



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

December 26, 2019

Mr. James Barstow, Vice President
Nuclear Regulatory Affairs and
Support Services
Tennessee Valley Authority
1101 Market Street
LP 4A-C
Chattanooga, TN 37402-2801

SUBJECT: BROWNS FERRY NUCLEAR PLANT, UNITS 1, 2, AND 3 - ISSUANCE OF
AMENDMENT NOS. 310, 333, AND 293 REGARDING MAXIMUM EXTENDED
LOAD LINE LIMIT ANALYSIS PLUS (EPID L-2018-LLA-0048)

Dear Mr. Barstow:

The U.S. Nuclear Regulatory Commission (NRC or the Commission) has issued the enclosed Amendment Nos. 310, 333, and 293, to Renewed Facility Operating License (RFOL) Nos. DPR-33, DPR-52, and DPR-68, for the Browns Ferry Nuclear Plant, Units 1, 2, and 3, respectively. These amendments are in response to Tennessee Valley Authority's application dated February 23, 2018, as supplemented by letters dated March 7, 2018; July 23, 2018; December 13, 2018; December 14, 2018; January 16, 2019 (two letters), January 25, 2019; March 13, 2019; April 24, 2019; April 29, 2019; November 25, 2019; and December 19, 2019.

The amendments allow operation in the expanded Maximum Extended Load Line Limit Analysis Plus operating domain and use of the Detect and Suppress Solution - Confirmation Density stability solution.

The NRC staff has completed its review of the information provided by the licensee. Enclosure 4 provides the staff's safety evaluation (SE). The staff has determined that it contains proprietary information pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) Section 2.390, "Public Inspections, Exemptions, Requests for Withholding." Accordingly, the NRC staff has prepared a redacted nonproprietary version (Enclosure 5). The NRC staff will delay placing the nonproprietary SE in the public document room for a period of 10 working days from the date of this letter to allow Tennessee Valley Authority to comment on any proprietary aspects. If you believe that any information in Enclosure 5 is proprietary, please identify such information line by line and define the basis pursuant to the criteria of 10 CFR 2.390. After 10 working days, the nonproprietary SE will be made publicly available.

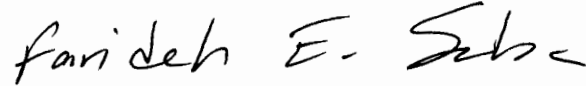
Enclosure 4 transmitted herewith contains Sensitive Unclassified Non-Safeguard Information. When separated from Enclosure 4, this document is decontrolled.

J. Barstow

- 2 -

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

Sincerely,

A handwritten signature in black ink, appearing to read "Farideh E. Saba".

Farideh E. Saba, Senior Project Manager
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-259, 50-260, and 50-296

Enclosures:

1. Amendment No. 310 to DPR-33
2. Amendment No. 333 to DPR-52
3. Amendment No. 293 to DPR-68
4. Safety Evaluation (Proprietary Information)
5. Safety Evaluation (Nonproprietary Information)

cc w/Enclosures 1, 2, 3, and 5: Listserv (**10 days after issuance of the amendments to the licensee**)



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

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-259

BROWNS FERRY NUCLEAR PLANT UNIT 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 310
Renewed License No. DPR-33

1. The Nuclear Regulatory Commission (the Commission) has found that:
 - A. The application for amendment by the Tennessee Valley Authority (the licensee) dated February 23, 2018, as supplemented by letters dated March 7, 2018; July 23, 2018; December 13, 2018; December 14, 2018; January 16, 2019 (two letters), January 25, 2019; March 13, 2019; April 24, 2019; April 29, 2019; November 25, 2019; and December 19, 2019, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;
 - B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;
 - C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;
 - D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and
 - E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.
2. Accordingly, the license is amended by changes to the Renewed Facility Operating License (RFOL) and Technical Specifications as indicated in the attachment to this license amendment.

3. This license amendment is effective as of its date of issuance and shall be implemented within 1 year from the date of its approval.

FOR THE NUCLEAR REGULATORY COMMISSION

A handwritten signature in black ink, appearing to read 'Undine Shoop', is written over the printed name.

