature from TRACG are reasonable and the differences can be attributed to conservatism in the TRACE prediction of critical power. Without this early dryout the TRACE predictions would likely be in closer agreement with the TRACG results. A more reasonable basis for comparison, therefore, would be the CHAN-299 PCT, which reaches about 1600 °F. This temperature is lower also because the signal is based on an averaged powered rod in CHAN-299, but this rod is compared because it is not subject to early dryout. The CHAN-299 heat-up occurs during the unstable phase with the PCT increasing around 75 sec – leading to only a short time (~20 sec) to achieve PCT before operator intervention. This is more consistent with the TRACG predictions that do not show an early heat up during the initial phase of the transient. In either case, the TRACE calculations show margin to fuel damage.

The PCT is much lower in the TRACG calculation because there is a slight difference in the prediction of the timing of instability onset – with TRACE predicting instability whereas TRACG predicts essentially no large amplitude oscillations due to early operator intervention. As will be shown in Case 2-2, if the operator intervention is a little later, the TRACE and TRACG results are in much closer agreement.

Figure 3.9.3.3-4 compares the Case 2-2 and Case 2-1 PCT responses. Because the heat transfer coefficient of the gap between the fuel pellet and cladding (HGAP) and FW temperature ramps are the same in Case 2-1 and Case 2-2, the responses are identical prior to manual action to control level. Major trends are consistent in Case 2-2 with Case 2-1 except that the FW flow drops later, resulting in later level reduction and later FW flow restoration. A comparison of Case 2-2 with results provided in the response to RAI SRXB-7 indicate relatively good agreement in plant parameters as well as PCT. The TRACE predicted result is 1988 °F compared to **[[** **]]** in provided by the licensee. The TRACE result can be expected to be higher due to the early dryout leading to higher PCT in the TRACE calculation compared to TRACG. Therefore, a more reasonable basis for comparison is the CHAN-299 response, which is a hot assembly that does not undergo the early dryout shown in the Core PCT response. The CHAN-299 PCT is 1646 °F. This value is an underestimate because the CHAN-299 signal is based on an average powered rod instead of a hot rod. Therefore, removing the early dryout effect means that the PCT would be between 1646 °F and 1988 °F, which compares well with the TRACG result of **[[** **]]**. These results are in closer agreement because they are not as sensitive to the difference in instability onset timing since the operator action to lower level occurs after heat-up in both cases.

Overall, results of the NRC staff's confirmatory analysis do not indicate fuel damage, and therefore, confirm acceptable performance. These results support the NRC staff's conclusion that, with respect to ATWS-I and associated mitigating operator actions, BSEP operation in the MELLLA+ is acceptable.

Table 3.9.3.3-2: Summary of ATWS-I Confirmatory Results

Case	HGAP (kW/m^2-K)	PCT (°F)
1	9000	2109
2-1	12000	1988
2-2	12000	1988

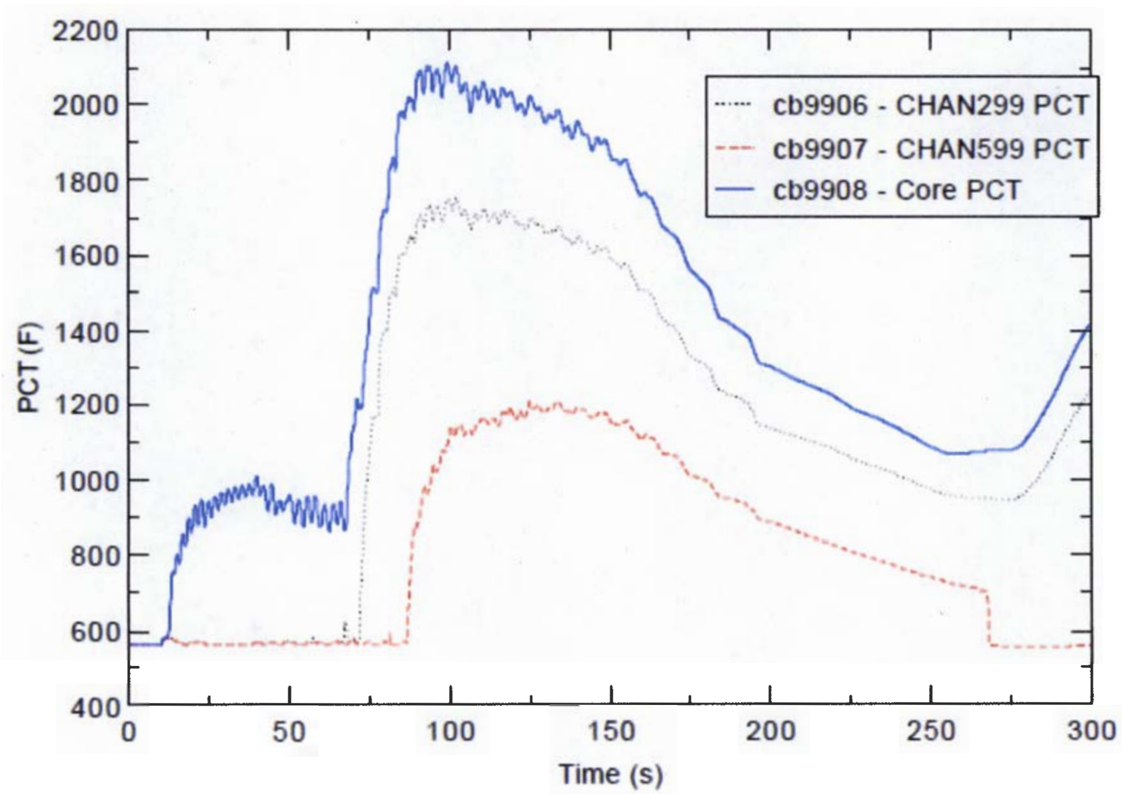

Figure 3.9.3.3-2: Case 1 – Peak Cladding Temperature

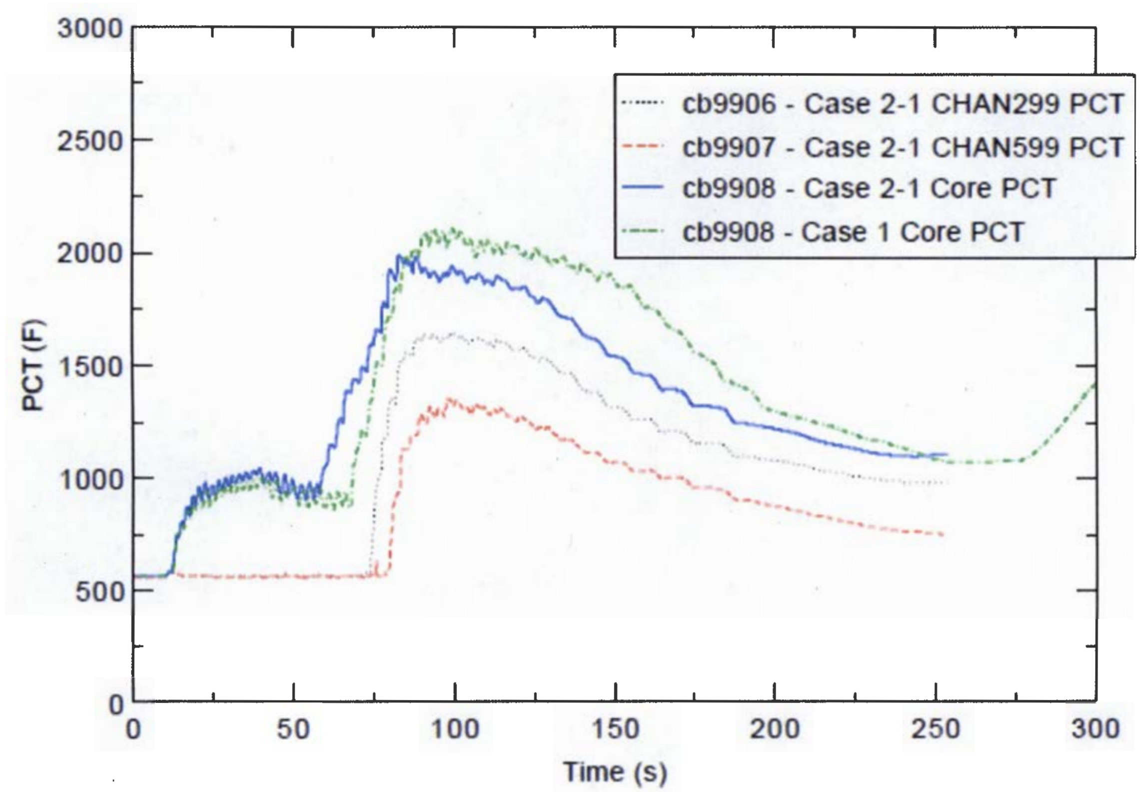

Figure 3.9.3.3-3: Case 2-1 – Peak Cladding Temperature

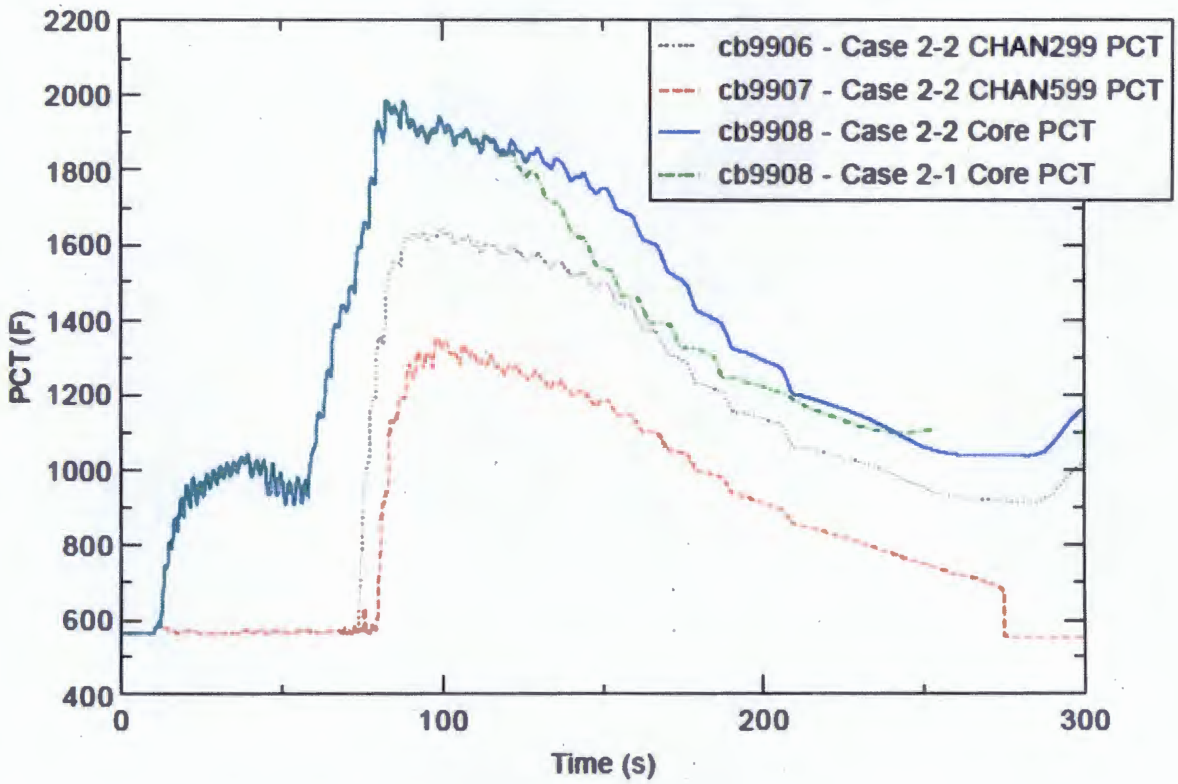

Figure 3.9.3.3-3: Case 2-2 – Peak Cladding Temperature

Conclusions on ATWS

The NRC staff concludes that the plant design and operator actions adequately addresses ATWS events and meets the SRP acceptance criteria and the requirements of 10 CFR 50.62. This conclusion is based on the following:

- The applicant's plant design includes the ATWS risk reduction features prescribed by the ATWS Rule.
- These features are independent and diverse from the reactor trip system and are designed to be reliable, as required under the ATWS rule.
- The licensee has also provided or referenced information, analyses, and/or evaluations that demonstrate that limiting ATWS and event sequences have been considered and that features included in the design pursuant to the rule result in reasonable assurance, that unacceptable plant conditions, as defined during the rulemaking, will not occur because of ATWS events.
- Results of the NRC staff's confirmatory analysis do not indicate fuel damage, and therefore, confirm acceptable system performance.

3.10 BSEP SAR Section 10.0, "Other Evaluations"

3.10.1 BSEP SAR Section 10.1, "High Energy Line Break"

3.10.1.1 BSEP SAR Section 10.1.1, "Steam Lines"

The licensee confirmed that the generic M+ LTR treatment of the high energy line break (HELB) steam lines topic is applicable to BSEP. Specifically, a review of the heat balances produced for the BSEP MELLLA+ operation confirms there is no effect on the steam pressure or enthalpy at the postulated break locations (e.g., MS, HPCI, and RCIC).

The NRC staff concludes that the generic M+ LTR resolution is applicable to BSEP because MELLLA+ has no effect on the mass and energy releases from a HELB in a steam line.

3.10.1.2 BSEP SAR Section 10.1.2, "Balance-of-Plant Liquid Lines"

The licensee confirmed that the generic M+ LTR treatment of the HELB balance-of-plant (BOP) liquid lines topic is applicable to BSEP. Specifically, a review of the heat balances produced for BSEP MELLLA+ operation confirmed there is no effect on the liquid line conditions at the postulated FW break locations. In addition, the mass and energy release for operation in the MELLLA+ domain is not affected.

The NRC staff concludes that the generic M+ LTR resolution is applicable to BSEP because the current operating conditions are bounding with respect to mass and energy releases from a HELB in the BOP liquid lines.

3.10.1.3 BSEP SAR Section 10.1.3, "Other Liquid Lines"

The licensee's generic resolution of the HELB of other liquid lines topic in the M+ LTR states that [[

]] The effects on subcompartment pressures and temperatures, pipe whip, jet impingement, and flooding are included in the scope of liquid line break evaluations in the MELLLA+ operating domain.

The licensee also stated that the heat balances for the BSEP MELLLA+ operating domain confirm that there is no effect on the liquid line conditions (excluding FW addressed in Section 10.1.2) at the postulated break locations. [[

]] The scope of these evaluations includes MELLLA+ operating domain expansion effects on the subcompartment pressures and temperatures, pipe whip, jet impingement, and flooding, consistent with the plant licensing basis.

The NRC staff concludes that the BSEP-specific evaluation for the HELB in other liquid lines is acceptable because the current licensing basis analysis for the most limiting break, bounds the liquid line break analysis in the MELLLA+ operating domain.

3.10.2 BSEP SAR Section 10.2, "Moderate Energy Line Break"

The licensee stated in the BSEP SAR that the moderate energy line breaks are not included in BSEP current licensing basis (i.e., UFSAR). Therefore, based on the current UFSAR description, the NRC staff concludes moderate energy line break in the M+ LTR is not applicable to BSEP.

3.10.3 BSEP SAR Section 10.3, "Environmental Qualification"

3.10.3.1 BSEP SAR Section 10.3.1, "Electrical Equipment"

The licensee confirmed that the generic M+ LTR treatment of the electrical equipment environmental qualification (EQ) topic is applicable to BSEP. Specifically, for BSEP under the MELLLA+ operating conditions, there is no change in reactor power, radiation levels, decay heat, reactor operating pressure, MS flow rate, or FW flow rate. In addition, **[[**

]]

The NRC staff concludes that the generic M+ LTR resolution is applicable to BSEP because the EQ of electrical equipment is unaffected by operation in the MELLLA+ operating domain.

3.10.3.2 BSEP SAR Section 10.3.2, "Mechanical Equipment with Non-Metallic Components"

The licensee confirmed that the generic M+ LTR treatment of the mechanical equipment with non-metallic components EQ topic is applicable to BSEP. Specifically, implementing MELLLA+ does not change the normal process temperatures or radiation levels in any of the plant areas where safety-related equipment is located. **[[**

]]

The NRC staff concludes that the generic M+ LTR resolution is applicable to BSEP because the EQ of mechanical equipment with non-metallic components is unaffected by operation in the MELLLA+ operating domain.

3.10.3.3 BSEP SAR Section 10.3.3, "Mechanical Component Design Qualification"

The licensee confirmed that the generic M+ LTR treatment of the mechanical component design qualification topic is applicable to BSEP. Specifically, implementation of MELLLA+ does not change normal process temperatures, pressures, and flow rates. **[[**

]]

The NRC staff concludes that the generic M+ LTR resolution is applicable to BSEP because mechanical component design qualification is unaffected by operation in the MELLLA+ operating domain.

3.10.4 BSEP SAR Section 10.4, "Testing"

As discussed in the NRC staff SE for the M+ LTR, when the MELLLA+ operating range expansion is implemented, plant-specific testing will be performed to confirm operational performance and control aspects of the MELLLA+ changes.

Section 10.4 of the BSEP SAR provides a brief description of the following plant-specific testing for implementation of the BSEP MELLLA+:

- Testing will be performed for steam separator-dryer performance similar to the original plant startup test program. The testing will be performed to determine the magnitude and trend of MCO.

- The APRM system will be calibrated and functionally tested to confirm that the trips, alarms, and rod blocks perform as intended in the MELLLA+ operating domain.

- A core performance test will be performed to evaluate the results of core thermal power, fuel thermal margin, and core flow performance against projected values and operational limits.

- A pressure control system test will be performed to confirm that the settings, established for operation with the current power versus flow upper boundary at CLTP, are adequate in the MELLLA+ operating domain. No changes to current settings are expected.

- Reactor water level changes will be introduced into the FW control system to verify the FW control system can provide acceptable reactor water level control in the MELLLA+ operating domain.

- A neutron flux surveillance test will verify that the neutron flux noise level in the reactor is within expectations in the MELLLA+ operating domain.

The NRC staff finds these tests to be acceptable because they will help confirm that plant operation is consistent with the analyses performed and reviewed to support the safe operation in the proposed MELLLA+ domain.

3.10.5 BSEP SAR Section 10.5, "Individual Plant Examination"

The licensee provided a plant-specific probabilistic risk assessment (PRA) in accordance with the M+ LTR Limitation and Condition 12.21.

The NRC staff reviewed the LAR and determined that it was not risk-informed but did provide risk insights related to the implementation of MELLLA+. Specifically, the licensee augmented the generic risk discussion contained in the M+ LTR with plant-specific information on initiating event frequencies, component reliability, operator response, success criteria, external events, shutdown risk, and PRA quality. The licensee reported an increase in core damage frequency (CDF) of 1.3E-7/year for Unit 1 and 1.17E-7/year for Unit 2 and an increase in large early release frequency (LERF) of 4.6E-8/year for Unit 1 and 4.56E-8/year for Unit 2.

Consistent with the NRC's guidance on non-risk-informed LARs (SRP, Chapter 19.2, Appendix D), the NRC staff reviewed SAR Section 10.5 to determine whether "special circumstances" were present (e.g., a risk increase exceeding the RG 1.174 acceptance guidelines) that would warrant a more detailed risk evaluation. Based on the risk information provided by the licensee, the NRC staff concluded that the expected increase in risk associated with implementation of MELLLA+ at BSEP would be well within the risk acceptance guidelines delineated by RG 1.174. Therefore, the NRC staff's review did not identify any "special circumstances" that would warrant an in-depth PRA review.

3.10.6 BSEP SAR Section 10.6, "Operator Training and Human Factors"

The regulatory guidance that the NRC staff considered in its review regarding operator training and human factors are as follows:

- NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," Chapter 18
- NUREG-1764, "Guidance for the Review of Changes to Human Actions"
- NUREG-0711, "Human Factors Engineering Program Review Model"

The NRC staff reviews the human performance aspects of LAR using the review guidance in Reference 36. In accordance with the generic risk categories established in Appendix A to NUREG-1764, the tasks under review are involved in the safety injection sequence and actions involving risk-important systems, and are, therefore, considered "risk-important." Due to this risk importance, the NRC staff will perform a "Level One" review, the most stringent of the graded reviews possible under the guidance of NUREG-1764. Note: This assessment of risk is only for purposes of scoping the human factors review and may not necessarily align with the licensee's assessment of risk importance or that of other portions of the NRC staff review. This assessment is not intended to be equivalent to the assessment of risk performed with other methods, especially those using plant-specific data and NRC-accepted methods of probabilistic risk analysis and human reliability analysis.