Undine Shoop, Chief
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Renewed Facility Operating
License and Technical Specifications

Date of Issuance: December 26, 2019

ATTACHMENT TO LICENSE AMENDMENT NO. 310

BROWNS FERRY NUCLEAR PLANT, UNIT 1

RENEWED FACILITY OPERATING LICENSE NO. DPR-33

DOCKET NO. 50-259

Replace the following pages of Renewed Facility Operating License No. DPR-33 with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

REMOVE

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INSERT

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

REMOVE

3.3-2a

3.3-5
3.3-6
3.3-7
3.4-1
3.4-2
5.0-24
5.0-24c
5.0-25

INSERT

3.3-2a
3.3-2b
3.3-5
3.3-6
3.3-7
3.4-1
3.4-2
5.0-24
5.0-24c
5.0-25

- (3) Pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess, and use at any time any byproduct, source, and special nuclear material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;
- (4) Pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess, and use in amounts as required any byproduct, source, or special nuclear material without restriction to chemical or physical form for sample analysis or equipment and instrument calibration or associated with radioactive apparatus or components;
- (5) Pursuant to the Act and 10 CFR Parts 30 and 70, to possess but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

C. This renewed operating 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 3952 megawatts thermal.

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 310, are hereby incorporated in the renewed operating license. The licensee shall operate the facility in accordance with the Technical Specifications.

For Surveillance Requirements (SRs) that are new in Amendment 234 to Facility Operating License DPR-33, the first performance is due at the end of the first surveillance interval that begins at implementation of the Amendment 234. For SRs that existed prior to Amendment 234, including SRs with modified acceptance criteria and SRs whose frequency of performance is being extended, the first performance is due at the end of the first surveillance interval that begins on the date the surveillance was last performed prior to implementation of Amendment 234.

- (h) The results of the visual inspections of the steam dryer shall be submitted to the NRC staff in a report in accordance with 10 CFR 50.4. The report shall be submitted to the NRC within 90 days following startup from each of the first two respective refueling outages.
- (i) Within 6 months following completion of the second refueling outage, after the implementation of the EPU, the licensee shall submit a long-term steam dryer inspection plan based on industry operating experience along with the baseline inspection results.

The license condition described above shall expire: (1) upon satisfaction of the requirements in items (g) and (h), provided that a visual inspection of the steam dryer does not reveal any new unacceptable flaw(s) or unacceptable flaw growth that is due to fatigue, and (2) upon satisfaction of the requirements specified in item (i).

(19) Neutron Absorber Monitoring Program

The licensee shall, at least once every ten years, withdraw a neutron absorber coupon from the spent fuel pool and perform Boron-10 (B-10) areal density measurement on the coupon. Based on the results of the B-10 areal density measurement, the licensee shall perform any technical evaluations that may be necessary and take appropriate actions using relevant regulatory and licensing processes.

(20) Radiological Consequences Analyses Using Alternative Source Terms

TVA shall perform facility and licensing basis modifications to resolve the non-conforming/degraded condition associated with the Alternate Leakage Treatment pathway such that the current licensing basis dose calculations (approved in License Amendment Nos. 251/282 (Unit 1), 290/308 (Unit 2) and 249/267 (Unit 3)) would remain valid. These facility and licensing basis modifications shall be completed prior to initial power ascension above 3458 MWt.

(21) Prior to extending the frequency for the Integral Leakage Rate Testing described in TS 5.5.12, the licensee shall implement the modifications, that are modeled in the Fire PRA and described in Table S-2, Plant Modifications Committed, of Tennessee Valley Authority letter CNL-18-100, dated October 18, 2018; as supplemented by letter CNL-19-027, dated February 13, 2019.

(22) Maximum Extended Load Line Limit Analysis Plus (MELLLA+) Special Consideration

The licensee shall not operate the facility within the MELLLA+ operating domain with more than a 10°F reduction in feedwater temperature below the design feedwater temperature.

(23) Maximum Extended Load Line Limit Analysis Plus (MELLLA+) Implementation

Prior to the first implementation of MELLLA+, TVA shall perform reload safety analyses using codes that have been corrected for the errors described in TVA letter CNL-19-125, dated December 19, 2019.

- D. The UFSAR supplement, as revised, submitted pursuant to 10 CFR 54.21(d), shall be included in the next scheduled update to the UFSAR required by 10 CFR 50.71(e)(4) following the issuance of this renewed operating license. Until that update is complete, TVA may make changes to the programs and activities described in the supplement without prior Commission approval, provided that TVA evaluates such changes pursuant to the criteria set forth in 10 CFR 50.59 and otherwise complies with the requirements in that section.
- E. The UFSAR supplement, as revised, describes certain future activities to be complete prior to the period of extended operation. TVA shall complete these activities no later than December 20, 2013, and shall notify the NRC in writing when implementation of these activities is complete and can be verified by NRC inspection.
- F. All capsules in the reactor vessel that are removed and tested must meet the test procedures and reporting requirements of the most recent NRC-approved version of the Boiling Water Reactor Vessels and Internals Project (BWRVIP) Integrated Surveillance Program (ISP) appropriate for the configuration of the specimens in the capsule. Any changes to the BWRVIP ISP capsule withdrawal schedule, including spare capsules, must be approved by the NRC prior to implementation. All capsules placed in storage