Description of Operator Action(s) Added/Changed/Deleted

The licensee stated in BSEP SAR Section 10.5.3, "Operator Response" that there are no new operator actions to be added to operating procedures and there is no significant reduction in the time for operator actions. However, the existing operator action to initiate lowering Reactor Pressure Vessel (RPV) water level within 120 sec assumed in the safety analysis to mitigate ATWS events will be classified as a Time Critical Operator Action (TCOA). This action is being classified as a TCOA due to the change in the dynamics of ATWS instability events associated with the MELLLA+ operating domain expansion.

Operating Experience Review

The licensee stated that the BSEP MELLLA+ application used the following NRC-approved GEH Licensing Topical Reports (LTRs):

- NEDC-33006P-A, (M+ LTR), Revision 3 and its associated SE.
- NEDC-33173P-A, (Methods LTR), Revision 4 and its associated SE.
- NEDC-33075P-A, (DSS-CD LTR), Revision 8 and its associated SE.

In addition to using NRC-approved methodologies to develop the BSEP MELLLA+ implementation at BSEP, the licensee provided a list with summaries of industry precedents considered including Monticello Nuclear Generating Plant (March 28, 2014), Grand Gulf Nuclear Station (August 31, 2015), Nine Mile Point Nuclear Station, Unit No. 2 (September 2, 2015) and Peach Bottom Atomic Power Station (Mar 2016).

The NRC staff concludes that the licensee's analysis using NRC approved methodology and industry precedents is acceptable to address operating experience resulting from the MELLLA+ implementation.

Functional Requirements Analysis and Function Allocation

The licensee stated that the process governing changes and the addition of operator requirements is part of the configuration change control process at BSEP. This process provides the necessary direction and guidance to evaluate configuration changes to the facility, including impact assessments that identify procedures and training material that require revisions for the planned configuration change.

Implementation of MELLLA+ at BSEP does not replace any existing automatic functions with manual actions or vice versa. However, a new automatic function, automated backup stability protection (ABSP), is being added by the power range neutron monitoring (PRNM) system modification (Detect and Suppress Solution - Confirmation Density (DSS-CD)). The ABSP function is a backup to the DSS-CD function in the event that the DSS-CD function is not available.

The NRC staff concludes that the licensee's configuration control process is sufficient to address changes and additions to operator requirements resulting from the MELLLA+ amendment.

Task Analysis

BSEP operations with the MELLLA+ improvements do not change the required operator actions or significantly reduce the time for operator actions. As there have been no changes to operator actions or functions, no new task analysis was performed.

The NRC staff concludes that revision of the licensee's task analysis is not necessary because the actions associated with this proposed change are not new and are proceduralized. In addition, the existing actions are straight forward and do not require changes to physical interfaces.

Staffing

The licensee stated that no new or additional operator actions are required to implement MELLLA+ at BSEP. Nor are there any new or additional qualifications required to perform the actions within the unchanged time constraints. Therefore, operation in the MELLLA+ domain is not expected to increase operator workload. Because no additional staffing or qualifications or changes are needed, the NRC staff finds the licensee's staffing plan to implement MELLLA+ to be acceptable.

Probabilistic Risk and Human Reliability Analysis

As discussed in Section 3.10.5 (Individual Plant-Examination), the NRC staff reviewed the risk information provided by the licensee and concluded that the expected increase in risk associated with implementation of MELLLA+ at BSEP would be well within the risk acceptance guidelines delineated by RG 1.174. Therefore, the NRC staff did not identify any "special circumstances" that would warrant an in-depth PRA review.

Human-System Interface Design

Implementation of the MELLLA+ expansion at BSEP involves changes to the main control room computer display of the power/flow map. In addition, Power Range Neutron Monitoring System requires hardware and software changes through implementation of the DSS-CD solution, including an ABSP. Therefore, some changes are required to main control room panel board alarm settings and automatic actuation setpoints to support the MELLLA+ operating domain expansion. However, the licensee stated that these changes do not involve major physical changes to the main control room controls, displays or alarms.

Based on the above and the onsite audit in February 2018, the NRC staff finds that there are no substantial changes to the human-system interface design associated with the implementation of the MELLLA+ expansion and the licensee's treatment of this review element is acceptable.

Procedure Design

As part of the implementation of the MELLLA+ amendments upon approval, the licensee stated that necessary changes to procedures will be consistent with existing Duke Energy's configuration change control process for other plant modifications, including evaluations to determine the specific changes required. Training and implementation requirements, including any effects on the simulator, will be evaluated. Simulator changes and fidelity validation will be performed in accordance with applicable American National Standards Institute (ANSI) standards currently being used at the training simulator.

The NRC staff concludes that, because existing licensee processes for updating procedures and training operators must satisfy the requirements in 10 CFR 50.59 and 10 CFR 50.120, as well as the approved quality assurance program, they are acceptable to address the impact of MELLLA+ implementation.

Training Program Design

The licensee stated that changes to operator training and the plant simulator will be identified and incorporated into the BSEP MELLLA+ implementation plan. Per the licensee's supplement dated March 29, 2018 (Reference 35), BSEP simulator changes and fidelity validation will be performed in accordance with ANSI/ANS 3.5-2009.

As part of the implementation of the MELLLA+ amendments upon approval, the licensee is required to update the BSEP training program in accordance with Duke Energy's current plant training program requirements. Classroom training to address the various aspects of operation in the MELLLA+ expansion will be conducted prior to operation in the MELLLA+ domain. Plant operating experience, once MELLLA+ is implemented, will be evaluated to determine the need for additional training. While specific simulator training for plant transients is not anticipated since the plant dynamics do not substantially change for MELLLA+ operation, enhanced training will be provided for ATWS event mitigation in the MELLLA+ domain.

The approach described above is consistent with the current licensing basis and uses approved methods to incorporate any changes to the human-system interface, procedures, and operational considerations into the training program. Therefore, the NRC staff finds the licensee's treatment of the training program to be acceptable.

Human Factors Verification and Validation

The BSEP simulator has been updated to reflect the MELLLA+ analysis to support the implementation of the amendment. Additionally, procedure revisions will be completed as part of the implementation in accordance with the licensee's configuration change control process. As discussed above, operators have completed training associated the MELLLA+ analysis. The BSEP supplement dated March 29, 2018, stipulates that operators are required to initiate lowering RPV water level to mitigate ATWS instability events within 120 sec. During an audit by NRC staff at BSEP in February 2018, the NRC staff observed ATWS and DSS-CD manual backup stability protection scenarios for timing validation. The demonstrations showed the operators successfully initiating FW flow reduction well within the allowed time. The NRC staff also observed successful initiation of the DSS-CD manual backup when the operators recognized (or where made aware) that the automatic BSP was inoperable. This demonstration provides reasonable assurance that the actions are feasible within the time constraints.

The results of the MELLLA+ human factors review determined that changes to plant procedures will not alter the current mitigation strategies. Changes associated with setpoints will not introduce a level of complexity that would lead to misunderstanding the parameters.

Per the licensee's submittal the BSEP MELLLA+ implementation plan will determine the changes required to implement MELLLA+ consistent with Duke Energy's current plant training program requirements. The operator training program and plant simulator will be evaluated to determine the specific changes required.

Based on the above, the NRC staff finds the licensee's treatment of human factor verification and validation to be acceptable.

Human Performance Monitoring Strategy

The change control process includes a review by operations and training personnel. Training and implementation requirements are identified and tracked, including effects on the simulator and verification of training is required as part of the design change closure process. Operator actions in response to an ATWS with MELLLA+ remain consistent with the current operator actions without MELLLA+. While no new operator actions are involved for MELLLA+ to be implemented, the 120-sec time requirement to initiate lowering RPV water level to mitigate ATWS instability events will be classified as a TCOA and tracked and managed in accordance with BSEP plant procedure 0AP-064, "Time Critical Operator Actions."

Based on the above and existing TS 5.4 "Procedures" requirements to control this type of procedure, the NRC staff finds the licensee's treatment of human performance monitoring strategy to be acceptable.

Overall, the NRC staff finds the proposed MELLLA+ amendment to be acceptable with respect to operator training and human factors.

3.10.7 BSEP SAR Section 10.7, "Plant Life"

3.10.7.1 BSEP SAR Section 10.7.1, "Irradiated Assisted Stress Corrosion Cracking"

The licensee confirmed that the generic resolution in the M+ LTR with respect to irradiated assisted stress corrosion cracking (IASCC) is applicable to BSEP. Specifically the life of most equipment is not affected by the MELLLA+ operating domain. Fluence calculations for the reactor internals indicate that the top guide, core plate, and shroud exceed the fluence threshold limit that could potentially lead to a minor increase in IASCC. The licensee's current inspection for these reactor internals follows the guidance recommended in BWR Vessel and Internal Project (BWRVIP) -25, -26, and -76, which are based on component configuration and field experience. This inspection strategy is adequate to manage a minor increase in the IASCC potential.

Because the licensee continues to follow the guidance in the areas of detection, inspection, repair, or mitigation recommended in BWRVIP-25, -26, -76, and -183 to ensure long-term function of components affected by fluence in the MELLLA+ operating domain, the NRC staff concludes that the licensee's resolution of the effects of IASCC due to MELLLA+ is acceptable.

3.10.7.2 BSEP SAR Section 10.7.2, "Flow Accelerated Corrosion"

The licensee confirmed that the generic M+ LTR treatment of the flow-accelerated corrosion (FAC) topic is applicable to BSEP. Specifically, for BSEP, there are no significant changes in MS or FW temperatures or MS or FW flow rates in the MELLLA+ operating domain compared to current plant operating conditions. As discussed in BSEP SAR Section 3.3.3, MCO values under MELLLA+ conditions may increase in the MS lines, which may slightly increase the FAC rates for a small period of time during the cycle when the plant is operating at or near the MELLLA+ minimum core flow. **[[**

]] The licensee stated that:

> The evaluation of and inspection for flow-induced erosion/corrosion in piping systems affected by FAC is addressed by compliance with NRC GL 89-08 [Erosion/Corrosion-Induced Pipe Wall Thinning]. The requirements of GL 89-08 are implemented at BSEP by utilization of the Electric Power Research Institute generic program, CHECWORKS^{TM}. BSEP-specific parameters are entered into this program to develop requirements for monitoring and maintenance of specific system components. No changes are required to the BSEP specific parameters that are entered into the CHECWORKS^{TM} program....
>
> In addition to FAC, a periodic non-destructive examination for the inspection of safety-related piping and heat exchangers at known or suspected high corrosion,

> biofouling or silt buildup areas in response to GL 89-13 [Service Water System Problems Affecting Safety-Related Equipment]....
>
> The Maintenance Rule (10 CFR 50.65) provides oversight for other mechanical and electrical equipment important to safety, to monitor performance and protect against age-related degradation. The longevity of the effects of FAC in the MELLLA+ operating domain at BSEP equipment is not affected by the MELLLA+ operating domain expansion.
>
> In the response to EMIB-RAI-1, the licensee also stated that the plant's flow accelerated corrosion (FAC) program, which includes the main steam line piping, monitors susceptible areas for corrosion and factors the results into the piping replacement program at the plant site so that any adverse impact on the MSL piping will be monitored by the plant.
>
> The NRC staff finds that the generic M+ LTR is applicable and the plant specific MCO in MELLLA+ is addressed by licensee's existing FAC program.

Because FAC under MELLLA+ operating conditions is bounded by the current plant operation, the NRC staff concludes that the generic M+ LTR resolution is acceptable.

3.10.8 BSEP SAR Section 10.8, "NRC and Industry Communications"

The licensee confirmed that the generic M+ LTR treatment of the NRC and industry communications topic is applicable to BSEP. The licensee stated that:

> Because these evaluations of plant design and safety analyses inherently included any effects as a result of NRC and industry communications, it is not necessary to review prior communications and no additional information is required in this area.

The NRC staff concludes that the licensee's incorporation of NRC and industry communications related to MELLLA+ design is acceptable because the NRC staff did not identify any NRC-industry communications, such as operation experience of MELLLA+ operations at other plants and Part 21 reports related to the M+ LTR, to suggest BSEP falls outside the applicability scope of the original M+ LTR.

3.10.9 BSEP SAR Section 10.9, "Emergency and Abnormal Operating Procedures"

The licensee stated that EOPs and abnormal operating procedures (AOPs) can be affected by operating in the MELLLA+ domain. The EOPs include variables and limit curves, which define conditions where operator actions are indicated. The EOPs are symptom-based. AOPs include event-based operator actions.

The licensee also stated that the EOPs and AOPs will be reviewed for any effect due to MELLLA+ operation and revised prior to MELLLA+ implementation. In addition, any changes to these procedures will be included in operator training to be conducted prior to implementation of the MELLLA+ amendment.

The NRC staff concludes that, because existing licensee processes for updating procedures and training operators must satisfy the requirements in 10 CFR 50.59 and 10 CFR 50.120, the process described above are acceptable to address the impact of MELLLA+ implementation on the EOPs and AOPs.

4.0 RENEWED FACILITY OPERATING LICENSE AND TECHNICAL SPECIFICATION CHANGES

In the LAR, the licensee proposed changes to both Appendix A, Technical Specifications and license condition in Appendix B, "Additional Conditions" of the BSEP Renewed Facility Operating Licenses. The detailed evaluation of these proposed changes are discussed in Section 3.0 of this safety evaluation.

4.1 License Condition - Feedwater Temperature

The licensee requested that Renewed Facility Operating License No. DPR-71 (Unit 1) be amended by the addition of a new license condition to Appendix B, "Additional Conditions," as indicated in the attachment to this amendment, which reads as follows:

Amendment Number	Additional Conditions	Implementation Date
285	The licensee shall not operate the facility within the MELLLA+ operating domain with Feedwater Temperature Reduction (FWTR) as defined in the Core Operating Limits Report.	Upon implementation of Amendment No. 285

The licensee also requested the Renewed Facility Operating License No. DPR-62 (Unit 2) be amended by the addition of a new license condition to Appendix B, "Additional Conditions," as indicated in the attachment to this amendment, which reads as follows:

Amendment Number	Additional Conditions	Implementation Date
313	The licensee shall not operate the facility within the MELLLA+ operating domain with Feedwater Temperature Reduction (FWTR) as defined in the Core Operating Limits Report.	Upon implementation of Amendment No. 313

The licensee proposed these conditions to address M+ LTR L&C 12.5.b, which states:

> 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 stated that this limitation is applicable. Before the MELLLA+ operating domain, BSEP has the operational flexibility of final FW temperature reduction (FFWTR) and FWHOOS at MELLLA but not at the MELLLA+ domain. Instead of a new TS LCO restriction, the licensee proposed the above license conditions to prevent FW temperature reduction in the MELLLA+ domain and performed evaluations to determine that this license condition is acceptable for BSEP MELLLA+. The NRC staff reviewed the evaluation and finds that the conditions are acceptable, because they are supported by the BSEP SAR and M+LTR and will ensure safe operations in the4 MELLLA+ domain.

4.2 Technical Specification Changes

The NRC staff evaluated the following TS changes proposed for Units 1 and 2 in the licensee's LAR the sections of this SE, as noted:

- TS 3.1.7 Standby Liquid Control (SLC) System

  Change sodium pentaborate B-10 enrichment from ≥ 47 atom percent to ≥ 92 atom percent in SR 3.1.7.8 and Figure 3.1.7-1:

  SR 3.1.7.8, which reads: Verify sodium pentaborate enrichment is ≥ 92 atom percent B-10. FREQUENCY: Prior to addition to SLC tank

  Figure 3.1.7-1 (page 1 of 1) Sodium Pentaborate Solution Volume Versus Concentration Requirements

  The NRC staff reviewed the licensee's analyses related to the effects of the proposed operating domain extension on the SLC system and associated TS 3.1.7 change above to reflect the increased enrichment. Based on the result of the review in SE Sections 3.2.3 and 3.4.2.1 of this SE, the NRC staff finds that this higher B-10 enrichment SLC system is acceptable to meet the requirements of 10 CFR 50.62(c)(4) and GDC 26 following implementation of the proposed operating domain extension.

- TS 3.3.1.1, Reactor Protection System (RPS) Instrumentation

  Change Required Action I.1 a single action to initiate an alternate method of detecting and suppressing thermal hydraulic instability to three separate actions as follows:

  I.1 Initiate action to implement the Manual BSP Regions defined in the Core Operating Limits Report (COLR). (Completion Time: Immediately)

  AND

  I.2 Implement the Automated BSP Scram Region using the modified APRM Simulated Thermal Power - High scram setpoints defined in the COLR. (Completion Time: 12 Hours)

  AND

  I.3 Initiate action in accordance with Specification 5.6.7. (Completion Time: Immediately)

  Based on the evaluation in Sections 3.2.4 and 3.5 above, the NRC staff finds these changes are consistent with the approved LTR ensure that safety limits are met, and, are therefore, acceptable.

- TS 3.3.1.1, Required Actions J.1, J.2, and J.3

  Change Required Action J.1 from one action to three, to address the situation where Required Action and associated Completion Time of Condition I is not being met. Based on the evaluation in Section 3.2.4 and 3.5 of this SE, the NRC staff finds these changes are consistent with the approved LTR, ensure that safety limits are met, and are therefore, acceptable.

- TS 3.3.1.1, Required Action K.1

  Add a new required action K.1 to address the situation where the Completion Time of Condition J is not met. The action is to reduce THERMAL POWER to less than 18% of Rated Thermal Power and this action must be completed within 4 hours. Based on the evaluation in Section 3.2.4 and 3.5 of this SE, the NRC staff finds these changes are consistent with the approved LTR, ensure that safety limits are met, and are therefore, acceptable.

- TS 3.3.1.1, Surveillance Requirement (SR) 3.3.1.1.19

  Delete this SR. The licensee stated that this requirement is no longer needed because the DSS-CD function is designed to automatically arm itself when plant conditions require it. The automatic arming functionality of the DSS-CD trip capability is described in Section 3.1 of the Approved LTR (Reference 8). Based on the evaluation in Section 3.2.4 and 3.5 above, the NRC staff finds these changes are consistent with the approved LTR, ensure that safety limits are met, and are therefore, acceptable.

- TS 3.3.1.1, Table 3.3.1.1-1, Function 2.b

  Change the allowable value for Function 2.b in Table 3.3.1.1-1, "Simulated Thermal Power – High" from ≤ .55W+ 62.0% Reactor Thermal Power (RTP) to ≤ .61W+ 65.2% RTP. In addition, add a note (e) to address the OPRM Upscale function inoperable condition.

  The revised allowable value formula reflects the changed curve for determining the Simulated Thermal Power trip setpoint based on power level and core flow. Based on the evaluation in Section 3.2.4 and 3.5 of this SE, the NRC staff finds these changes are consistent with the approved LTR to ensure that safety limits are met, therefore, acceptable.

- TS 3.3.1.1, Table 3.3.1.1-1, Function 2.f

  Change the specified condition associated with Function 2.f of Table 3.3.1.1-1 from ≥ 20% RTP to ≥ 18% RTP and a new Footnote (f) is added to indicate an exception to the arming requirements of the DSS-CD function during the first reactor startup and first controlled shutdown that passes completely through the DSS-CD Armed region.

  Surveillance 3.3.1.19, which required a periodic verification that the OPRM is not bypassed when the APRM Simulated Thermal Power is ≥ 25% and recirculation flow is ≤ 60%, is deleted. This Surveillance is no longer required because DSS-CD functions automatically arm when pre-determined conditions are met.

  Modify footnote (d) to reflect the change from Period Based Detection algorithms to Confirmation Density Algorithms, which will be credited.

  Based on the evaluation in Section 3.2.4 above, the NRC staff finds these changes are consistent with the approved LTR, ensure that safety limits are met, and are therefore, acceptable.

- TS 3.4.1, Recirculation Loops Operating, LCO 3.4.1

  Revise TS LCO to prohibit single recirculating loop operation when the reactor is in the MELLLA+ operating domain. Based on the evaluation in Section 3.2.4 above, the NRC staff finds these changes are consistent with the approved LTR, ensure that safety limits are met, and are therefore, acceptable.