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
I. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	I.1 Initiate action to implement the Manual Backup Stability Protection (BSP) Regions defined in the COLR.	Immediately
	<u>AND</u>	
	I.2 Implement the Automated BSP Scram Region using the modified APRM Flow Biased Simulated Thermal Power-High scram setpoints defined in the COLR.	12 hours
	<u>AND</u>	
	I.3 Initiate action to submit an OPRM report 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
	<u>AND</u>	
	J.2 Reduce operation to below the BSP Boundary defined in the COLR.	12 hours
	<u>AND</u>	
	J.3 <u>-----NOTE-----</u> LCO 3.0.4 is not applicable. <u>Restore required channel to OPERABLE status.</u>	120 days
K. Required Actions and associated Completion Time of Condition J not met.	K.1 Reduce THERMAL POWER to < 18% RTP.	4 hours

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.1.10	Perform CHANNEL CALIBRATION.	184 days
SR 3.3.1.1.11	(Deleted)	
SR 3.3.1.1.12	Perform CHANNEL FUNCTIONAL TEST.	24 months
SR 3.3.1.1.13	<p>-----NOTE----- Neutron detectors are excluded.</p> <p>-----</p> <p>Perform CHANNEL CALIBRATION.</p>	24 months
SR 3.3.1.1.14	Perform LOGIC SYSTEM FUNCTIONAL TEST.	24 months
SR 3.3.1.1.15	Verify Turbine Stop Valve — Closure and Turbine Control Valve Fast Closure, Trip Oil Pressure — Low Functions are not bypassed when THERMAL POWER is $\geq 26\%$ RTP.	24 months
SR 3.3.1.1.16	<p>-----NOTE----- For Function 2.a, not required to be performed when entering MODE 2 from MODE 1 until 12 hours after entering MODE 2.</p> <p>-----</p> <p>Perform CHANNEL FUNCTIONAL TEST.</p>	184 days
SR 3.3.1.1.17	(Deleted)	

RPS Instrumentation
3.3.1.1

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 3.3.1.1.1 SR 3.3.1.1.3 SR 3.3.1.1.5 SR 3.3.1.1.6 SR 3.3.1.1.9 SR 3.3.1.1.14	≤ 120/125 divisions of full scale
	5(a)	3	H	SR 3.3.1.1.1 SR 3.3.1.1.4 SR 3.3.1.1.9 SR 3.3.1.1.14	≤ 120/125 divisions of full scale
b. Inop	2	3	G	SR 3.3.1.1.3 SR 3.3.1.1.14	NA
	5(a)	3	H	SR 3.3.1.1.4 SR 3.3.1.1.14	NA
2. Average Power Range Monitors					
a. Neutron Flux - High, Setdown	2	3(b)	G	SR 3.3.1.1.1 SR 3.3.1.1.6 SR 3.3.1.1.7 SR 3.3.1.1.13 SR 3.3.1.1.16	≤ 13% RTP
b. Flow Biased Simulated Thermal Power - High	1	3(b)	F	SR 3.3.1.1.1 SR 3.3.1.1.2 SR 3.3.1.1.7 SR 3.3.1.1.13 SR 3.3.1.1.16	≤ 0.61W + 68.3% RTP and ≤ 120% RTP(c)(e)
c. Neutron Flux - High	1	3(b)	F	SR 3.3.1.1.1 SR 3.3.1.1.2 SR 3.3.1.1.7 SR 3.3.1.1.13 SR 3.3.1.1.16	≤ 120% RTP
(continued)					

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

(b) Each APRM channel provides inputs to both trip systems.

(c) [0.55 W + 65.5 – 0.55 Δ W] RTP when reset for single loop operation per LCO 3.4.1, "Recirculation Loops Operating."

(e) With OPRM Upscale (Function 2.f) inoperable, the Automated BSP Scram Region setpoints are implemented in accordance with Action I of this Specification.