- LCO 3.4.1, Conditions B and C

  Reassign the current Condition B and Required Action B.1 of LCO 3.4-1 as Condition C with Required action C.1

  Add a new Condition B with Action B.1 to LCO 3.4-1, which states that Operation in the MELLLA+ domain with a single recirculation loop in operation will require an immediate action to exit the MELLLA+ operating domain.

  As stated in Section 3.2.4 above, the NRC staff finds these changes are consistent with the approved LTR, ensure that safety limits are met, and are therefore, acceptable.

- TS 5.6.5, Core Operating Limits Report (COLR), Item a.4 and Item b.19

  Replace Item a.4 the period based detection algorithm (PBDA) setpoint requirement with a new reference to the DSS-CD LTR to reflect the new COLR setpoint requirements associated with the DSS-CD reactor trip function. The PBDA will no longer be credited in the safety analysis.

  Also, replace Item b.19 NEDO-32465-A with NEDC-33075P-A to reflect the change in the approved analytical method associated with the DSS-CD methodologies.

  Based on the review in Section 3.2.4 of this SE, the NRC staff finds these changes are consistent with the approved LTR, ensure that safety limits are met, and are therefore, acceptable.

  Add a new TS 5.6.7, Oscillation Power Range Monitor (OPRM) Report to specify when a report required by Condition I of LCO 3.3.1.1, RPS Instrumentation, shall be submitted and what the contents of this report shall be.

Based on the evaluation in Section 3.2.4 of this SE, the NRC staff finds these changes are consistent with the approved LTR, ensure that safety limits are met, and are therefore, acceptable.

5.0 TECHNICAL EVALUATION CONCLUSION

The NRC staff reviewed the licensee's analyses related to the effect of the proposed amendment for BSEP to operate in the MELLLA+ domain. Based on its review, the NRC staff concludes from this review that the broadening of the BSEP operating domain by lowering the flow at high powers without additional limitations would reduce the safety margin. However, the licensee has proposed the following solutions in the BSEP SAR that are technically acceptable to satisfy the regulatory criteria while operating in the MELLLA+ domain:

- FWHOOS and FFWTR operation is prohibited in the MELLLA+ domain by a license condition (added to Appendix B "Additional Conditions" of the renewed operating licenses as Amendment Nos. 285 and 313 for Unit 1 and Unit 2, respectively).

- SLO is prohibited in the MELLLA+ domain by a license condition.

To provide additional protection against spurious, noise-induced scrams on the DSS-CD system, **[[**

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- The SLC system Boron enrichment was increased to reduce the integrated heat load to containment during ATWS under MELLLA+ conditions.

- Operator actions will be credited in the MELLLA+ ATWS analyses for water level reduction (time-critical operator action) and SLC system Boron injection. These are unchanged from the current EOPs.

Additionally, the NRC staff concludes that the use of TRACG, for this application, is acceptable with the proposed EOP operator actions. Therefore, the applicable ATWS acceptance criteria (i.e., demonstrating core coolability is maintained) are satisfied during ATWS-I events for BSEP. The staff reviewed plant-specific information (e.g., EOPs), and specific aspects of the TRACG computer code applied in the context of the BSEP ATWS-I analysis provided by the licensee (e.g., updates to the quench model and revision to the T_{min} correlation in TRACG). The NRC also conducted confirmatory analyses using its TRACE methodology and the results showed additional conservatism in the licensee's analyses.

Overall, the NRC staff reviewed the LAR to confirm that:

- All L&Cs from the approved methodology topical reports have been addressed.

- The generic assessments are applicable to BSEP.

- The plant-specific assessments meet the regulatory criteria and, where calculations were necessary, the appropriate input assumptions and methods were used.

- Technical specification and license condition changes are appropriate and necessary to ensure safe operations in the expanded core flow region.

Based on the considerations noted above and the discussion contained in this SE, the NRC staff concludes that the proposed MELLLA+ amendments for BSEP are acceptable.

6.0 STATE CONSULTATION

In accordance with the Commission's regulations, the appropriate official for the State of North Carolina was notified of the NRC's proposed issuance of the amendments on July 26, 2018. The State official had no comments.

7.0 ENVIRONMENTAL CONSIDERATION

The amendments change a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and change surveillance requirements. 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, and there has been no public comment on such finding (82 FR 158, dated January 3, 2017). 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.

8.0 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.

9.0 REFERENCES

1. Duke Energy Letter to NRC dated September 6, 2016, "Request for License Amendment Regarding Core Flow Operating Range Expansion" (ADAMS Accession No. ML16257A418).

2. Duke Energy Letter to NRC dated November 9, 2016, "Brunswick, Units 1 and 2 - Response to Request for Supplemental Information for License Amendment Request Regarding Core Flow Operating Range Expansion" (ADAMS Accession No. ML16330A400).

3. NRC, Office of Nuclear Reactor Regulation, "Review Standard for Extended Power Uprates," RS-001, Revision 0, dated December 2003 (ADAMS Accession No. ML033640024).

4. NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, LWR Edition," March 2007 (ADAMS Accession No. ML070660036).

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

6. GE-Hitachi Nuclear Energy, Licensing Topical Report, NEDC-33173P-A, Revision 4, "Applicability of GE Methods to Expanded Operating Domains," dated November 2012 (ADAMS Accession No. ML123130130).

7. Duke Energy Letter to NRC dated February 5, 2018, "Brunswick Steam Electric Plant, Unit Nos. 1 and 2, Response to Request for Additional Information Regarding Core Flow Operating Range Expansion" (ADAMS Accession No. ML18037A660).

8. GE-Hitachi Nuclear Energy, Licensing Topical Report, NEDC-33075P-A, Revision 8, "General Electric Boiling Water Reactor Detect and Suppress Solution-Confirmation Density," dated November 2013 (ADAMS Accession No. ML13324A097).

9. GE-Hitachi Nuclear Energy, Licensing Topical Report, NEDC-33173, Supplement 3P-A, Revision 1, "Applicability of GE Methods to Expanded Operating Domains -Supplement for GNF2 Fuel," dated July 2011 (ADAMS Accession No. ML111960462).

10. Duke Energy Letter to NRC dated March 14, 2018, "Brunswick, Units 1 and 2, Additional Testing Information Relating to the Request for License Amendment Regarding Core Flow Operating Range Expansion" (ADAMS Accession No. ML18073A236).

11. Duke Energy Letter to NRC dated April 10, 2018, "Brunswick, Units 1 and 2, Supplement to Response to Request for Additional Information SRXB-RAI-2Regarding Request for License Amendment Regarding Core Flow Operating Range Expansion" (ADAMS Accession No. ML18100A230).

12. Siemens Power Corporation, Licensing Topical Report, EMF-2158(P)(A), Revision 0, "Siemens Power Corporation Methodology for Boiling Water Reactors: Evaluation and

Validation of CASMO-4/MICROBURN-B2," dated October 1999 (ADAMS Accession No. ML003698495).

13. AREVA NP, Licensing Topical Report, ANP-10307PA, Revision 0, "AREVA MCPR Safety Limit Methodology for Boiling Water Reactors," dated June 2011 (ADAMS Accession No. ML11259A021).

14. AREVA NP, Licensing Topical Report, ANP-10298PA, Revision 0, "ACE/ATRIUM-10XM Critical Power Correlation," dated March 2010 (ADAMS Accession No. ML101190042).

15. AREVA NP, Licensing Topical Report, EMF-2209PA, Revision 3, "SPCB Critical Power Correlation," dated September 2009 (ADAMS Accession No. ML093650229).

16. NRC Memorandum dated August 22, 2018, from Michael Case, Office of Nuclear Regulatory Research to Mirela Gavrilas, Office of Nuclear Reactor Regulation: "Transmittal of Preliminary, Interim Analysis of KATHY ATWS-I Temperature Test Data (NRR-2015-010)" (ADAMS Accession No. ML17223A716).

17. GE-Hitachi Nuclear Energy, Licensing Topical Report, NEDO-32465A, "Reactor Stability Detect and Suppress Solutions Licensing Basis Methodology for Reload Applications," dated August 1996 (ADAMS Accession No. ML14093A210).

18. GE-Hitachi Nuclear Energy, Licensing Topical Report, NEDE-33147, Revision 4 "DSS-CD TRACG Application" dated August 2013 (ADAMS Accession No. ML13224A319).

19. AREVA NP, Licensing Topical Report, ANF-913(P)(A), Revision 1 "COTRANSA2: A Computer Program for Boiling Water Reactor Transient Analyses," dated August 1990 (ADAMS Accession No. ML081340229) (Proprietary, withheld per 10 CFR 2.390).

20. GE-Hitachi Nuclear Energy, Licensing Topical Report, NEDC-24154P-A, Revision 1, "Qualification of the One-Dimensional Core Transient Model (ODYN) for Boiling Water Reactors (Supplement 1 -Volume 4)," dated February 2000 (ADAMS Accession No. ML062650284) (Proprietary, withheld per 10 CFR 2.390).

21. GE-Hitachi Nuclear Energy, NEDO-32047-A, "ATWS Rule Issues Relative to BWR Core Thermal-Hydraulic Stability," dated June 1995 (ADAMS Accession No. ML102230093) (Proprietary withheld per 10 CFR 2.390).

22. GE-Hitachi Nuclear Energy, NEDO-32164, "Mitigation of BWR Core Thermal-Hydraulic Instabilities in ATWS," dated December 1992 (ADAMS Accession No. ML102350204).

23. NRC letter to Exelon dated March 21, 2016, "Peach Bottom Atomic Power Station, Units 2 and 3 - Issuance of Amendments Re: Maximum Extended Load Line Limit Analysis Plus" (ADAMS Accession No. ML16034A372).

24. T. A. Bjornard and P. Griffith, "PWR Blowdown heat Transfer," in *Symposium on the Thermal and Hydraulic Aspects of Nuclear Reactor Safety, Vol. 1, ASME*, New York, 1977.

25. AREVA NP, Licensing Topical Report, BAW-10247PA, Revision 0 "Realistic Thermal-Mechanical Fuel Rod Methodology for Boiling Water Reactors," dated April 2008 (ADAMS Accession No. ML081340208).

26. Global Nuclear Fuel, Licensing Topical Report, NEDC-33256PA, NEDC-33257PA, and NEDC-33258PA, Revision 1, "The PRIME Model for Analysis of Fuel Rod Thermal-Mechanical Performance," dated September 2010 (ADAMS Accession No. ML102600259).

27. GE-Hitachi Nuclear Energy, Licensing Topical Report, NEDE-32906P, Supplement 3-A, Revision 1, "Migration to TRACG04 / PANAC11 from TRACG02 / PANAC10 for TRACG AOO and ATWS Overpressure Transients," dated April 2010 (ADAMS Accession No. ML110970401).

28. AREVA NP, Licensing Topical Report, 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," dated March 1983 (ADAMS Accession No. ML081850204) (Proprietary).

29. Duke Energy Letter to NRC dated March 1, 2018, "Brunswick Steam Electric Plant, Units 1 and 2 - Response to Request for Additional Information SNPB-RAI-2 Regarding Request for License Amendment Regarding Core Flow Operating Range Expansion" (ADAMS Accession No. ML18075A330).

30. Duke Energy Letter to NRC dated November 1, 2017, "Brunswick Steam Electric Plant, Units 1 and 2 - Response to Request for Additional Information Regarding Request for License Amendment Regarding Core Flow Operating Range Expansion" (ADAMS Accession No. ML17305A875).

31. Duke Energy Letter to NRC dated November 1, 2017, "Brunswick Steam Electric Plant, Units 1 and 2 – Update to Request for License Amendment Regarding Core Flow Operating Range Expansion" (ADAMS Accession No. ML17325B599).

32. GE Nuclear Energy, Topical Report, NEDE-24222, "Assessment of BWR Mitigation of ATWS, Volume II (NUREG-0460 Alternate No. 3)," dated December 1979 (ADAMS Accession No. ML102220441) (Proprietary, withheld per 10 CFR 2.390).

33. Duke Energy Letter to NRC dated April 6, 2017, "Brunswick Steam Electric Plant, Units 1 and 2 - Response to Request for Additional Information Regarding Request for License Amendment Regarding Core Flow Operating Range Expansion" (ADAMS Accession No. ML17096A482).

34. Duke Energy Letter to NRC dated February 14, 2018, "Withdrawal of Information Relating to Containment Accident Pressure Credit for the Request for License Amendment Regarding Core Flow Operating Range Expansion (TSC-2016-01)" (ADAMS Accession No. ML18045A859).

35. Duke Energy Letter to NRC dated March 29, 2018, "Brunswick Steam Electric Plant, Units 1 and 2 - Response to Request for Additional Information Regarding Request for License Amendment Regarding Core Flow Operating Range Expansion" (ADAMS Accession No. ML18090A004).

36. NUREG-1764 Rev. 1, "Guidance for the Review of Changes to Human Actions" September 30, 2017 (ADAMS Accession No. ML072640413).

37. NRC Memorandum dated April 25, 2018, from Chris Hoxie, Office of Nuclear Regulatory Research to Jennifer Whitman, Office of Nuclear Reactor Regulation: Delivery of Partial Deliverable Associated with Maximum Extended Load Line Limit Analysis Plus (MELLLA+) Anticipated Transient without SCRAM with Instability (ATWS-I) Part 1 for Brunswick Steam Electric Plant (BSEP), Work done to support User Need NRR-2016 009 (ADAMS Accession No. ML18116A497).

38. Safety Evaluation by the Office of Nuclear Reactor Regulation Related to Amendment No. 222 to Facility Operating License No. DPR-71 and Amendment No. 247 to Facility Operating License No. DPR-62 Carolina Power & Light Company Brunswick Steam Electric Plant, Units 1 and 2 Docket Nos. 50-325 and 50-324, (ADAMS Accession No. ML021440346).

39. Letter from NRC to Carolina Power and Light dated February 2, 2005, "Brunswick Steam Electric Plant, Units 1 and 2 - Issuance of Amendment Re: Elimination of Requirements For Hydrogen And Oxygen Monitors Using The Consolidated Line Item Improvement Process (TAC Nos. MC3866 and MC3867)" (ADAMS Accession No. ML050330233).

40. Letter from NRC to Carolina Power and Light dated April 13, 2009, "Brunswick Steam Electric Plant, Units 1 and 2 - Issuance of Amendments Regarding Adoption of TSTF 478, Revision 2, "BWR Technical Specification Changes that Implement the Revised Rule for Combustible Gas Control" (ADAMS Accession No. ML090550055).

41. GE Nuclear Energy, "Mark I Containment Program Load Definition Report," NEDO-21888, Revision 2, November 1981.

42. GE Nuclear Energy, "Mark I Containment Program Plant Unique Definition Brunswick Steam Electric Plant: Units 1 and 2," NEDE-24582, Revision 1, October 1981.

43. GE Nuclear Energy, "Mark I Containment Program Quarter Scale Plant Unique Tests," NEDE-21944-P, Volume 1, April 1979.

44. GE Company, "Mark I Containment Program, Full Scale Test Program Final Report, Task Number 5.11," NEDE-24539-P, April 1979 (ADAMS Accession No. ML081900313) (Proprietary).

45. General Electric Company, M3CPT, "The General Electric Mark III Pressure Suppression Containment Analytical Model," NEDO-20533, June 1974, and Supplement 1, September 1975.

46. Letter from NRC to AREVA, Inc. dated January 5, 2018, Final Safety Evaluation For Areva Inc. Topical Report ANP-10300P, Revision 0, "AURORA-B: An Evaluation Model For Boiling Water Reactors; Application To Transient And Accident Scenarios" (ADAMS Accession No. ML17346B115).

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Appendices:

A. Limitations from the Final Safety Evaluation for LTR NEDC-33173P, "Applicability of GE Methods to Expanded Operating Domains"
B. Limitations from the Final Safety Evaluation for LTR NEDC-33006, "General Electric Boiling Water Reactor Maximum Extended Load Line Limit Analysis Plus"
C. Limitations from the Final Safety Evaluation for LTR NEDC- 33075P Revision 7, "General Electric Boiling Water Reactor Detect and Suppress Solution – Confirmation Density"
D. Fuel Parameter Sensitivities
E. AREVA Codes Used for Brunswick MELLLA+ Application and Evaluation for MELLLA+ Applicability
F. Evaluation of RAI Responses
G. Acronyms and Initialisms

APPENDIX A

Limitations from the Final Safety Evaluation for LTR NEDC-33173P, "Applicability of GE Methods to Expanded Operating Domains"

The following is the NRC staff's evaluation to determine whether the limitations in the final safety evaluation NEDC-33173P (Reference 6) apply and are properly addressed.

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 stated that this limitation is applicable for GEH methods, and the licensee has used TGBLA06 and PANAC11 to develop the BSEP equilibrium core for all calculations involving GEH methods, including the MELLLA+ stability and ATWS evaluations. The NRC staff finds that this limitation is applicable to BSEP and is met.

The licensee stated that the limitation is not applicable to AREVA methods but that they use CASMO4/MICROBURN-B2, which is an approved method. The NRC staff reviewed the reference and determined that the limitation is applicable to BSEP. In Appendix E of this SE, the NRC staff concludes that the use of CASMO4/MICROBURN-B2 in the BSEP MELLLA+ domain is an acceptable extension of the existing approval and finds that this limitation is applicable to BSEP and is met.

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 stated that this limitation is not applicable, as BSEP uses the POWERPLEX core monitoring system based on the NRC-approved CASMO-4/MICROBURN-B2 methodology. Furthermore, the uncertainties associated with POWERPLEX CMS are used in the statistical analyses that are performed by AREVA for SLMCPR and linear heat generation rate (LHGR). The NRC staff concludes that this limitation is not applicable because the licensee used a different approved methodology from the GEH's TGBLA04/PANAC10 and the uncertainties are properly accounted for in the LAR.

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 licensee stated that this limitation is applicable. Only the low flow/high power point in the MELLLA+ domain (point M in Figure 3.1-1) exceeds 50 MWt/Mlbm/hr (with a value of 53.55 MWt/Mlbm/hr). However, the licensee stated that this point is not used for extended periods of operation, and that this limitation is not intended to place operation restrictions on the plant (Reference 6). Furthermore, the licensee stated that the requirement to provide power distribution assessment for plants exceeding 50 MWt/Mlbm/hr does not apply to AREVA methods, as the use of AREVA methodology at BSEP, including the applicability of power distribution uncertainties, is addressed in ANP-3108P (Enclosure 12, Reference 1). The NRC staff reviewed ANP-3108P and requested additional information that justified the AREVA methods at low flow/high point in the MELLLA+ domain of not including an additional SLMCPR penalty (described in Limitation 9.5 below) in SRXB-RAI-12.

In the RAI response, the licensee provided seven cycles of 2D TIP uncertainty data for BSEP, with average 2D uncertainty of approximately [[ ]] and the highest uncertainty being approximately [[ ]]. This is less than the [[ ]] uncertainty conservatively assumed for the SLMCPR calculation for BSEP, which is taken from EMF-2158P. The BSEP measured uncertainties are lower primarily because of the use of gamma TIPs in BSEP, which tend to have lower uncertainties and less sensitivity to void fraction. In ANP-3108P, TIP data were presented for other plants as well. None of these TIP data show a discernible trend in uncertainty with respect to power-to-flow ratio, core average void fraction, or power; however, the Brunswick data only extend as high as approximately 39 MWt-hr/Mlb, with the majority of data below 38 MWt-hr/Mlb. Although the data for the other plants included power-to-flow ratios up to 52 MWt-hr/Mlb, few data were obtained above 42 MWt-hr/Mlb. For BSEP, 42 MWt-hr/Mlb encompasses a large portion of the MELLLA+ domain, with 52 MWt-hr/Mlb being exceeded in only a small corner of the MELLLA+ domain, which will not typically be entered during normal cycle operation.

[[

]] Therefore, the NRC staff finds it reasonable to conclude that the bundle power distribution uncertainties will not increase sufficiently at the higher MELLLA+ power-to-flow ratios to make the power distribution uncertainties in EMF-2158P inapplicable.

In Reference 10, the licensee described testing that will be performed at BSEP prior to the first cycle of MELLLA+ operation, including collection of TIP data on each unit near 100% power and 85% core flow, and near 77.6% power and 55% core flow. If TIP uncertainties exceeding a value representing the upper bound of previous TIP uncertainties at BSEP (which support the uncertainties assumed in EMF-2158P), the licensee will enter the adverse condition in the Corrective Action Program and determine appropriate corrective actions.

[[

]] the low TIP uncertainties associated with the BSEP gamma TIP system (which was not credited in the BSEP SLMCPR analysis), and since the licensee plans to collect TIP data at low flow / high power points in the MELLLA+ domain.

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 stated that the limitation is not applicable, as the NRC did not impose an added SLMCPR on AREVA methods for EPU operation. The NRC staff finds that this limitation is not applicable because the staff did not impose a SLMCPR adder for AREVA methods for EPU operation.

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 licensee stated that the limitation is not applicable, as the NRC did not impose an added value on AREVA methods for EPU operation. The NRC staff reviewed the AREVA methods and finds that the 0.03 SLMCPR penalty from the M+ LTR SE Limitation and Condition 9.5 to be unnecessary for MELLLA+ operation at BSEP using AREVA methods due to the [[

]] the low TIP uncertainties associated with the BSEP gamma TIP system (which was not credited in the BSEP SLMCPR analysis), and since the licensee plans to collect TIP data at low flow / high power points in the MELLLA+ domain. Therefore, the NRC staff concludes that this limitation is not applicable.

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 stated that the limitation is applicable, and that the corresponding factors in AREVA methods (K-factors) are determined with their existing ACE/ATRIUM-10XM methodology documented in ANP-10298PA (Reference 14). The NRC staff reviewed this methodology and determined that the K-factors are calculated [[

]] Therefore, the NRC staff concludes that this limitation is applicable to BSEP and is met.

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 maximum average planar linear heat generation rate (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 stated that the limitation is applicable, and that its LOCA calculations include top-peaked and mid-peaked power shapes, as well as large and small break PCTs. However, the licensee stated that the AREVA LOCA methodology does not require a calculation of an upper bound PCT. Based on the review of the LOCA analysis and the AREVA methodology, the NRC staff concludes that this limitation is applicable to BSEP and is met.

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 5, 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 stated that the limitation is applicable. The NRC staff confirmed that calculations for the maximum and minimum core flow at rated EPU power and the transition statepoint have been performed for both power shapes. Therefore, the NRC staff concludes that this limitation is applicable to BSEP and is met.

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 UO2 and the limiting Gadolinium (Gd) O_2 rods.

The licensee stated that the limitation is applicable. The NRC staff concludes that compliance with the T-M acceptance criteria for AOOs has been demonstrated and documented using the most recent NRC-approved method in LTR BAW-10247PA (Reference 25), including the use of RODEX4. Therefore, the NRC staff concludes that this limitation is applicable to BSEP and is met.

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 stated the limitation is applicable, and that T-M calculations will be performed each cycle and reported in the cycle-specific RSAR. Compliance to transient T-M acceptance for the reference MELLLA+ cycle (Unit 1 Cycle 19) is documented in ANP-3280 Revision 1 (Enclosure 15, Reference 1). Therefore, the NRC staff concludes that this limitation is applicable to BSEP and is met.

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 stated that the limitation is not applicable because AREVA's approved T-M methodology does not have a void history bias. The NRC staff reviewed the AREVA methodology and finds this determination is acceptable for the reason stated.

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 stated that the limitation is applicable for GEH methods, and that the approved PRIME methodology (Reference 26) is used for all GEH analyses. The staff finds this limitation is application and met for GEH method because the licensee is used the PRIME methodology.

The licensee stated that the limitation is not applicable for AREVA methods, and that it is using the most current NRC-approved methods in LTR BAW-10247PA. The NRC staff reviewed the reference and determined that the methodology meets the underlying purpose of this limitation and conditions. Thus, the NRC staff finds this limitation met as to AREVA methods.

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 [thermal overpower] and MOP [mechanical overpower]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 stated that the limitation is not applicable, because it uses the most current NRC-approved T-M methods in LTR BAW-10247PA and neutronics methods LTR EMF-2158PA. The NRC staff reviewed the references and determined that these more current generic methods are appropriate for BSEP application. Therefore, the NRC staff concludes that this limitation is not applicable.

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 stated that the limitation is not applicable, and that the evaluation of the impact of pellet thermal conductivity degradation on AREVA methods is described in ANP-3108P (Enclosure 12, Reference 1), Appendix F. The NRC staff reviewed ANP-3108P, Appendix F, and concluded that the use of RODEX2 and RODEX4 in the BSEP MELLLA+ domain is an acceptable extension of the existing approval. Therefore, the NRC staff concludes that the licensee's determination on this limitation is 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 stated that the limitation is not applicable, and that related information for AREVA methods is provided in ANP-3108P (Enclosure 12, Reference 1), Appendix B. The NRC staff has reviewed ANP-3108P, Appendix B, and has concluded that the void reactivity coefficient bias and uncertainties used in AREVA methods are representative of the ATRIUM-10XM fuel loaded into the BSEP core for MELLLA+ operation. Therefore, the NRC staff concludes the licensee's determination on this limitation is 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 stated that the limitation is not applicable because AREVA has not identified any bias related to void history and has determined that the calculated void coefficient is accurate and provides the best possible information for the transient analysis, as documented in ANP-3108P (Enclosure 12, Reference 1). Specifically, AREVA methodology **[[** **]]** the reactivity coefficients used in the transient analysis, and the methodology provides conservative results that bound the reactivity coefficient uncertainties. The NRC staff reviewed ANP-3108P and concluded that MELLLA+ does not introduce any significant impact on the reactivity coefficients calculated by the AREVA methodology, and that the conservatism of the results has been demonstrated. Thus, the NRC staff concludes the licensee's determination on this limitation is acceptable.

Limitation 9.17, Steady-State 5 Percent Bypass Voiding

> The instrumentation specification design bases limit the presence of bypass voiding to 5 percent (LRPM (sic) 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.

The licensee stated that the limitation is applicable and that the required information regarding the cycle-specific bypass voiding calculations will be included in the cycle-specific RSAR for BSEP MELLLA+ operation. ANP-3280P Revision 1 (Enclosure 15, Reference 1) documented the bypass voiding calculation for the reference MELLLA+ cycle (Unit 1 Cycle 19), which demonstrated compliance with this limitation for the reference cycle. Therefore, the NRC staff concludes that this limitation is applicable to BSEP and is met for the reason stated.

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 stated that the limitation is not applicable, and that the issue of bypass voiding for BSEP under MELLLA+ conditions has been addressed in the SAR and found to be negligible [[ ]] at the D level LPRM. The NRC staff reviewed the information provided in the SAR and determined that the 5 percent and 2 percent penalties are not applicable to BSEP because bypass voiding has been evaluated and found to be negligible. Thus, the NRC staff concludes that the licensee's determination on this limitation is 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 Dix-Findlay 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 stated that the limitation is not applicable, and that the void-quality correlations used in AREVA methods for BSEP MELLLA+ (the [[ ]] and Ohkawa-Lahey void correlations) are addressed in ANP-3108P (Enclosure 12, Reference 1), Appendix B. The NRC staff reviewed the references and determined that the correlations are applicable to BSEP. Based on a review of the experimental data provided, the NRC staff concluded that the 0.01 OLMCPR penalty is not applicable to BSEP MELLLA+ because sufficient experimental data has been provided to validate the void fraction correlations for AREVA ATRIUM-10XM, including void fraction levels close to 100%. Therefore, the NRC staff finds that the licensee's determination on this limitation is 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 stated that this limitation is applicable for GEH methods. The NRC staff has reviewed the TRACG04/PANAC11 methodology used in this LAR and has found that the TRACG04/PANAC11 interfacial shear model complies with the NRC SE for NEDE-32906 Supplement 3-A (Reference 27), as required by this limitation. Thus, the NRC staff concludes that this limitation is met with respect to GEH methodology.

The licensee stated that this limitation is not applicable for AREVA methods. The NRC staff finds this determination acceptable because the void quality correlation 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 stated that this limitation is not applicable because BSEP is not a mixed vendor core and contains only AREVA fuel, and that this limitation relates to mixed cores with Global Nuclear Fuel (GNF) and non-GNF fuel. However, the NRC staff finds that the intent of this limitation is to generally address the use of GEH methods with non-GEH fuel, and therefore it is applicable for the GEH analyses performed for BSEP MELLLA+.

For BSEP MELLLA+, the licensee compared GBLA to MCNP to justify the application to ATRIUM-10XM lattices, using a similar approach as was previously endorsed by the NRC staff in NEDC-33173P, Supplement 3 (Reference 9) to expand the applicability of GEH methods to GNF2 fuel. However, the current approach used for ATRIUM-10XM in BSEP MELLLA+ was more limited in scope. In SRXB-RAI-11, the NRC staff requested additional details on the applicability of GE methods to ATRIUM-10XM fuel in BSEP MELLLA+. Based on additional information provided by the licensee in responses (as discussed in Appendix D herein), the NRC staff concludes that the licensee provided sufficient justification for the use of GE methods for ATRIUM-10XM fuel in BSEP MELLLA+ to satisfy the concerns raised in this limitation. Therefore, the NRC staff concludes that the licensee's determination on this limitation is 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
> - MOX fuel
>
> 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 stated that the limitation is not applicable because BSEP contains a full core of ATRIUM-10XM fuel, but that qualification of TGBLA06 for modeling ATRIUM-10XM fuel in the

BSEP equilibrium design using PANAC11 has been demonstrated to be acceptable based on MCNP benchmarking compared to the existing GNF 10x10 fuel products.

The NRC staff previously determined that the limitation is applicable to non-GE lattices, including ATRIUM-10XM. However, the NRC staff reviewed the benchmarking of TGBLA06 to MCNP for ATRIUM-10XM fuel in the BSEP equilibrium design and concludes, based on information in the LAR and in responses to SRXB-RAI-11, that sufficient qualification has been performed by the licensee to demonstrate that TGBLA06 is acceptable for the modeling of ATRIUM-10XM fuel in BSEP MELLLA+. Therefore, the NRC staff concludes that this limitation is met.

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 minimum critical power ratio importance parameter (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 licensee revised its response in Reference 2 and stated that it will evaluate and submit the requested information to the NRC after the first full operating MELLLA+ cycle for each unit using AREVA methods except for the MCPR Importance Parameter (MIP) criterion. The NRC staff previously determined that submittal of the MIP was not necessary in letter to GEH dated November 20, 2015 (ADAMS Accession No. ML15292A421). That NRC staff's determination was generic and is applicable to AREVA methods as well. Thus, the NRC concludes that the licensee's response to the request for supplement information is acceptable to meet this limitation.

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 stated that the limitation is applicable. The NRC staff reviewed the information provided in the BSEP SAR and has determined that all information has been provided as required in this limitation. Therefore, the NRC staff concludes that this limitation is met.

Conclusion:

Based on its review, the NRC staff finds the limitations in the final safety evaluation for NEDC-33173P (Reference 6) have been adequately addressed. These limitations did not result in changes to the BSEP TS for both units.

APPENDIX B

Limitations from the Final Safety Evaluation for LTR NEDC-33006, "General Electric Boiling Water Reactor Maximum Extended Load Line Limit Analysis Plus"

The following is the NRC staff's evaluation of the limitations in NEDC-33006P (Reference 5).

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 stated that this limitation is applicable. AREVA's CPR correlations have well-defined ranges of applicability that have been reviewed by the NRC staff, and include conservative actions to be applied in the event that these ranges are exceeded. The NRC staff reviewed the information and discussions provided on MELLLA+ for BSEP presented in ANP-3108P (Enclosure 12, Reference 1). The NRC staff finds that this limitation is applicable to BSEP MELLLA+ and is met.

Limitation 12.2, Related LTRs

> Plant-specific MELLLA+ applications must comply with the L&Cs 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 stated that this limitation is applicable. The licensee has reviewed the applicable limitations for AREVA and GEH methods and addressed them satisfactorily. The NRC staff reviewed the applicability of the MELLLA+ L&Cs and found the licensee adequately addressed this limitation. See Appendix A and C of this SE for the NRC staff's evaluation of the L&Cs for NEDC-33173P (Reference 6) and NEDC-33075P (Reference 8). The NRC staff notes that the limitations of NEDC-33147P, Revision 2 no longer need to be addressed because TRACG is now approved for DSS-CD stability solution calculations in NEDC-33075PA, Revision 8.

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 stated that this limitation is applicable. The LAR analyses comply with all operating condition changes that were implemented at BSEP in support of EPU and MELLLA+. The NRC staff finds that this limitation is applicable to BSEP MELLLA+ and is met.

Limitation 12.3b

> For all topics in LTR NEDC-33006P that are reduced in scope or generically resolutioned, the plant-specific application will provide justification that the reduced scope or generic resolution is applicable to the plant. If changes that invalidate the LTR resolutions 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 stated that this limitation is applicable. All fuel-related events that were included in the BSEP MELLLA+ application have been analyzed or resolutioned adequately for MELLLA+. The NRC staff finds that this limitation is applicable to BSEP MELLLA+ and is met.

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 stated that this limitation is applicable. Plant-specific calculations (including ATWS) have been performed using BSEP MELLLA+ conditions. The NRC staff finds that this limitation is applicable to BSEP MELLLA+ and is met.

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 stated that this limitation is applicable. Plant-specific calculations, including ATWS and ATWS-I, have been performed using BSEP MELLLA+ conditions and AREVA ATRIUM-10XM fuel. The NRC staff has reviewed the calculations provided in the SAR and has determined that the design features and performance of ATRIUM-10XM fuel have been adequately accounted for both GEH and AREVA methods. The NRC staff finds that this limitation is applicable to BSEP MELLLA+ and is met.

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 resolutioned 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 stated that this limitation is applicable. Plant-specific calculations have been performed using BSEP MELLLA+ conditions and AREVA ATRIUM-10XM fuel. The NRC staff reviewed the LAR and finds that this limitation is applicable to BSEP MELLLA+ and is met.

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 stated that this limitation is applicable. The licensee will use DSS-CD, which is an NRC approved stability method, NEDC-33075PA (Reference 8). Plant-specific calculations have been performed using BSEP MELLLA+ conditions and AREVA ATRIUM-10XM fuel. Therefore, the NRC staff finds that this limitation is applicable to BSEP MELLLA+ and is met.

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 stated that this limitation is applicable. BSEP MELLLA+ adopted the approved DSS-CD stability solution, including an automated backup stability solution. Therefore, the NRC staff finds that this limitation is applicable to BSEP MELLLA+ and is met.

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 BSEP SAR. Additionally, the SRLR for the initial MELLLA+ implementation cycle shall be submitted for NRC staff confirmation.

The licensee stated that this limitation is applicable and that a BSEP-specific RSAR (which is the AREVA equivalent to the SRLR) will be submitted for the initial MELLLA+ cycle for confirmation purpose. The NRC staff finds that this limitation is applicable to BSEP MELLLA+ and is met.

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 stated that this limitation is applicable. TSs were updated to support the BSEP MELLLA+ LAR and associated equipment out-of-service limitations, including a limitation prohibiting SLO in the MELLLA+ domain. Therefore, the NRC staff finds that this limitation is applicable to BSEP MELLLA+ and is met.

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 stated that this limitation is applicable. BSEP has the operational flexibility of final FW temperature reduction (FFWTR) and FWHOOS at MELLLA but not at MELLLA+. The licensee proposed a license condition to prevent FW temperature reduction in the MELLLA+ domain and performed evaluations to determine that this license condition is acceptable for BSEP MELLLA+. Therefore, the NRC staff finds that this limitation is applicable to BSEP MELLLA+ and is met.

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 stated that this limitation is applicable. The power-flow operating map has been provided and is reproduced in Figure 3.1-1 of this SE. Therefore, the NRC staff finds that this limitation is applicable to BSEP MELLLA+ and has been addressed adequately.

Limitation 12.6, SLMCPR Statepoints and CF Uncertainty

> Until such time when the SLMCPR methodology for off-rated SLMCPR calculation is approved by the NRC 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 stated that this limitation is applicable. SLMCPR values have been provided at each of the specified BSEP statepoint corners and have been calculated using the off-rated CF uncertainty where appropriate. Therefore, the NRC staff finds that this limitation is applicable to BSEP MELLLA+ and is met.

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 stated that this limitation is applicable. The DSS-CD solution provides an automated backup stability solution that fulfills this requirement. Based on its review of the LAR, the NRC staff finds that this limitation is applicable to BSEP MELLLA+ and is met.

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 stated that this limitation is applicable and that fluence calculations have previously been performed in Progress Energy Calculation Note 0B11-0012, Revision 1 and WCAP-17660, Revision 0 in accordance with NRC Regulatory Guide 1.190 for MELLLA+ conditions in BSEP. These calculations indicate a lower neutron flux under MELLLA+ than under pre-MELLLA+ conditions, and that BSEP MELLLA+ continues to meet the regulatory requirements for reactor vessel fracture toughness. Therefore, the NRC staff finds that this limitation is applicable to BSEP MELLLA+ and is met.

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 stated that this limitation is applicable. The NRC staff reviewed the information provided in the SAR on the BSEP Augmented Inservice Inspection examination program and determined that this program is adequate to ensures that any RCPB component degradation in other than Category "A" materials occurring during MELLLA+ operation is identified and addressed in an acceptable manner. Therefore, the NRC staff finds that this limitation is applicable to BSEP MELLLA+ and is met.

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 BSEP SAR will justify why the transition statepoint ECCS-LOCA response bounds the 55 percent CF statepoint. The BSEP 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 BSEP 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 stated that the limitation is applicable. **[[**

]] Therefore, the NRC staff finds that this limitation is applicable to BSEP MELLLA+ and is met.

Limitation 12.10b

> LOCA analysis is not performed on cycle-specific basis; therefore, the thermal limits applied in the BSEP 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 stated that this limitation is applicable. [[

]] Therefore, the NRC staff finds that this limitation is applicable to BSEP MELLLA+ and is met.

Limitation 12.10c

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

The licensee stated that this limitation is applicable. [[

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Therefore, the NRC staff finds this resolution 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 stated that this limitation is applicable. The offrated set down is applied using the flow-dependent LHGR multipliers (LHGRFACf multipliers), which are included in the core monitoring system. Therefore, the NRC staff finds that this limitation is applicable to BSEP MELLLA+ and is met.

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 stated that this limitation is applicable. Top- and mid-peaked power shapes were used in the plant-specific calculations reported in ANP-3105P, Revision 1 (Enclosure 24, Reference 1) as required by this limitation. Therefore, the NRC staff finds that this limitation is applicable to BSEP MELLLA+ and is met.

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 stated that the limitation is applicable, and that the AREVA methodology only calculates and reports Appendix K PCTs. The Appendix K calculations are reported in ANP-3105P, Revision 1 (Enclosure 24, Reference 1) using the approved AREVA uncertainty methodology. The NRC staff finds this approach an acceptable methodology to evaluate LOCA criteria. Therefore, the NRC staff finds that this limitation is applicable to BSEP MELLLA+ and is met.

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 stated that this limitation is applicable, and the AREVA methodology only calculates and reports Appendix K PCTs. The appendix K calculations are reported in ANP-3105P, Revision 1 (Enclosure 24, Reference 1) using the approved AREVA uncertainty methodology. The NRC staff finds this approach an acceptable methodology to evaluate LOCA criteria. Therefore, the NRC staff finds that this limitation is applicable to BSEP MELLLA+ and is met.

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.

[[

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Therefore, the NRC staff finds that this limitation is applicable to BSEP MELLLA+ and is met.

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 stated that this limitation is applicable. A large number of break sizes were evaluated at different flow rates and are reported in ANP-3105P, Revision 1 (Enclosure 24, Reference 1). Therefore, the NRC staff finds that this limitation is applicable to BSEP MELLLA+ and is met.

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 **[[ ]]** 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 stated that this limitation is applicable. Bypass boiling was evaluated for BSEP MELLLA+ 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 that this limitation is applicable to BSEP MELLLA+ and is met.

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 BSEP SAR shall provide a discussion of the analyses performed and the results.

The licensee did not state the applicability of this limitation to BSEP. The NRC reviewed the AREVA methodology to determine if the intent of this limitation was met. The licensee satisfied this limitation because AREVA methods do not use a generic RBM setpoint in the control rod withdrawal error analysis. Instead, the AREVA analyses are performed each cycle and will use the BSEP RBM setpoints. Therefore, the NRC staff finds that this limitation is met.