RPS Instrumentation
3.3.1.1

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)					
d. Inop	1,2	3(b)	G	SR 3.3.1.1.16	NA
e. 2-Out-Of-4 Voter	1,2	2	G	SR 3.3.1.1.1 SR 3.3.1.1.14 SR 3.3.1.1.16	NA
f. OPRM Upscale	≥18% ^(f)	3(b)	I	SR 3.3.1.1.1 SR 3.3.1.1.7 SR 3.3.1.1.13 SR 3.3.1.1.16	NA
3. Reactor Vessel Steam Dome Pressure - High ^(d)	1,2	2	- G	SR 3.3.1.1.1 SR 3.3.1.1.8 SR 3.3.1.1.10 SR 3.3.1.1.14	≤ 1090 psig
4. Reactor Vessel Water Level - Low, Level 3 ^(d)	1,2	2	G	SR 3.3.1.1.1 SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≥ 528 inches above vessel zero
5. Main Steam Isolation Valve - Closure	1	8	F	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≤ 10% closed
6. High Drywell Pressure	1,2	2	G	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≤ 2.5 psig
7. Scram Discharge Volume Water Level - High					
a. Resistance Temperature Detector	1,2	2	G	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≤ 50 gallons
	5(a)	2	H	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≤ 50 gallons

(continued)

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

(b) Each APRM channel provides inputs to both trip systems.

(d) During instrument calibrations, if the As Found channel setpoint is conservative with respect to the Allowable Value but outside its acceptable As Found band as defined by its associated Surveillance Requirement procedure, then there shall be an initial determination to ensure confidence that the channel can perform as required before returning the channel to service in accordance with the Surveillance. If the As Found instrument channel setpoint is not conservative with respect to the Allowable Value, the channel shall be declared inoperable. Prior to returning a channel to service, the instrument channel setpoint shall be calibrated to a value that is within the acceptable As Left tolerance of the setpoint; otherwise, the channel shall be declared inoperable.

The nominal Trip Setpoint shall be specified on design output documentation which is incorporated by reference in the Updated Final Safety Analysis Report. The methodology used to determine the nominal Trip Setpoint, the predefined As Found Tolerance, and the As Left Tolerance band, and a listing of the setpoint design output documentation shall be specified in Chapter 7 of the Updated Final Safety Analysis Report.

(f) Following Detect and Suppress Solution - Confirmation Density (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

-----NOTE-----
Single recirculation loop operation is prohibited in the MELLRA+ operating domain.

One recirculation loop may be in operation 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 3.3.1.1, "Reactor Protection System (RPS) Instrumentation," Function 2.b (Average Power Range Monitors Flow Biased Simulated Thermal Power - High), Allowable Value of Table 3.3.1.1-1 is reset for single loop operation.

APPLICABILITY: MODES 1 and 2.

Recirculation Loops Operating
3.4.1

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Requirements of the LCO not met.	A.1 Satisfy the requirements of the LCO.	24 hours
B. Operation in the MELLLA+ operating 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. <u>OR</u> No recirculation loops in operation.	C.1 Be in MODE 3.	12 hours

5.6 Reporting Requirements (continued)

5.6.4 (Deleted).

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 APLHGRs for Specification 3.2.1;
 - (2) The LHGR for Specification 3.2.3;
 - (3) The MCPR Operating Limits for Specification 3.2.2;
 - (4) The Manual Backup Stability Protection (BSP) Scram Region (Region I), the Manual BSP Controlled Entry Region (Region II), the modified APRM Flow Biased 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 RBM setpoints and applicable reactor thermal power ranges for each of the setpoints for Specification 3.3.2.1, Table 3.3.2.1-1.
- b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:
 - 1. NEDE-24011-P-A, Revision 16, General Electric Standard Application for Reactor Fuel, October 2007.
 - 2. XN-NF-81-58(P)(A) Revision 2 and Supplements 1 and 2, RODEX2 Fuel Rod Thermal-Mechanical Response Evaluation Model, Exxon Nuclear Company, March 1984.

(continued)

5.6 Reporting Requirements (continued)

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

18. EMF-CC-074(P)(A), Volume 4, Revision 0, BWR Stability Analysis: Assessment of STAIF with Input from MICROBURN-B2, Siemens Power Corporation, August 2000.
19. BAW-10255(P)(A), Revision 2, Cycle-Specific DIVOM Methodology Using the RAMONA5-FA Code, AREVA NP, May 2008.
20. BAW-10247PA Revision 0, Realistic Thermal-Mechanical Fuel Rod Methodology for Boiling Water Reactors, AREVA NP, February 2008.
21. ANP-10298PA Revision 0, ACE/ATRIUM 10XM Critical Power Correlation, AREVA NP, March 2010.
22. ANP-3140(P) Revision 0, Browns Ferry Units 1, 2, and 3 Improved K-factor Model for ACE/ATRIUM 10XM Critical Power Correlation, AREVA NP, August 2012.
23. NEDC-33075P-A, GE Hitachi Boiling Water Reactor Detect and Suppress Solution – Confirmation Density, Revision 8, November 2013.