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 stated that this condition is applicable. The licensee confirmed that the RHR capability in BSEP is not affected by LOOP; therefore, the licensee provided MSIVC and PRFO calculations but not LOOP calculations, which satisfies this condition. Therefore, the NRC staff finds that this limitation is applicable to BSEP MELLLA+ and is met.

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 stated that this condition is not applicable because the licensee chose to increase the Boron-10 enrichment, resulting in an ODYN-calculated peak suppression pool temperature at MELLLA+ conditions in BSEP less than the reference OLTP/75% flow calculation. The NRC staff finds licensee's determination of this limitation is acceptable base on the ODYN calculated result for BSEP.

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 stated that this condition is applicable. The licensee has chosen to increase the Boron-10 enrichment, resulting in an ODYN-calculated peak suppression pool temperature at M+ conditions in BSEP less than the reference OLTP/75% flow calculation. Therefore, the NRC staff finds that this limitation is applicable to BSEP MELLLA+ and is met.

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 stated that the requirement to calculate PCT for the initial overpressure phase is applicable and that the ODYN calculation of PCT during the initial overpressure phase satisfies this limitation. Therefore, the NRC staff finds this limitation regarding initial overpressure is met. The licensee stated that the requirement to calculate PCT for the emergency depressurization phase is not applicable because TRACG calculations are not required since the licensee increased the Boron-10 enrichment. The NRC staff finds this part of the limitation not applicable because no TRACG calculations were required for emergency depressurization for the reason stated.

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., SLC system 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 stated that this condition is applicable. The NRC staff has reviewed the licensee's ODYN ATWS calculations and has determined that the input parameters, calculation assumptions, and equipment out of service conditions are reflective of the allowed BSEP plant configuration in MELLLA+ and that all important parameters are included in the analyses. Therefore, the NRC staff finds that this limitation is applicable to BSEP MELLLA+ and is met.

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 stated that this condition is applicable. The NRC staff reviewed the licensee's ODYN ATWS analyses and determined that the input parameters are conservative because these input parameters will produce a result that would challenge but not exceed the acceptance criteria for ATWS. Therefore, the NRC staff finds that this limitation is applicable to BSEP MELLLA+ and is met.

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 stated that this condition is applicable and has provided key input parameters in Table 9-3 in the BSEP SAR as well as a discussion of the parameter values and uncertainty treatment applied. Therefore, the NRC staff finds that this limitation is applicable to BSEP MELLLA+ and is met.

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 stated that this condition is applicable. The licensee has performed plant-specific ATWS-I analyses at the most limiting operating and modeling conditions, including peak reactivity exposure conditions and the regional-mode nodalization scheme. Additionally, the licensee used the latest NRC-approved codes: TGBLA06/PANAC11 and TRACG04. Therefore, the NRC staff finds that this limitation is applicable to BSEP MELLLA+ and is met.

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 stated that this limitation is not applicable. The NRC staff finds this determination is acceptable because no generic ATWS-I solution has been approved at this time. Plant-specific ATWS-I analyses were performed for BSEP MELLLA+ in accordance with L&C 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 stated that this limitation is applicable and that a plant-specific probabilistic risk assessment (PRA) has been performed for BSEP MELLLA+ (ERIN Report, "Brunswick MELLLA+ Risk Assessment," Revision 1, February 2015) including Core Damage Frequency (CDF) and Large Early Release Frequency (LERF). The NRC staff reviewed the results of this PRA and determined that MELLLA+ constitutes a Region III (very small risk change) increase in CDF and LERF relative to MELLLA operation, which is acceptable on a risk basis. Therefore, the NRC staff finds that this limitation is applicable to BSEP MELLLA+ and is met.

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 5x1020 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 stated that this limitation is applicable. Plant-specific fluence calculation results provided in Section 10.7 of the SAR indicate that the top guide, core plate, and shroud exceed the 5E20 n/cm^2 threshold for irradiated assisted stress corrosion cracking (IASCC). However, the current inspection strategies in place are considered sufficient to address IASCC of reactor internals. Therefore, the NRC staff finds that this limitation is applicable to BSEP MELLLA+ and is met.

12.23 Limitations from the ATWS RAI Evaluations

Limitation 12.23.1

> See limitation 12.18.d.

The licensee stated that this limitation is applicable. Therefore, the NRC staff finds that this limitation is applicable to BSEP MELLLA+ and is met for the reasons given under L&C 12.18.d.

Limitation 12.23.2

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

The licensee stated that the condition is applicable. The licensee has provided the key calculation parameters in Section 1.1.3, Section 9.3.1, and Table 9-3 of the BSEP SAR, which satisfies this condition. Therefore, the NRC staff finds that this limitation is applicable to BSEP MELLLA+ and is met.

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 stated that this condition is applicable. The licensee used SRV setpoints at the upper analytical limit setpoints, which is consistent with the BSEP-specific performance and reasonably accounts for uncertainty and valve drift during MELLLA+ operation at BSEP. Therefore, the NRC staff finds that this limitation is applicable to BSEP MELLLA+ and is met.

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 stated that this condition is applicable. The NRC staff has reviewed the EPG/SAG parameters and confirmed that they remain applicable for BSEP for M+. The NRC staff has also reviewed the ATWS analyses and determined that they account for all relevant EOPs, including water level control, SLC system injection, and RHR suppression pool cooling. Therefore, the NRC staff finds that this limitation is applicable to BSEP MELLLA+ and is met.

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 stated that this condition is applicable. For BSEP, the power/flow ratio at the minimum allowable core flow rate in the M+ domain at 120 percent OLTP is 44.7 MWt/Mlbm/hr, per Table 1-3 of the BSEP SAR. Therefore, the NRC staff finds that this limitation is applicable to BSEP MELLLA+ and is met.

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 NRC staff. Note: this limitation does not apply to special test assemblies.

The licensee stated that this condition is applicable. The licensee has performed bounding plant-specific ATWS-I analyses fully accounting for the design features and performance of ATRIUM-10XM fuel under MELLLA+ operation conditions for BSEP. Therefore, the NRC staff finds that this limitation is applicable to BSEP MELLLA+ and is met.

Limitation 12.23.7

> See limitation 12.23.6 (Limitation 12.23.6 and 12.23.7 are the same per the MELLLA+ LTR SE).

The licensee stated that this limitation is applicable. Therefore, the NRC staff finds that this limitation is applicable to BSEP MELLLA+ and is met for the reasons given under L&C 12.23.6.

Limitation 12.23.8

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

The licensee stated that this condition is applicable. The NRC staff has reviewed the ATWS analyses and verified that all plant-specific and fuel-design-specific features are accounted for. Therefore, the NRC staff finds that this limitation is applicable to BSEP MELLLA+ and is met.

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 stated that this condition is applicable. The licensee has provided detailed information on NPSH and crucial safety systems, and has used BSEP-appropriate assumptions for the ATWS analyses. Therefore, the NRC staff finds that this limitation is applicable to BSEP MELLLA+ and is met.

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 stated that this condition is applicable. To support MELLLA+ operation, BSEP increased the SLC system Boron-10 enrichment to meet the revised TS 3.1.7 requirement. This results in the decreased heat load to the containment relative to pre-MELLLA+ operation conditions, based on ODYN analyses. Thus, the licensee determined the operation of safety-grade equipment is not adversely affected under MELLLA+ operation. Because of the decreased heat load resulted from increased Boron-10 enrichment to implement MELLLA+, the NRC staff finds that this limitation is applicable to BSEP MELLLA+ and is met.

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 stated that this condition is applicable. The suppression pool temperature limit is unchanged relative to pre-MELLLA+ operation; additionally, the peak suppression pool temperature from the ATWS analyses for MELLLA+ (crediting the increased Boron enrichment included with the MELLLA+ extension) is lower than for pre-MELLLA+ conditions, so the original justification remains applicable. Based on the review of the information presented, the NRC staff finds that this limitation is applicable to BSEP MELLLA+ and is met.

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 stated that this condition is applicable. In-channel water rod flow was explicitly modeled in TRACG04 for the ATWS and ATWS-I analyses. Therefore, the NRC staff finds that this limitation is applicable to BSEP MELLLA+ and is met.

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 stated that this limitation is applicable. Exit void fraction conditions were provided in the SAR. Based on its review of the LAR, the NRC staff finds that this limitation is applicable to BSEP MELLLA+ and is met.

Limitation 12.24.3

See limitation 12.6. (Limitation 12.24.3 and 12.6 are the same per the MELLLA+ LTR SE).

The licensee stated that this limitation is applicable. Thus, the NRC staff finds that this limitation is applicable to BSEP MELLLA+ and is met for the reasons stated under L&C 12.6, above.

Limitation 12.24.4

See limitation 12.18.d. (Limitation 12.23.4 and 12.18.d are the same per the MELLLA+ LTR SE).

The licensee stated that this limitation is applicable. The NRC staff finds that this limitation is applicable to BSEP MELLLA+ and is met for the reason stated under Limitation 12.18.d, above.

Conclusion:

Based on its review , the NRC staff finds that all applicable limitations in the final safety evaluation for NEDC-33006P (Reference 5) have been adequately addressed by the licensee.

APPENDIX C

Limitations from the Final Safety Evaluation for LTR NEDC-33075P, Revision 7, "General Electric Boiling Water Reactor Detect and Suppress Solution – Confirmation Density"

The following is the NRC staff's evaluation of the limitations in LTR NEDC-33075P, Revision 7 (Reference 8):

Limitation 5.1

> The NRC staff previously reviewed and approved the implementation of DSS-CD using the approved GEH Option III hardware and software. The DSS-CD solution is not approved for use with non-GEH hardware. The hardware components required to implement DSS-CD are expected to be those currently used for the approved Option III. If the DSS-CD hardware implementation deviates from the approved Option III solution, a hardware review by the NRC staff will be required. Implementations on other Option III platforms will require plant-specific reviews.

The licensee stated that this limitation is applicable. BSEP currently uses GEH Option III hardware. Based on its review, the NRC staff finds that this limitation is applicable to BSEP MELLLA+ and is met.

Limitation 5.2

> The CDA setpoint calculation formula and the adjustable parameters values are defined in NEDC-33075P, Revision 7 (Reference 49). Deviation from the stated values or calculation formulas is not allowed without NRC review. To this end, the subject TR, when approved and implemented by a licensed nuclear power plant, must be referenced in the plant TSs, so that these values become controlled and part of the licensing bases.

The licensee stated that this limitation is applicable. To satisfy this limitation, the licensee included the DSS-CD LTR in the Administrative Controls section of the TS (Section 5.6.5, see discussion in Section 3.2.4 in this safety evaluation). Based on its review, the NRC staff finds that this limitation is applicable to BSEP MELLLA+ and is met.

Limitation 5.3

> The NRC staff previously concluded that the plant-specific settings for eight of the FIXED parameters and three of the ADJUSTABLE parameters, as stated in Section 3.6.3 of the NRC staff's SE for NEDC-33075P, Revision 5 (see Reference 52), are licensing basis values. The process by which these values will be controlled must be addressed by licensees.

The licensee stated that this limitation is applicable. The licensee will control these parameters via GEH settings report, thus this limitation is satisfied. Note that that licensee is updating the fixed T_{min} parameter to minimize spurious alarms and trips. For detailed discussion and the NRC staff's evaluation see Section 3.2.4 of this SE. Therefore, the NRC staff finds that this limitation is applicable to BSEP MELLLA+ and is met.

Limitation 5.4

> 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 stated that this limitation is not applicable to BSEP since this limitation only applies to plants other than BSEP. Based on its review, the NRC staff finds that this limitation is not applicable to BSEP.

Limitations from NEDC-33147P, "DSS-CD TRACG Application"

As discussed under the evaluation of NEDC-33006P L&C 12.2 (Appendix B of this SE), the limitations of NEDC-33147P, Revision 2 are no longer applicable because TRACG is now approved for DSS-CD stability solution calculations in NEDC-33075PA, Revision 8 (Reference 8). Therefore, the NRC staff finds that this limitation has been addressed.

Conclusion:

Based on its review, the NRC staff finds the limitations in the final safety evaluation for NEDC-33147P, Revision 2 and NEDC-33075PA, Revision 8 have been adequately addressed.

APPENDIX D

Fuel Parameter Sensitivities

This Appendix describes the results from using nominal fuel parameter values as well as the bounding fuel parameter sensitivities. The bounding fuel parameter sensitivities attempt to conservatively account for differences between ATRIUM-10 and ATRIUM-10XM for potential biases in inputs for ATRIUM-10XM or to conservatively account for features/parameters that were not specifically modeled for ATRIUM-10XM fuel (however, many parameters, such as geometric parameters, were specifically modeled for ATRIUM-10XM fuel).

Overview

The BSEP MELLLA+ application represents the first use of GEH methods with ATRIUM-10XM fuel, and was therefore a primary topic of focus for the NRC staff during the review. However, GEH has experience with modeling ATRIUM-10 fuel transitions at four plants (LaSalle, Columbia, River Bend, and Grand Gulf). The approach used for ATRIUM-10XM fuel in BSEP was essentially an extension of the modeling bases used for ATRIUM-10 fuel, as discussed in the following paragraphs.

The ATRIUM-10XM fuel geometry and materials were explicitly modeled by GEH based on inputs from AREVA. Documentation on these inputs was reviewed by the NRC staff. To limit the ATRIUM-10XM fuel performance information that needed to be passed from AREVA to GEH, the ATRIUM 10 calculation bases (used by GEH for previous applications) were applied to ATRIUM-10XM fuel by increasing the applicable uncertainties and obtaining confirmation from AREVA that the applied uncertainty ranges are appropriate for ATRIUM-10XM fuel relative to ATRIUM 10 fuel.

The licensee performed a product requirements review to down-select a set of fuel parameters that were most the important to the ATWS and ATWS-I analyses, among the parameters that were not specifically modeled for ATRIUM-10XM fuel. For the ODYN ATWS analyses, these parameters were (with sensitivity ranges noted in brackets):

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- [[ ]]
- [[ ]]

For the TRACG ATWS-I analyses, these parameters were:

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- [[ ]]
- [[ ]]
- [[ ]]
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The justification for each of these parameter ranges is given in the following sections.

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Conclusions for ATWS-I Fuel Parameter Sensitivity Results

In the response to SRXB-RAI-6, the licensee provided results for additional TTWBP ATWS-I fuel parameter sensitivity calculations by varying each of the five fuel parameters individually, to the minimum and maximum values based on the stated sensitivity ranges (for a total of 10 individual sensitivity cases). The results of these analyses are summarized in Table SRXB-6-1 of the response to SRXB-RAI-6.

The licensee selected the bounding fuel parameter values (i.e., minimum, maximum, or nominal value for each parameter) based on which value gave the largest PCT in the individual sensitivity studies. These bounding values were then combined to give an overall "bounding" fuel parameter sensitivity result in which all five fuel parameters were varied in their individually most bounding direction.

The NRC staff notes that it is possible that the most limiting values could occur at some value between the minimum and maximum value for each fuel parameter sensitivity range. For example, previous staff analyses indicate that oscillations may be most unstable (i.e., highest decay ratio) at a particular value of gap conductance, with a lower decay ratio occurring for either higher or lower gap conductance values, and this local maximum may or may not occur within the range of gap conductances examined in the current study. Furthermore, the NRC staff believes that competing effects or interactions between fuel parameter values are possible, in which the overall most limiting set of fuel parameter sensitivity values (in terms of PCT) might not necessarily correspond to the set of fuel parameter sensitivity values chosen based on the individual sensitivity studies. For example, increasing the gap conductance and increasing the direct energy deposition fraction would both reduce the effective thermal time constant of the fuel; this is one of numerous interactions that may occur between the selected fuel parameter sensitivities.

However, the NRC staff finds that the licensee applied substantial conservatism in the selection of the sensitivity ranges for at least several of the fuel parameters, and that combining these conservative values for all five parameters together adds a substantial degree of additional conservatism. This conservatism is likely to envelope, by a significant margin, any possible "second-order" interactions or local maximum effects as described above. Therefore, the NRC staff concludes that the licensee's approach for applying the bounding fuel parameter sensitivities for ATRIUM-10XM fuel provides a reasonable and sufficient degree of conservatism for the purposes of the ATWS and ATWS-I analyses in BSEP for MELLLA+ and is, therefore, acceptable.

Consideration of DSS-CD Fuel Parameter Values

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Conclusion:

Based on its review, the NRC staff finds the fuel parameter sensitivity studies sufficiently bound the ATRIUM-10XM fuel type.

APPENDIX E

AREVA Codes Used for Brunswick MELLLA+ Application and Evaluation for MELLLA+ Applicability

This appendix provides a summary report on Framatome licensing methods and Framatome topical reports used for extended power/flow operating (EPFO) domain analysis at Brunswick Units. The NRC staff review of Framatome methods identified that there are no safety evaluation restrictions on power or flow for the AREVA topical reports. The review also indicated that there are no SE restrictions on the parameters most impacted by the increased power level at each core flow rate in the MELLLA+ domain: steam flow, FW flow, jet pump M-ratio, and core average void fraction.

The applicability determination of the Framatome methods to Brunswick EPFO domain analysis included an evaluation of the core and reactor conditions experienced under EPFO domain conditions to determine any challenges to the validity of the models. When the reactor power is increased and/or the core flow is decreased, the resultant impact on operating margin is mitigated to a large extent by a decrease in limiting assembly radial power factor that is necessary since the operating limits such as MCPR, MAPLHGR and LHGR are dependent on the limiting assembly power but are fairly insensitive to the core thermal power. Due to this, the following observations are made about the EPFO domain operating conditions:

- The reduction in the hot assembly radial peaking factor leads to a more uniform radial power distribution and consequently a more uniform core flow distribution. The net result being less flow starvation of the hottest assemblies.
- With the flatter radial power distribution, more assemblies and fuel rods are near thermal limits.
- There will be higher steam flow and FW flow rates for a given core flow at core flows previously constrained by the MELLLA operating boundary.
- With the increase in the average assembly power for a given core flow the core pressure drop will increase slightly resulting in a decrease in the jet pump M-ratio for a given core flow rate.
- Core average void fraction will increase.

Following is a list of approved codes that were used in the Brunswick EPFO domain analysis:

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 NRC 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)) frequency domain stability code, used for exclusion region calculations and thermal-hydraulic stability analysis.

7. 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.

8. 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.

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

10. 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.

CASMO4/MICROBURN-B2

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. The approving SE 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 SE 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 reviewed the applicable limitations and finds that operation in the EPFO domain regime does not invalidate any of the limitations. As discussed earlier in Section 3.2.3 of this SE, the NRC staff concludes that EPFO domain operating conditions at Brunswick are bounded in terms of void fraction, power, and flow by other BWRs where the use of CASMO4/MICROBURN-B2 is currently approved and the simulation result demonstrated good benchmarks against plant data.

SAFLIM3D

SAFLIM3D is the code used by Framatome 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. Brunswick operation in the EPFO domain does not impact the Monte Carlo process; therefore, the NRC staff concludes that the use of SAFLIM3D in the Brunswick EPFO 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 NRC 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 C [Exxon Nuclear Company, Inc.] 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 critical heat flux (CHF) correlation has been supplemented with the NRC-approved SPCB and ACE CHF correlations.

As discussed in Section 3.2 in this SE, the NRC staff concludes the EPFO domain operating conditions for Brunswick 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 Brunswick EPFO domain is an acceptable extension of the existing approval.

XCOBRA-T

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

The NRC SE for XN-NF-84-105(P)(A), *XCOBRA-T: A Computer Code for BWR Transient Thermal-Hydraulic Core Analysis, Exxon Nuclear Company, February 1987* (Accession No. ML081340188), 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 (note, the approval was subsequently expanded in letter dated May 31, 2000, ADAMS Accession No. ML003719373):

    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 Section 3.2 in this SE, the NRC staff concludes the EPFO domain operating conditions in Brunswick 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-3108P. XCOBRA-T uses the Ohkawa-Lahey correlation, **[[**

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Based on the review described above, the NRC staff finds that operation in EPFO domain 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 EPFO domain is not likely to be impacted by reverse bypass flow.

The NRC staff also finds that operation in the EPFO domain increases the core average void fraction, but Brunswick EPFO domain conditions are bounded by the operating experience of other plants that use approved AREVA methods successfully. In addition, AREVA has demonstrated **[[** **]]**

Therefore, the NRC staff concludes that the use of XCOBRA-T in the Brunswick MELLLA+ EPFO domain is an acceptable extension of the existing approval.

<u>COTRANSA2</u>

COTRANSA2 is the transient coupled neutronic T-H'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 SE for ANF-913PA 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 AOO analyses and ATWS overpressure.

The COTRANSA2 SE (Reference 19) 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 EPFO domain operating conditions in Brunswick. And none of the limitations in the original COTRANSA2 SE are violated by Brunswick operation in the EPFO domain.

Therefore, the NRC staff concludes that the use of COTRANSA2 in the Brunswick EPFO domain is an acceptable extension of the existing approval.

<u>STAIF</u>

STAIF is the frequency domain BWR thermal-hydraulic stability code, including reactivity feedback effects. The SE for the topical report (EMF-CC-074(P)(A)) frequency domain stability code 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 T-H 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 decay ratio conditions.

The limitations are implemented in the code itself when it collects data from MICROBURN-B2 and collapses it for calculation.

Therefore, the NRC staff concludes that the use of STAIF in the Brunswick EPFO 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.

The SE for the topical report (XN-NF-81-58(P)(A)) 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) justifies Gd fuel properties for up to 8 wt percent Gd with RODEX2 methods, which covers the expected operation of Brunswick in MELLLA+ EPFO domain.

The NRC staff finds operation in the EPFO 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 EPFO domain is not expected to impact the validity of the RODEX2 models. Based on this finding, the NRC staff concludes that the use of RODEX2 in the Brunswick EPFO 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.

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 SE for the topical report (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 LTR 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.

The NRC staff has reviewed these limitations and concludes that they will be satisfied in the EPFO domain in Brunswick without changes.

In the approved RODEX4 Supplement 1 (BAW-10247, Revision 0, Supplement 1PA, Revision 0), AREVA/Framatome has a corrosion model where the uniform oxidation rate is a two-stage model that is a function of both exposure, a corrosion enhancement factor (depends on reactor chemistry) and temperature at the metal-oxide interface, and therefore, linear heat generation rate (LHGR). It is recognized that both nodular corrosion and diffusion-controlled uniform corrosion occur on BWR cladding. Nodular corrosion is treated as athermal, and the diffusion-controlled corrosion is temperature-driven. RODEX4 does not have a nodular corrosion model. The uniform two-stage corrosion model includes a pre-transition model and a post transition model with the transition temperature a function of the metal-oxide interface temperature.

The approved RODEX4 Supplement 1 (BAW-10247PA, Revision 0, Supplement 1P, Revision 0) consists of a new hydrogen pickup model that uses a **[[**
]]. In the new model, the hydrogen pickup fraction is determined to be a function of
[[
]] The NRC staff concluded that the new hydrogen pickup model in RODEX4 Supplement 1 is acceptable for the CSWR or RX Zircaloy-2 cladding and may be used for analyses where hydrogen content is required.

The NRC staff finds that, as indicated by the RODEX2 evaluation above, operation in the EPFO 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 EPFO domain is not expected to impact the validity of the RODEX4 (including Supplement 1) models.

Therefore, the NRC staff concludes that the use of RODEX4 in the Brunswick EPFO 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 SE for the topical report (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 EPFO domain operation. Thus, the NRC staff concludes that the use of RELAX in the Brunswick EPFO 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 SE 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 EPF operation. In addition, the NRC staff has reviewed the SE limitations and operation in the EPFO domain does not affect them. Thus, the NRC staff concludes that the use of HUXY in the Brunswick EPFO domain is an acceptable extension of the existing approval.

Conclusion:

Based on its review, the NRC staff finds the methodologies in the LAR are acceptable for use in the MELLLA+ operating domain.

APPENDIX F

Evaluation of Request for Additional Information Responses

SRXB-RAI-1

Please provide the BSEP-specific noise data used to justify the use of a Detect and Suppress Solution Confirmation Density (DSS-CD) S_{AD} value of **[[** **]]** to ensure that power oscillations can be readily detected and suppressed, which, in part, ensures that fuel design limits are not exceeded.

<u>Evaluation</u>:

In the RAI response, the licensee provided a discussion of the statepoints at which noise data was collected for each unit, as well as the results of their analysis that revealed **[[**

]]

[[

]] This RAI is closed.

SRXB-RAI-2

Please provide a basis for the use of a DSS-CD time period lower limit (T_{min}) of **[[** **]]**, which is higher than the approved value of **[[** **]]** given in the DSS-CD LTR and reduces the range of oscillations indicative of an anticipated reactor instability. This basis will ensure that power oscillations can be readily detected and suppressed which, in part, ensures that fuel design limits are not exceeded. Please include the following components:

a. Provide justification, based on plant data, that indicates that the **[[** **]]** value may lead to an unacceptable or undesired likelihood of spurious scram in BSEP during Maximum Extended Load Line Limit Analysis Plus (MELLLA+) operation; and

b. Demonstrate that T-H instabilities are not expected to occur below a T_{min} of **[[** **]]**. As part of this demonstration, please include analysis based on the TRACG DSS-CD analyses that were provided in the Safety Analysis Report for Brunswick Steam Electric Plant Units 1 and 2 Maximum Extended Load Line Limit Analysis Plus (hereafter the SAR), as well as a justification that the T-H oscillation period would remain above **[[** **]]** in both units, at any cycle exposure, and at any other point in the DSS-CD armed region.

<u>Evaluation</u>:

In the RAI response, the licensee provided plant data **[[**

]]

The licensee also presented the calculated oscillation period values obtained with the DSS-CD PBA for the TRACG DSS-CD confirmatory analyses provided in the BSEP MELLLA+ SAR. [[

]]

The DSS-CD LTS is only capable of providing licensing basis SLMCPR protection if the oscillation period associated with any anticipated TH instability during plant operation remains above T_{min} and below T_{max} (the DSS-CD time period upper limit). [[

]]

[[
]] the NRC staff finds that a T_{min} of 1.0 sec follows the intent of the DSS-CD LTR [[
]]

Additionally, the NRC staff previously approved the position that, as stated in Section 2.6.2.3 of the Interim Methods LTR (Reference 6), "the existing GE thermal-hydraulic stability models reasonably and adequately model the magnitude and period of industry thermal-hydraulic instability events." The TH oscillation period is strongly tied to the coolant transit time through the BWR channel, which the NRC staff has found to be adequately represented in TRACG and which supports the NRC staff's conclusion that TRACG can reliably predict the oscillation period [[ ]]

Based on these considerations, the NRC staff concludes that a T_{min} of 1.0 sec [[ ]] provides adequate safety protection against anticipated TH oscillations in BSEP MELLLA+ and is an acceptable extension of the DSS-CD methodology for the plant-specific BSEP MELLLA+ application. This RAI is closed.

SRXB-RAI-3

In the SAR, Anticipated Transient without Scram (ATWS) and ATWS Instability (ATWS-I) results were presented using the limiting fuel parameter sensitivity values to ensure that the ATRIUM-10XM fuel in BSEP was adequately modeled using GEH methods. Please justify the acceptability of **[[**

]], for the DSS-CD confirmatory analyses. This will ensure that power oscillations can be readily detected and suppressed which, in part, ensures that fuel design limits are not exceeded for the ATRIUM-10XM fuel. Specifically, justify that the **[[**

]] used in the DSS-CD approach conservatively bounds the uncertainty associated with ATRIUM-10XM fuel when the limiting fuel parameter values are considered.

Evaluation:

In the RAI response, the licensee gave the justification that the **[[**

]]

[[

]] This RAI is closed.

SRXB-RAI-4

The following additional information is requested for the turbine trip with bypass (TTWBP) ATWS-I analyses to ensure that the transient is adequately modeled such that the operator actions credited are appropriate for the event:

a. Please describe the approach used to ensure that the maximum steady-state Linear Heat Generation Rate (LHGR) for the TRACG TTWBP analyses was **[[ ]]** of the Maximum LHGR limit. Were the steady state conditions (e.g. control rod positions) used to initiate the TRACG TTWBP analyses consistent with the equilibrium cycle conditions provided by AREVA?

b. What value of 'dtmax' (maximum allowed timestep size) was used in the TRACG TTWBP analyses presented in the SAR? Was this 'dtmax' value conservative for ATWS-I analyses? If not, please provide revised TRACG TTWBP analyses with an appropriate value for 'dtmax' to replace the analyses presented in the SAR.

Evaluation:

In the RAI response, the licensee referred to updated BSEP TRACG ATWS-I analyses in a letter to the NRC dated November 1, 2017, in which the TRACG parameter 'dtmax' was updated to an appropriate value of **[[ ]]** per the GEH ATWS-I process. This value reduces an unintended non-conservatism associated with the 'dtmax' value of **[[ ]]** used in the original results in the SAR.

Additionally, the licensee indicated that the updated analysis used the core conditions and control rod pattern consistent with the equilibrium cycle conditions provided by AREVA. The maximum steady-state LHGR of **[[ ]]** of the LHGR limit was obtained by increasing the hot rod relative power (i.e., hot rod peaking factor) in all assemblies until the maximum LHGR throughout the core was [[ ]] of the limit. The NRC staff finds this approach acceptable because it ensures the core conditions remain as similar to the AREVA-specified conditions as possible while introducing added conservatism by increasing the hot rod power peaking factor. This RAI is closed.

SRXB-RAI-5

The following additional information is requested to ensure that the transient is adequately modeled such that the operator actions credited are appropriate for the ATWS-I TTWBP event:

a. Please provide a diagram of the TRACG channel grouping used for the regional and core-wide mode TTWBP ATWS-I analyses. Did the regional mode analyses result in higher PCT values than the core-wide analyses?

b. Figure 9-10 of the SAR provides total core values for neutron flux and inlet flow rate versus time; however, this does not sufficiently describe the local assembly behavior during regional mode oscillations. Please provide additional time-dependent results at symmetric core locations (including the limiting Peak Cladding Temperature (PCT) assembly) showing the amplitude of the regional oscillations for the revised TTWBP analyses requested in SRXB-RAI-4. Please include time-dependent assembly power, assembly inlet flow rate, and maximum cladding temperature for each of these two assemblies, as well as the axial location where the maximum PCT occurs.

Evaluation:

In the RAI response, the licensee provided the requested information on TRACG TTWBP channel grouping for the core wide and regional oscillation modes and confirmed that the regional mode analyses were more limiting than the core wide mode analyses in terms of maximum PCT. Therefore, the licensee's presentation of regional mode (and not core wide mode) TTWBP results in the SAR is justified.

The licensee also provided the requested results for the limiting channel TH and neutronic behavior, along with corresponding results for the radially symmetric channel in the core, during the TTWBP event. This information allowed the NRC staff to understand the behavior of the core during the regional oscillations more fully than examination of the full-core results alone. The NRC staff reviewed this additional information and found that the specified channels exhibit qualitatively the expected flow, power, and cladding temperature behavior based on the NRC staff's experience with stability events. Thus, the NRC staff's concerns have been addressed, and this RAI is closed.

SRXB-RAI-6

Please provide the following information regarding the limiting fuel parameter sensitivities to ensure that the transient is adequately modeled such that the operator actions credited are appropriate for the ATWS-I TTWBP event:

a. Please provide the process used to determine the parameters used for the ATRIUM-10XM fuel parameter sensitivity study for the ATWS and ATWS-I analyses for BSEP MELLLA+. For each parameter, please provide information on how the sensitivity range was determined.

b. Please provide a table showing the maximum PCT value for the TRACG TTWBP analyses (including the revisions described in SRXB-RAI-4, if necessary) obtained by adjusting each fuel parameter individually to the minimum and maximum value within the appropriate sensitivity range to demonstrate each parameter's impact on the results.

Evaluation:

In the RAI response, the licensee described the process for determining the limiting fuel parameters and their bounding sensitivity ranges for the ATWS and ATWS-I analyses for BSEP MELLLA+. A detailed discussion and staff evaluation of this process is provided in Appendix D.

Based on this evaluation, the NRC staff concludes that the limiting fuel parameters and their bounding sensitivity ranges adequately and conservatively bound the performance of ATRIUM-10XM fuel relative to ATRIUM-10 fuel for ATWS and ATWS-I in BSEP MELLLA+. This RAI is closed.

SRXB-RAI-7

Based on recent NRC funded ATWS-I test experiments (KATHY), the failure to rewet (FTR) temperature is in reasonable agreement with homogeneous nucleation plus contact temperature and provides a reasonable representation of the cladding temperature behavior during ATWS-I oscillations. The homogeneous nucleation plus contact temperature is more conservative compared to Modified Shumway T_{min} model used in the SAR. Therefore, the following sensitivity studies are requested to ensure that the transient is adequately modeled such that the operator actions credited are appropriate for the event

a. Please provide TRACG sensitivity studies for the TTWBP event in BSEP using the homogeneous nucleation temperature plus contact temperature model for T_{min}. Sensitivity studies may include realistic assumptions for input parameters such as FW temperature versus time, operator action time to reduce water level, maximum initial LHGR value, and use of the TRACG quench model. Please also provide sensitivity studies using the limiting fuel parameter sensitivity values in conjunction with the homogeneous nucleation T_{min} model.

b. If the limiting fuel parameter sensitivities in any of the cases performed in (1) led to a larger increase in maximum PCT than in the cases provided in the SAR, please explain the reason for this larger increase. This explanation may include consideration of the location of maximum PCT, the average or peak LHGR at this location, or other considerations as appropriate.

Evaluation:

In the RAI response, the licensee provided the requested ATWS-I sensitivity results. To ensure that the 2200°F PCT criterion was not exceeded when using nominal fuel parameter values, the licensee included sensitivity results using the "best-estimate" FW temperature reduction rate of 0.5°F/sec, which was justified in the response to SRXB-RAI-8, as opposed to 1.3°F/sec assumed in the nominal ATWS-I case. Using the 0.5°F/sec FW temperature reduction rate along with the homogeneous nucleation plus contact temperature (HN+CT) T_{min} model, the maximum PCT was calculated to be **[[ ]]**. Failure-to-rewet occurred in the simulation, but the slower decrease in FW temperature meant that the operator actions at 120 sec were successful in preventing the PCT from exceeding 2200°F. An additional sensitivity case with an operator action time of 96 sec prevented TH oscillations altogether, with a PCT of **[[ ]]** However, a 120 sec action time remains the licensing basis for Brunswick under MELLLA+, per the BSEP MELLLA+ SAR.

The NRC staff examined an additional sensitivity case that used the 0.5°F/sec FW temperature reduction rate, the HN+CT T_{min} model, and the limiting fuel parameter sensitivity values. In this case, the increased oscillation growth rate meant that a cladding temperature of 2200°F was exceeded in at least one rod in each of 18 fuel bundles. Based on the evaluation in Appendix D of this SE, the NRC staff concludes that the limiting fuel parameter study constitutes a highly conservative representation of ATRIUM-10XM fuel performance for ATWS-I, relative to the nominal fuel parameter values. Furthermore, consistent with previous staff evaluations for NEDO-32047A (Reference 21), the NRC staff concludes that core coolability can be reasonably expected to be maintained given the relatively small number of fuel rods predicted to exceed 2200°F even under conservative fuel parameter assumptions. Therefore, the NRC staff concludes that the ATWS acceptance criteria are satisfied for BSEP MELLLA+ operation with ATRIUM-10XM fuel. This RAI is closed.

SRXB-RAI-8

Please justify the FW temperature reduction rate that was used in the ATWS-I TTWBP analyses presented in the SAR, as well any reduced rate used in the ATWS-I sensitivity studies to ensure that the transient is adequately modeled such that the operator actions credited are appropriate for the ATWS-I TTWBP event. Justify this using appropriate plant data or simulator results, if available. If plant simulator data is provided, please justify the adequacy of the simulator for determining the FW temperature reduction rate.

<u>Evaluation</u>:

In the RAI response, the licensee presented plant data for four turbine trip events at BSEP in which the majority if not all of the coolant inventory following the trip was supplied by the FW system. The measured FW temperature in each event exceeded the FW temperature that would occur assuming a constant 0.5° F/sec rate of decrease initiated at the time of the trip or first significant turbine load decrease.

In these events, after plant scram, the FW demand after the trip was low. By contrast, during a high power ATWS, the FW demand remains relatively high as the operators follow the EOP reactor water level strategy. This means that the inventory of heated water in the FW piping after the trip will enter the vessel sooner, leading to a more rapid decrease in FW temperature for the ATWS. To account for this, in Figure SRXB-8-5 of the RAI response, the licensee presented the same measured FW temperature data versus the integrated FW mass that has entered the RPV since the turbine trip. In each case, the FW temperature versus integrated FW mass is conservatively bounded by the case of a 0.5° F/sec FW temperature reduction rate.

After the turbine trip, essentially no heat is added to or lost from the FW inventory between the condenser and RPV. Therefore, the NRC staff finds that the results in Figure SRXB-8-5 adequately represent the expected FW temperature response during an ATWS, based on consideration of measured plant data and fundamental principles of heat transfer. For this reason, the NRC staff finds the use of a 0.5° F/sec FW temperature reduction rate acceptable for the ATWS-I TTWBP analyses for BSEP MELLLA+. This RAI is closed.

SRXB-RAI-9

To ensure that the event is adequately modeled such that the operator actions credited are appropriate for the ATWS-I TTWBP event, the NRC staff has the following questions regarding the use of the GEXL correlation for ATWS-I analyses:

a. Please discuss the GEXL correlation's applicability to oscillatory conditions.

b. For ATWS-I oscillations such as those calculated in the TTWBP analyses for BSEP MELLLA+, is the critical heat flux temperature (in the oscillation growth phase and in the limit cycle phase) during a given oscillation period typically determined by **[[** **]]**?

Evaluation:

In the RAI response, the licensee discusses that oscillatory test data has been used to assess the onset of boiling and return to nucleate boiling predictions for the GEXL correlations for GEH/GNF fuel products since GE9. The NRC staff finds that the use of GEXL to determine the onset of boiling transition during ATWS-I is consistent with its use for this purpose in the DSS-CD application. During ATWS-I, calculation of boiling transition as well as heat transfer coefficients is determined with a combination of correlations including GEXL, Zuber, and Biasi depending on factors such as flow rate. Thus, the NRC staff considers this RAI closed.

SRXB-RAI-10

To ensure that the event is adequately modeled such that the operator actions credited are appropriate for the ATWS-I TTWBP event, the NRC staff requests the following additional information regarding the use of R-factors in the TRACG ATWS-I analyses:

a. Please provide the R-factors used in each assembly in the TRACG TTWBP models, and justify how these R-factors were determined. If applicable, please provide TTWBP sensitivity results that indicate the effect that changing the GEXL critical power value has on maximum PCT during this event.

b. The stated range of validity of R-factors in the GEXL97 correlation for ATRIUM-10 fuel is **[[** **]]**. If an R-factor less than **[[** **]]** was used in the ATWS-I analyses, please justify the use of these ATRIUM-10XM R-factors outside of the stated range of validity. Does the GEXL correlation behave properly for R < **[[** **]]**? (For example, is the **[[** **]]** term intended to be used for R < **[[** **]]**?)

c. For safety analyses such as AOOs, higher hot-rod R-factor values are typically more limiting. If a lower R-factor value was found to be more limiting in Part (a), please explain why it was more limiting or please provide additional sensitivity analyses using the following assumptions for assembly R-factors, to better understand the limiting combination of R-factors for the TTWBP event:

1. Hot channel R-factor taken from bounding high value from AREVA inputs, and remaining R-factors taken as the low value
2. Hot channel R-factor taken as the low value, and remaining R-factors taken as the bounding high value
3. All R-factors taken as the low value
4. All R-factors taken as the bounding high value

Evaluation:

In the RAI response, the licensee indicated that the ATWS-I TTWBP sensitivity case with higher R-factor values (lower critical power values) yielded slightly lower PCT than the nominal case, and vice versa. The lower critical power values led to slightly earlier boiling transition, which by itself would be expected to give a higher PCT. However, the NRC staff acknowledges that the lower critical power values could affect the transient progression on other ways as well, such as affecting the boundaries between correlations used for determining heat transfer coefficients and possibly indirectly affecting the interaction between peak and average rods in neighboring fuel assemblies when boiling transition occurs.

Due to the competing effects involved, the NRC staff finds it plausible that an increase in critical power could cause a slight increase in PCT, and vice versa. Moreover, the response to SRXB-RAI-6 indicated that the PCT is only weakly sensitive to adjustments to the GEXL critical power values. Therefore, the NRC staff finds the licensee's selection of higher critical power values in the most limiting fuel parameter sensitivity case to be acceptable. Thus, then NRC staff considers this RAI closed.

RAI SRXB-RAI-11

Please provide steady-state core simulator comparisons for a representative BSEP MELLLA+ cycle using GEH and AREVA methods, to support the licensee's conclusion that the GEH methods have modeled ATRIUM-10XM in a satisfactory manner. Include comparisons of calculated results such as power and exposure distributions, hot eigenvalue, active or bypass flow rates, and core pressure drop.

Evaluation:

In the RAI response, the licensee has demonstrated that the power distribution, burnup distributions, hot eigenvalue, flow rates, and core pressure drop are in sufficient agreement between GEH and AREVA methods for the NRC staff to conclude that ATRIUM-10XM fuel has been modeled in a satisfactory manner using GEH methods for BSEP MELLLA+. Thus, the NRC staff considers this RAI closed.

SRXB-RAI-12

Please provide justification that SLMCPR Penalty of 0.03 is not applicable to BSEP. Please provide data representative of the requested MELLLA+ Operating Domain for BSEP to justify the use of MICROBURN-B2 in the domain. If the data doesn't cover the entire operating domain, please identify the range that is not covered and identify any additional conservatisms that can be used to justify the adequacy of the method in this range.

Evaluation:

In the RAI response, the licensee provided seven cycles of 2D TIP uncertainty data for BSEP, with average 2D uncertainty of approximately [[ ]] and the highest uncertainty being approximately [[ ]] This is significantly less than the [[ ]] uncertainty conservatively assumed for the SLMCPR calculation for BSEP, which is taken from EMF-2158P. The BSEP measured uncertainties are lower primarily because of the use of gamma TIPs in BSEP, which tend to have lower uncertainties and less sensitivity to void fraction. In ANP-3108P, TIP data were presented for other plants as well. None of these TIP data show a discernible trend in uncertainty with respect to power-to-flow ratio, core average void fraction, or power; however, the Brunswick data only extend as high as approximately 39 MWt-hr/Mlb, with the majority of data below 38 MWt-hr/Mlb. Although the data for the other plants included power-to-flow ratios up to 52 MWt-hr/Mlb, few data were obtained above 42 MWt-hr/Mlb. For BSEP, 42 MWt-hr/Mlb encompasses a large portion of the MELLLA+ domain, with 52 MWt-hr/Mlb being exceeded in only a small corner of the MELLLA+ domain, which will not typically be entered during normal cycle operation.

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Therefore, the NRC staff finds it reasonable to conclude that the bundle power distribution uncertainties will not increase sufficiently at the higher MELLLA+ power-to-flow ratios to make the power distribution uncertainties in EMF-2158P inapplicable.

In Reference 10, the licensee described testing that will be performed at BSEP prior to the first cycle of MELLLA+ operation, including collection of TIP data on each unit near 100% power and 85% core flow, and near 77.6% power and 55% core flow. If TIP uncertainties exceeding a value representing the upper bound of previous TIP uncertainties at BSEP (which support the uncertainties assumed in EMF-2158P), the licensee will enter the adverse condition in the Corrective Action Program and investigated to determine appropriate corrective actions.

[[

]] the low TIP uncertainties associated with the BSEP gamma TIP system, which adds was not credited in the BSEP SLMCPR analysis. Thus, the NRC staff considers this RAI closed.

SRXB-RAI-13

AREVA ANP-3280P - *Brunswick Unit 1 Cycle 19 MELLLA+ Reload Safety Analysis, May 2016*, includes reference reload analyses for a mixed ATRIUM-10XM/ATRIUM-10 core. However, Section 1.0 of ANP-3280P indicates that ATRIUM-10 limits and analyses are not included in these reload analyses. Please explain this approach in more detail to ensure that it is appropriate for a full core of ATRIUM-10XM, which will be the fuel type when MELLLA+ is implemented. For example, were the geometric and performance features of the ATRIUM 10 fuel assemblies modeled explicitly, while the SLMCPR calculations (such as those shown in Table 4.2 of ANP-3280P) considered only the ATRIUM-10XM fuel assemblies for Cycle 18? How does Section A.1 of AREVA ANP-3108P *Applicability of AREVA NP BWR Methods to Brunswick Extended Power Flow Operating Domain, July 2015*, regarding mixed cores relate to the analyses performed in ANP-3280P?

Evaluation:

In the RAI response, the licensee clarifies that the analyses in ANP-3280P were performed for the cycle-specific loading for BSEP Unit 1 Cycle 19, including appropriate input data for both ATRIUM-10 and ATRIUM-10XM fuel. ATRIUM-10 fuel-specific results were omitted from ANP-3280P.

The NRC staff finds this approach acceptable because the cycle calculations, including SLMCPR, were fully representative of the exact mixed core configuration of Unit 1 Cycle 19 and the ATRIUM-10 specific results were simply not presented because they were less relevant to BSEP MELLLA+, which includes only ATRIUM-10XM. Therefore, the NRC staff finds that the reference reload analyses were performed in an acceptable manner applicable to BSEP MELLLA+, and this RAI is closed.

SNPB-RAI-1

Section 6.0 of ANP-3108 *Mechanical Limits Methodology* describes how the fuel mechanical design criteria are satisfied. Provide the details for the following aspects of the mechanical design for the extended power/flow operation at the Brunswick units:

a. Provide a summary description how fuel rod design criteria was applied for Extended Power/Flow Operating Domain (EPFO domain) operation.

b. How the fuel design limits such as LHGR and burnup are established for the EPFO domain operation.

c. Provide details of how the uncertainties (operating power, code model parameters, and fuel manufacturing tolerances) are utilized in the mechanical design analysis. Also, list the values and source of uncertainties utilized in the analysis

Evaluation

(a) The licensee responded with a summary of the application of the analysis methodology to the ATRIUM 10XM design provided in Section 3.2 of ANP-2950P, Revision 0, "ATRIUM 10XM Fuel Rod Thermal and Mechanical Evaluation for Brunswick Unit 2 Cycle 20 Reload BRK2-20, October 2010.

(b) The licensee responded that the determination of the Fuel Design Limit (FDL) and fuel rod exposure limit are unchanged for operation in MELLLA+.

The utilization of uncertainties (operating power, code model parameters and fuel manufacturing tolerances) is fully described in BAW-10247PA, Revision 0, "Realistic Thermal-Mechanical Fuel Rod Methodology for Boiling Water Reactors," AREVA NP, February 2008. The manufacturing uncertainties are compiled on an annual basis and statistically abstracted on a triennial basis. The most recent triennial statistics are shown in Table SNPB-1-1 of the RAI response document (Reference 7). These statistics typically show very little drift from year to year.

The NRC staff reviewed the above response and the supporting references and finds the fuel mechanical design criteria are satisfied. Thus, the NRC staff considers this RAI closed.

SNPB-RAI-2

Section 6.2 of ANP-3280P, Revision 1, "Brunswick Unit 1 Cycle 19 MELLLA+ Reload Safety Analysis," provides a short summary of Control Rod Drop Accident (CRDA). The CRDA analysis for both A and B sequence startups was performed/resolutioned using the methodology described in Topical Report (TR), XN-NF-80-19(P)(A) Volume 1 Supplements 1 & 2, "Exxon Nuclear Methodology for Boiling Water Reactors – Neutronic Methods for Design and Analysis," (1983). The licensee has reported that that the maximum fuel rod enthalpy is less than the NRC threshold of 280 cal/g.

Appendix B of Standard Review Plan (NUREG-0800, March 2007) provides interim acceptance criteria for the reactivity initiated accidents, such as, CRDA for BWRs. The technical and regulatory basis for the interim criteria is documented in a memorandum dated January 19, 2007 (ADAMS No. ML070220400). This memorandum, with respect to the criteria on fuel enthalpy for maintaining coolable geometry, stated that the 280 cal/g criteria has been found inadequate to ensure fuel rod geometry and long term coolability. NUREG-0800 Section 4.2 defines reactivity-initiated accident fuel clad failure criteria as (1) radial average fuel enthalpy greater than 170 cal/g for BWR at zero or low power, and (2) local heat flux exceeding fuel thermal design limits (CPR) at-power events in BWRs. The technical basis for the BWR fuel failure criteria is detailed in the January 19, 2007 memorandum.

The draft regulatory guide (DG-1327 of November 2016) defines fuel cladding failure thresholds, analytical limits and guidance for demonstrating compliance with applicable regulations governing reactivity limits. The empirically based PCMI failure thresholds are shown in Figures 2 through 5 for fully recrystallized annealed (RXA) and stress relief annealed cladding types at both low and high temperature reactor coolant conditions. The PCMI cladding failure threshold is expressed in peak radial average fuel enthalpy rise (Δcal/g) versus excess cladding hydrogen content in weight parts per million.

Based on the background information given, please provide an evaluation to show that the fuel failure criteria as proposed by either SRP Section 4.2 (2007) or DG-1327 (2016) can be met for the control rod drop accident analysis at BSEP Units 1 and 2.

<u>Evaluation</u>:

The licensee has considered the anticipated change in the regulatory guidance and provided an assessment of the CRDA analysis using the new draft criteria provided in the Draft Regulatory guide DG-1327. The new assessment performed by Framatome is described in the Framatome report, ANP-3656P, Revision 0, *Brunswick MELLLA+ CRDA Assessment with Draft Criteria* (Reference 29). The results are reported in this SE Section 3.9.1. The NRC staff reviewed the submitted report and find that the results from the new CRDA assessment are within the acceptance criteria established in the draft regulatory guide DG-1327. Thus, the NRC staff considers this RAI closed.

SNPB-RAI-3

Appendix B of ANP-3108P describes void-quality correlations, **[[ ]]** and Ohkawa-Lahey, and AREVA's independent validation of these correlations for application to ATRIUM 10XM and Brunswick plant operation at EPFO domain conditions. Please provide responses for the following items.

(a) Detailed description of AREVA's independent validation of the **[[ ]]** and Ohkawa-Lahey correlations using the test data from FRIGG experiments that generated Table B-1 and Figures B-1 through B-4 of ANP-3108P.

(b) Supporting calculation detail that shows how the uncertainties and biases are utilized in the analyses described in Sections B.2 and B.3 of ANP-3108P.

<u>Evaluation</u>

(a)

Framatome's independent validation of the BWR void fraction correlations against the FRIGG experiments was performed to supplement and confirm the validation performed by the original developers of the correlations. For each of the experimental tests, the nominal test conditions were used to calculate the void fraction at each of the measurement locations. These calculated void fractions were then compared to the nominal measured void fractions and plotted in Figures B-1 and B-3 in Appendix B of ANP-3108P. Concurrent with the Framatome fuel design development that lead to the introduction of partial length fuel rods (PLFRs) and swirl vane spacer designs, fundamental testing was conducted to assess the adequacy of current methodologies in predicting steady-state behavior (critical power, pressure drop and void fraction) as well as dynamic behavior (i.e., channel decay ratio and instabilities). Framatome performed void fraction measurements to specifically assess the impact of the ATRIUM-10 fuel design attributes. The void fraction tests were performed at the KATHY test facility using a prototypical BWR CHF test assembly. The test assembly used 8 PLFRs, mixing vane grids, a **[[**
]] typical of CHF tests.

To assess the high void fraction performance associated with EPFO domain, AREVA performed gamma tomography measurements on the ATRIUM-10XM (i.e., the fuel design used in Brunswick). Again, the tests were performed at the KATHY test facility using a prototypical BWR CHF test assembly.

The NRC staff reviewed the response from Framatome and finds that the modern AREVA fuel design attributes and the potential operation at high void fractions for EPFO domain result in no significant change in the void-quality correlation uncertainties relative to the historic validation. Thus, the NRC staff considers this RAI closed.

(b)

The analyses in Sections B.2 and B.3 of ANP-3108P present the OLMCPR impact of biasing the nominal void fraction correlations over the uncertainty range relative to the inherent conservatism of the approved licensing methodologies on the OLMCPR. Three cases are presented relative to the nominal licensing application in the response. For each of the cases listed, a multi-cycle depletion with MICROBURN-B2 was performed to establish the end-of-cycle core conditions (exposure, power flow and reactivity parameters) in equilibrium with the revised void quality correlation. The end-of-cycle condition was then used in the downstream transient analyses to quantify the impact on the OLMCPR. The NRC staff reviewed the RAI response that provide the supporting calculation that shows how the uncertainties and biases are properly used in the analyses described in Sections B.2 and B.3 of ANP-3108P. Thus, the NRC staff finds the response acceptable and considers this RAI is closed.

SNPB-RAI-4

It has been stated in Section 9.3.3 *ATWS with Core Instability* of DUKE-0B21-1104-000(P) (M+ LTR) that sensitivity studies are performed with GEXL by varying from **[[ ]]**, and conservative GEXL parameters have been developed by AREVA for Application to ATRIUM 10XM fuel. The NRC staff could not find any reference to this GEXL correlation and could not determine which version of GEXL correlation has been utilized and which sensitivity analysis has been performed.

(a) Please provide details of the formulation of the above GEXL correlation formulation.

(b) Do the biases and uncertainties associated with this GEXL correlation comply with the requirements for the ATRIUM 10XM fuel design?

(c) Is this GEXL correlation compatible with TRACG code?

(d) Are the R-factors (K-factors) and additive constants specifically derived for the ATRIUM 10XM fuel design?

Evaluation:

(a)

The GEXL97 correlation for ATRIUM-10 fuel was transmitted by GEH to Framatome. This transmittal included both the correlation form and the coefficients for ATRIUM-10 fuel. Framatome then modified the coefficients as necessary to conservatively model the ATRIUM 10XM fuel. The data used to benchmark this correlation is the same data set used to benchmark the ACE/ATRIUM 10XM correlation. No changes to the correlation form were permitted, so only changes to the correlation coefficients were allowed. The range of applicability for this correlation was chosen to be within the range of the data available. The NRC staff finds this response provided information requested and thus considers this RAI closed.

(b)

The GEXL correlation for ATRIUM 10XM was not constructed as a general use correlation. Instead, it was biased to provide a conservative CPR prediction specifically for the ATWS-I event. As such, this correlation is not valid for all applications. The correlation was designed to provide conservative estimates for critical power during ATWS-I scenario. This correlation was created to be closer to best estimate, but still conservatively low, at the low flow conditions of interest for the ATWS-I scenario. For other assembly flows the correlation is more conservative. To illustrate this trend, the estimated CPR versus mass flux for the benchmarking data set was provided in the response. This figure shows the trend of increasing conservatism with increasing flow, and also shows the overall conservatism of the correlation. The NRC staff finds this response provided the information requested and considers this RAI closed.

(c)

The licensee responded that this GEXL correlation is compatible with the TRACG code. The NRC staff finds this response provided the information requested and considers this RAI closed.

(d)

Framatome utilized the ACE K-factor in place of the GEXL R-Factor in the CPR calculation. The subsequent benchmarking to the data described above showed this to be a reasonable assumption. The assembly specific R-factors were calculated by Framatome for all assemblies at each core exposure analyzed for ATWS-I using the K-factor method. In order to ensure that the K-factors indeed produced conservative results with the GEXL correlation, the GEXL correlation was implemented in MICROBURN-B2. Critical power calculations were performed for the limiting assemblies at two exposures (BOC and EOC) to demonstrate that the K-factors and the provided GEXL correlation produced a conservative prediction of critical power when compared to the approved ACE/ATRIUM 10XM correlation. These calculations were performed at both high flow and low flow statepoints and confirmed the overall conservatism of the correlation, as well as the trend in increasing conservatism with increasing flow. The NRC staff finds this information responsive and considers this RAI closed.

SNPB-RAI-5

Appendix A page A-2 indicates that "*At the time of the creation of this document, Reference 7 (ANP_10298PA Revision 0 Supplement 1P Revision 0 Improved K-factor Model for ACE/ATRIUM 10XM Critical Power Correlation, AREVA NP, December 2011) had not been generically approved.*" The NRC staff notices that Revision 1 of ANP-10298P-A was published as the accepted version (-A) incorporating Reference 7 in to Reference 2 (ANP-10298PA Revision 0) in March 2014. However, the Brunswick EPFO domain analysis was performed using the Revision 0 when Revision 1 of the topical report was available.

Please explain why the Revision 1 was not used for the Brunswick analysis. What is the impact on results if the Revision 1 is used in the analysis?

<u>Evaluation</u>:

The licensee/Framatome responded that the analyses for the representative cycle (BSEP Unit 1 Cycle 19) were performed using the critical power correlation consistent with the TS at the time of BSEP Unit 1 Cycle 19. Since that time, the BSEP Unit 1 TS has been updated (i.e., Amendment 269) to reference Revision 1 of ANP-10298PA (ADAMS Accession No. ML16019A029). Comparison of ΔCPRs between Revision 0 and Revision 1 of ANP-10298PA for BSEP limiting events has shown that there is no adverse impact for using

Revision 0 of the LTR. The NRC staff finds this information responsive and considers this RAI closed.

SNPB-RAI-6

This RAI is withdrawn upon satisfactory response during the audit.

SNPB-RAI-7

For ATWS (Licensing basis) calculations (Section 9.3.1 of DUKE-0B21-1104-000(P) (M+ LTR)) and for ATWS with core instability (Section 9.3.3 of M+ LTR), to address the effect of ATRIUM 10XM fuel, sensitivity analyses were performed. Sensitivity ranges are selected to include expected variation of the parameters; direct energy deposition, gap conductance as applied to ATRIUM 10XM as supplied by PRIME, thermal and hydraulic channel losses. Please respond to the following questions.

a. Discuss the suitability of using ODYN code for ATRIUM 10XM fuel design.

b. It is stated that gap conductance data as applied to ATRIUM 10XM fuel data is supplied by the PRIME code. However, Section 4.0 of safety evaluation for PRIME *limits the use of PRIME to "approved GNF fuel rods designs clad in RXA Zircaloy-2."* This limitation further states that *"In case core transition from one vendor to another, PRIME may be applied to generate inputs for the downstream safety analyses and overpower limit compliance for the non-GNF BWR fuel, provided that the design and operating parameters for the non-GNF fuel must be within the range approved for the PRIME models;*

> 1. Since the cladding type for the ATRIUM 10XM is different from the cladding type of GNF fuel designs (RXA), please justify how PRIME can be used for the ATRIUM 10XM analysis.
>
> 2. To satisfy the limitation for PRIME stated above, show that the ATRIUM 10XM design and operating parameters are within the ranges of the parameters approved for PRIME In order to use PRIME for the ATRIUM 10XM analyses, has any modifications been performed in the PRIME code to enable it to perform ATRIUM 10XM analyses?

<u>Evaluation</u>:

a.

The licensee stated in the LAR that the GEH transient analysis model, ODYN, had been used with and is qualified to apply to different fuels. The same method for GNF fuel application is applied for ATRIUM fuel. ATRIUM-10 fuel has been used with ODYN for LaSalle, Columbia, River Bend, and Grand Gulf. ATRIUM-10 ATWS calculations have also been performed using ODYN to support Susquehanna. ATRIUM-10 and ATRIUM 10XM are very similar except for the inventory of full- and part-length rods. For the BSEP ATWS analyses, the ATRIUM 10XM fuel geometry was modeled explicitly. The core parameters were introduced into ODYN in the same manner via PANAC wrap-ups as would be done for GNF and ATRIUM-10 fuel designs. ODYN is suitable for ATRIUM-10, and as a result remains suitable for ATRIUM 10XM. The NRC staff finds this information responsive and considers this RAI closed.

b.

The licensee stated the PRIME computer model provides best-estimate predictions of the thermal mechanical performance of nuclear fuel rods. For BSEP ATWS analyses, the PRIME code was not used for any thermal mechanical evaluation. The PRIME code was only used to create the fuel files needed to determine the gap conductance.

1. Any differences in heat transfer from differences in the cladding thermal properties are insignificant compared to the gap conductance heat transfer. Gap conductivity sensitivities cover a large range of uncertainty and cover any variation in cladding heat transfer properties. The impact of the large variability in gap conductance had a small impact on PCT.

2. The PRIME code was only used to create the fuel files needed to determine the gap conductance; therefore, many of the operating parameters are not applicable. The fuel parameter sensitivity study accounts for uncertainty in the fuel design.
No modifications were made to the PRIME code.

The NRC staff finds this information responsive and considers this RAI closed.

SRXB-C-RAIs

The following is a summary of the NRC staff's evaluation of licensee's responses to the SRXB-C-RAIs on containment analysis in the following licensee submittals:

- SRXB-C-RAI 1: Letter dated April 6, 2017 (Reference 33)
- SRXB-C-RAI 2 through 7: Letter dated November 1, 2017 (Reference 31)

In Enclosure 11 to the LAR, the licensee provided responses to the items in Enclosure 1 of SECY-11-0014 regarding crediting containment accident pressure (CAP) for the net positive suction head (NPSH) analysis for the emergency core cooling system (ECCS) and containment heat removal system pumps that draw water from the suppression pool during a design basis accident or special events. By a supplement letter dated February 14, 2018, the licensee informed the NRC that it is withdrawing Enclosure 11 to the LAR and the related RAIs SRXB-C RAI 1, 5, 6, and 7 responses and stated its position that SECY 11-0014 would not apply to BSEP MELLLA+ changes. The NRC staff finds the withdrawal acceptable because in the MELLLA+ NPSH analysis, the licensee did not credit additional CAP credit beyond what was approved for the EPU.

The following is a summary of NRC staff evaluation of the licensee's responses to the SRXB RAIs SRXB-C-RAI 2, 3, and 4.

SRXB-C-RAI 2

To assure that the containment design conditions (i.e., its design pressure and temperature) are not exceeded during a Loss-of-Coolant Accident (LOCA) in the MELLLA+ operating domain, it is necessary to determine their peak values of these conditions for a bounding case.

Refer to Section 4.1.1 of the SAR; provide the list of the analyzed cases for the recirculation and main steam line break LOCAs that formed the basis for the limiting primary containment response due to a postulated LOCA as initiated from 102% power / 85% core flow) (Figure 1-1 in SAR, MELLLA+ statepoint N). Include the calculated primary containment pressure and temperature results corresponding to each case analyzed in the list. Provide justification that the list is complete and no further cases are necessary to be analyzed.

If the MELLLA+ statepoint N in Figure 1-1 of SAR does not generate the limiting primary containment temperature and pressure responses, please include the analysis cases and results from the other statepoints that determined the limiting case.

Evaluation:

The licensee stated that for all BWRs with Mark I containment, the limiting break for a design basis accident for short-term containment pressure and temperature is the double-ended guillotine recirculation suction line break (RSLB). Therefore, [[ ]] To determine the limiting power/flow point in the MELLLA+ domain, referring to Figure 1-1 in SAR the licensee performed sensitivity analysis for [[ ]]

In the sensitivity analysis for the limiting MELLLA+ power/flow points, the licensee's calculated the short-term peak drywell pressure to be 45.8 psig and the peak drywell temperature to be 292.3°F, both limiting for the RSLB LOCA case. The MELLLA+ short-term peak drywell pressure and temperature are bounded by their current values 46.4 psig and 293.0°F and remain below the design limits of 62 psig and 340°F respectively.

Based on the sensitivity analysis, the NRC staff finds that the short term containment pressure and temperature response in the MELLLA+ operating domain is bounded by the current short term pressure and temperature response and the GDC 16 and 50 requirements are met in the MELLLA+ operating domain. The NRC staff finds this information responsive and considers this RAI closed.

SRXB-C-RAI 3

In order to meet the requirement of GDC 4, it is necessary to assure that the Condensation Oscillation (CO) load, which is one of the dynamic load imposed on the containment and its internal SSCs, during a LOCA in the MELLLA+ operating domain is within the design limits and the SSCs are adequately protected.

Section 4.1.1 of SAR under heading "Condensation Oscillation Loads" states:

> The Mark I CO [[ ]] load definition was developed from test data from Full Scale Test Facility (FSTF) tests (Reference 33 [GE Nuclear Energy, "Mark I Containment Program, Full Scale Test Program Final Report, Task Number 5.11," NEDE-24539-P, April 1979]) to simulate LOCA thermal-hydraulic conditions (i.e., [[ ]]. The tests are bounding for all US Mark I plants, including the BSEP, considering MELLLA+ conditions.

Explain why the FSTF tests results are bounding for the BSEP Units 1 and 2 LOCA CO loads in the MELLLA+ operating domain.

Evaluation:

The licensee stated that the [[

]]

The NRC staff finds the licensee's explanation and results acceptable because the [[

]] The GDC 4 requirements are met. The NRC staff finds this information responsive and considers this RAI closed.

SRXB-C-RAI 4

In order to meet the requirement of GDC 4, it is necessary to assure that the chugging load, which is one of the dynamic load imposed on the containment and its internal SSCs during a LOCA in the MELLLA+ operating domain is within the design limits and the SSCs are adequately protected.

Section 4.1.1 of SAR under heading "Chugging Loads" states:

> The thermal-hydraulic conditions for these tests [[
>]] were selected to produce maximum chugging amplitudes so that it bound all Mark I plants. Therefore, the current chugging load definitions remain applicable at MELLLA+ conditions for BSEP.

Explain why the FSTF tests (NEDE-24539-P, April 1979 (ADAMS Accession No. ML081900313)) results are bounding for the BSEP Units 1 and 2 LOCA chugging loads in the MELLLA+ operating domain.

Evaluation:

The licensee stated that [[

]] Regarding the chugging test program the licensee stated that the Mark I containment test program [[

]]

The NRC staff finds the licensee's RAI response on the LOCA chugging loads is acceptable because the [[

]] Thus, the NRC staff finds GDC 4 requirements are met and considers this RAI closed.

APHB-RAI-1

The licensee stated in Section 10.6 of the LAR:

Consistent with the requirements for the plant-specific analysis as described in the M+ LTR, the operator training program and plant simulator will be evaluated to determine the specific changes required. Simulator changes and fidelity validation will be performed in accordance with applicable American National Standards Institute (ANSI) standards currently being used at the training simulator.

The NRC staff typically uses ANSI/ANS 3.5-2009, "Nuclear Power Plant Simulators for Use in Operator Training" as a review tool for training program development.

a. Please specify the ANSI standard(s) that the licensee proposes to use at the training simulator.

b. If ANSI/ANS 3.5-2009 is not the standard the licensee proposes for the license amendment, briefly explain whether this standard is appropriate.

Evaluation:

The licensee responded that BSEP simulator changes and fidelity validation will be performed in accordance with ANSI/ANS 3.5-2009. The NRC staff finds this information responsive and considers this RAI closed.

APHB-RAI-2

The licensee stated in Enclosure 6, Section 10.5.3, "Operator Response," that there are no new operator actions and there is no significant reduction in the time for operator actions. In the same section, the licensee stated that a "new operator action to require initiation of lowering RPV [reactor pressure vessel] water level within 120 sec to mitigate ATWS [Anticipated Transient Without Scram] instability events is judged not to impact the PRA."

a. Clarify if there are any time-critical operator actions that will be added, deleted, or changed to support the proposed license amendment, including the reduction in time available to complete the action(s).

b. If there is reduction in time available for operators to complete the action, justify and provide the process used to validate that all necessary operator actions can be reliably completed within the available time.

Evaluation:

The licensee responded that are no new operator actions required for MELLLA+ to be implemented at BSEP and that the operator actions in response to an ATWS with MELLLA+ remain consistent with the current operator actions without MELLLA+. While no new operator actions are required for MELLLA+ to be implemented, the 120-sec time requirement to initiate lowering RPV water level to mitigate ATWS instability events will be classified as a TCOA and tracked and managed in accordance with BSEP plant procedure 0AP-064, "Time Critical Operator Actions." The response also stated that there is no reduction in time available for operators to initiate this action. The NRC staff finds this information responsive and this RAI closed.

EMIB-RAI-1

NEDC-33006P-A, Rev 3 (Reference 5) states that the relief and safety valves and the reactor protection system provide overpressure protection for the reactor coolant pressure boundary (RCPB) during power operation. 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 GDC-14, GDC-33, GDC-34, and GDC-35, which require that the RCPB be designed to assure that it behaves in a non-brittle manner and that the probability of rapidly propagating type failures is minimized. In addition, specific review criteria are contained in SRP Section 5.2.2 and Matrix 8 Table of RS-001 provide guidance for using pressure relief systems for RCPB overpressure protection. The overall objective is to prevent failure of the nuclear system pressure boundary and uncontrolled release of fission products, during abnormal operational transients, the ASME Upset overpressure protection event, and postulated ATWS events. Section 9.3.1 of the BSEP SAR evaluates the ATWS response for operation at the MELLLA+ operating domain.

Since the evaluation of the steam separator and dryer performance at MELLLA+ conditions indicates an increase in moisture carry over (MCO) of < 0.20 wt % where the original MCO performance specification was 0.10 wt. Please discuss the impact of the higher moisture concentration on components in the main steam lines, including MSIVs and flow restrictors.

Evaluation:

The licensee responded that:

> Any increase in the steam moisture content has the potential to increase erosion/corrosion rates in the main steam line (MSL) piping and components. It can be reasonably assumed that the rate of flow accelerated corrosion (FAC) in the MSLs, moisture separator, and steam drains is proportional to the local steam moisture content. The increase in the steam dryer outlet moisture from 0.1 wt. % to 0.2 wt. % is insignificant with respect to pipe erosion/corrosion. The 0.1 wt. % increase in MCO will cause a commensurate increase in steam moisture content at any location in the MSL. Nonetheless, the small increase in steam moisture content (or corresponding reduction in steam quality) will have only a negligible effect on MSL pipe erosion/corrosion. The plant's FAC program, which includes the MSL, monitors susceptible areas for corrosion and factors the results into the piping replacement program at the plant site so that any adverse impact on the MSL piping will be monitored by the plant. As discussed in

Section 10.7.2 of BSEP SAR, "The BSEP FAC implementing documents...consider the FAC susceptibility of lines with steam quality above 99.5% to be low." In addition, Section 3.3.4 of BSEP SAR concludes that, "The amount of time BSEP is operated with higher than the original design moisture content (0.10 wt.%) is expected to be minimal" since the increased MCO is only expected to occur when the plant is operating at or near the MELLLA+ minimum core flow. Therefore, the slight increase in steam moisture (i.e., an increase of 0.1 wt. % at any location) is not expected to have a significant impact on the MSL.

The principal MSL components that are potentially susceptible to higher steam moisture content are the main steam isolation valves (MSIVs), the safety relief valves (SRVs), the flow elements (or flow restrictors), and the high-pressure turbine. For BWR/4 plants equipped with GE-manufactured turbines such as BSEP, the MSIV is typically the limiting MSL component with respect to increased MCO.

MSIVs

The local steam moisture content at the MSIVs is expected to increase slightly from approximately 0.25 wt.% to 0.26 wt.%. Because all critical surfaces of the MSIVs are hardfaced, erosion of these surfaces due to the slightly higher moisture levels is not a concern. Further, since the MSIVs are leak tested at every outage, potential degradation will be monitored and corrected if observed, as required by plant operating procedures. Therefore, no adverse impacts to the MSIVs are expected under MELLLA+ operating conditions with the steam dryer outlet moisture of 0.2 wt.%.

SRVs

The SRVs are required to operate under high-quality two-phase flow conditions and have been demonstrated to have the capability to function with water flows at low-pressure conditions. The purchase specification for the SRVs stipulate a steam moisture content of < 1.0 wt.% under normal operating conditions. Since the increase in MCO from 0.1 wt.% to 0.2 wt.% does not cause the moisture content at the SRVs to exceed this value, operation at the higher moisture content is acceptable.

Flow Elements (Flow Restrictors)

Increasing MCO from 0.1 wt.% to 0.2 wt.% produces a negligible (i.e., < 0.1%) change in the specific volume of the steam leaving the reactor pressure vessel. The resulting difference in specific volume at the steam flow restrictors does not affect steam flow measurement. Similarly, the increase in MCO produces a negligible (i.e., < 0.1%) increase in the steam density, which causes a slight reduction in the margin to choke flow (i.e., ~0.4%) through the flow restrictors. In addition, the increase in MCO to 0.2 wt.% reduces the local steam quality at the flow elements from approximately 99.75 wt.% to 99.65 wt.%, but remains above the 99.5 wt.% steam quality criterion for low FAC susceptibility. Therefore, the slight increase in MCO has no impact on the flow restrictors.

High-Pressure Turbine

The higher MCO increases the steam moisture content entering the high-pressure turbine. At 0.20 wt.% MCO, the moisture content of steam entering the high-pressure turbine is expected to increase from 0.51 wt.% to 0.61 wt.%. Nonetheless, this increase remains well below the maximum inlet steam moisture content for the turbine **[[ ]]**. Therefore, the slight increase in steam moisture has no impact on the high-pressure turbine.

In summary, there are no adverse impacts to the MSL piping and components associated with continuous reactor operation with MCO less than or equal to 0.2 wt.%.

The NRC staff review the RAI response and concludes that the generic M+ LTR treatment remain applicable to BSEP even with the slight increase of 0.1 wt.% in MCO during operation in the MELLLA+ domain because this slight increase in MCO is bounded by current plant operation with Steam Separator and Dryer Performance as well as Steam Line Moisture Performance Specifications.

<u>Conclusion</u>:

Based on its review, the NRC staff finds the information provided the license are responsive and considers all the RAI closed.

APPENDIX G

Acronyms and Initialisms

Term	Definition
1RPT	one reactor pump trip
2RPT	two reactor pump trip
ABSP	automated backup stability protection
AEC	Atomic Energy Commission
ALARA	as low as reasonably achievable
ANSI	American National Standards Institute
AOO	anticipated operational occurrence
AOP	abnormal operating procedure
APRM	average power range monitor
ARI	alternate rod insertion
ASME	American Society of Mechanical Engineers
ATWS	anticipated transient without scram
ATWS-I	anticipated transient without scram - instability
BOC	beginning of cycle
BOP	balance of plant
BSEP	Brunswick Steam Electric Plant, Units 1 and 2
BSEP SAR	BSEP MELLLA+ Safety Analysis Report, also called the M+ SAR
BSP	backup stability protection
BWR	boiling-water reactor
BWRVIP	Boiling-Water Reactor Vessel and Internal Project
CAP	containment accident pressure
CDA	confirmation density algorithm
CDF	core damage frequency
CF	core flow
CFR	Code of Federal Regulations
CHF	critical heat flux
CLTP	current licensed thermal power
CO	condensation oscillation
COLR	core operating limits report
CPR	critical power ratio
CRDA	control rod drop accident
CRWE	control rod withdrawal error
CS	core spray
DBA	design-basis accident
DORL	Division of Operating Reactor Licensing

Term	Definition
DR	decay ratio
DSS-CD	detect and suppress solution - confirmation density
ECCS	emergency core cooling system
ENC	Exxon Nuclear Company, Inc.
EOC	end of cycle
EOP	Emergency Operating Procedure
EPFO	extended power/flow operating
EPG	Emergency Procedure Guideline
EPU	extended power uprate
FAC	flow accelerated corrosion
FDL	fuel design limit
FFWTR	final feedwater temperature reduction
FIV	flow induced vibration
FSTF	Full Scale Test Facility
FTR	failure to rewet
FW	feedwater
FWCF	feedwater controller failure
FWHOOS	feedwater heater out of service
GDC	General Design Criterion / Criteria
GE	General Electric
GEH	GE-Hitachi Nuclear Energy Americas LLC
GL	Generic Letter
GNF	Global Nuclear Fuel
HCTL	heat capacity temperature limit
HELB	high energy line break
HGAP	heat transfer coefficient of the gap between the fuel pellet and cladding
HN+CT	homogeneous nucleation plus contact temperature
HPCI	High-pressure coolant injection
HSBW	hot shutdown Boron weight
HVAC	heating, ventilation and air conditioning
IASCC	irradiated assisted stress corrosion cracking
ICF	increased core flow
IEEE	Institute of Electrical and Electronics Engineers
IGSCC	intergranular stress corrosion cracking
L&C	limitation and condition
LAR	license amendment request
LCO	limiting condition for operation
LDR	Load Definition Report
LFWH	loss of feedwater heating

Term	Definition
LHGR	linear heat generation rate
LHGRFACf	flow dependent LHGR multiplier
LOCA	loss-of-coolant accident
LOOP	loss of offsite power
LPRM	local power range monitor
LRNB	load reject no bypass
LTR	Licensing Topical Report
LTS	long-term stability solution
M+ LTR	NEDC-33006P-A, Rev 3, GEH MELLLA+ Licensing Topical Report
M+ SAR	MELLLA+ Safety Analysis Report
M+ SE	Safety Evaluation Report for NEDC-33006P-A, Rev 3, GEH MELLLA+ LTR
MAPLHGR	maximum average planar linear heat generation rate
MCNP	Monte Carlo N-Particle
MCO	moisture carryover
MCPR	minimum critical power ratio
MELLLA	Maximum Extended Load Line Limit Analysis
MELLLA+	Maximum Extended Load Line Limit Analysis Plus
Methods LTR	NEDC-33173P-A, Rev 4, Applicability of GE Methods to Expanded Operating Domains LTR
Methods SE	Safety Evaluation Report for NEDC-33173P-A, Rev 4, Applicability of GE Methods to Expanded Operating Domains LTR
MIP	minimum critical power ratio importance parameter
MOV	motor-operated valve
MS	main steam
MSIV	main steam isolation valve
MSIVC	main steam isolation valve closure
MWt	megawatts thermal
NPSH	net positive suction head
NRC	U.S. Nuclear Regulatory Commission
NSSS	nuclear steam supply system
OLMCPR	operating limit minimum critical power ratio
OLTP	original licensed thermal power
OPRM	oscillating power range monitor
PBA	period based algorithm
PCMI	pellet-cladding mechanical interaction
PCT	peak cladding temperature
PFR	partial flow reduction
PLFR	partial length fuel rod
PRA	probabilistic risk assessment

Term	Definition
PRFO	pressure regulator failure open
Prms	parameter "root-mean-square pressure"
psig	pounds per square inch gauge
Pu	Plutonium
RAI	Request for Additional Information
RBM	rod-block monitor
RCF	reactor core flow
RCIC	reactor core isolation cooling
RCPB	reactor coolant pressure boundary
RCS	reactor coolant system
RG	Regulatory Guide
RHR	residual heat removal
RIPD	reactor internal pressure difference
RPT	reactor pump trip
RPV	reactor pressure vessel
RS-001	Review Standard 001
RSAR	Reload Safety Analysis Report
RSLB	recirculation suction line break
RWCU	reactor water cleanup
RWE	rod withdrawal error
S_{AD}	amplitude discriminator setpoint
SAFDL	specified acceptable fuel design limit
SAG	severe accident guideline
SAR	safety analysis report
SCC	successive confirmation count
SBO	station blackout
SE	Safety Evaluation
sec	second(s)
SGTS	standby gas treatment system
SLC	standby liquid control
SLMCPR	safety limit minimum critical power ratio
SLO	single-loop operation
SNPB	Nuclear Performance and Code Review Branch
SPCB	Siemens Power Corporation B
SRLR	Supplemental Reload Licensing Report
SRM	Staff Requirements Memorandum
SRP	Standard Review Plan
SRV	safety relief valve

Term	Definition
SRVOOS	safety relief valve out of service
SRXB	Reactor Systems Branch
SSC	system, structure, and component
TCD	thermal conductivity degradation
TCOA	time critical operator action
TH	thermal hydraulic
TIP	traversing in-core probe
TIPOOS	traversing in-core probe out of service
TLO	two-loop operation
T-M	thermal mechanical
T_{max}	time period upper limit
T_{min}	time period lower limit
TS	technical specification
TTNB	turbine trip no bypass
TTWBP	turbine trip with bypass
UFSAR	Updated Final Safety Analysis Report
USI	unresolved safety issue

SUBJECT: BRUNSWICK STEAM ELECTRIC PLANT, UNITS 1 AND 2 – ISSUANCE OF AMENDMENT REGARDING CORE FLOW OPERATING RANGE EXPANSION (MELLLA+) (EPID L-2016-LLA-0009) DATED SEPTEMBER 18, 2018

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