(continued)

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

- 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 PAM Report

When a report is required by Condition B or G 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 OPRM Report

When an OPRM report is required by Condition I of LCO 3.3.1.1, "RPS Instrumentation," the 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.



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

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-260

BROWNS FERRY NUCLEAR PLANT, UNIT 2

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 333
Renewed License No. DPR-52

1. The Nuclear Regulatory Commission (the Commission) has found that:
 - A. The application for amendment by the Tennessee Valley Authority (the licensee) dated February 23, 2018, as supplemented by letters dated March 7, 2018; July 23, 2018; December 13, 2018; December 14, 2018; January 16, 2019 (two letters), January 25, 2019; March 13, 2019; April 24, 2019; April 29, 2019; November 25, 2019; and December 19, 2019, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;
 - B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;
 - C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;
 - D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and
 - E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.
2. Accordingly, the license is amended by changes to the Renewed Facility Operating License (RFOL) and Technical Specifications as indicated in the attachment to this license amendment.

3. This license amendment is effective as of its date of issuance and shall be implemented within 1 year from the date of its approval.

FOR THE NUCLEAR REGULATORY COMMISSION

A handwritten signature in black ink, appearing to read "Undine Shoop", written in a cursive style.

Undine Shoop, Chief
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Renewed Facility Operating
License and Technical Specifications

Date of Issuance: December 26, 2019

ATTACHMENT TO LICENSE AMENDMENT NO. 333

BROWNS FERRY NUCLEAR PLANT, UNIT 2

RENEWED FACILITY OPERATING LICENSE NO. DPR-52

DOCKET NO. 50-260

Replace the following pages of Renewed Facility Operating License No. DPR-52 with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

REMOVE

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

REMOVE

3.3-3

3.3-6
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5.0-24b
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INSERT

3.3-3
3.3-3a
3.3-6
3.3-7
3.3-8
3.4-1
3.4-2
5.0-24
5.0-24b
5.0-25

sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

- (4) Pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess, and use in amounts as required any byproduct, source, or special nuclear material without restriction to chemical or physical form for sample analysis or equipment and instrument calibration or associated with radioactive apparatus or components;
- (5) Pursuant to the Act and 10 CFR Parts 30 and 70, to possess but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

C. This renewed operating 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 3952 megawatts thermal.

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 333, are hereby incorporated in the renewed operating license. The licensee shall operate the facility in accordance with the Technical Specifications.

For Surveillance Requirements (SRs) that are new in Amendment 253 to Facility Operating License DPR-52, the first performance is due at the end of the first surveillance interval that begins at implementation of the Amendment 253. For SRs that existed prior to Amendment 253, including SRs with modified acceptance criteria and SRs whose frequency of performance is being extended, the first performance is due at the end of the first surveillance interval that begins on the date the surveillance was last performed prior to implementation of Amendment 253.

- 3) The licensee is authorized to relocate certain requirements included in Appendix A and the former Appendix B to licensee-controlled documents. Implementation of this amendment shall include the relocation of these requirements to the appropriate documents, as described in the licensee's

- (h) The results of the visual inspections of the steam dryer shall be submitted to the NRC staff in a report in accordance with 10 CFR 50.4. The report shall be submitted to the NRC within 90 days following startup from each of the first two respective refueling outages.
- (i) Within 6 months following completion of the second refueling outage, after the implementation of the EPU, the licensee shall submit a long-term steam dryer inspection plan based on industry operating experience along with the baseline inspection results.

The license condition described above shall expire: (1) upon satisfaction of the requirements in items (g) and (h), provided that a visual inspection of the steam dryer does not reveal any new unacceptable flaw(s) or unacceptable flaw growth that is due to fatigue, and (2) upon satisfaction of the requirements specified in item (i).

(19) Neutron Absorber Monitoring Program

The licensee shall, at least once every ten years, withdraw a neutron absorber coupon from the spent fuel pool and perform Boron-10 (B-10) areal density measurement on the coupon. Based on the results of the B-10 areal density measurement, the licensee shall perform any technical evaluations that may be necessary and take appropriate actions using relevant regulatory and licensing processes.

(20) Radiological Consequences Analyses Using Alternative Source Terms

TVA shall perform facility and licensing basis modifications to resolve the non-conforming/degraded condition associated with the Alternate Leakage Treatment pathway such that the current licensing basis dose calculations (approved in License Amendment Nos. 251/282 (Unit 1), 290/308 (Unit 2) and 249/267 (Unit 3)) would remain valid. These facility and licensing basis modifications shall be completed prior to initial power ascension above 3458 MWt.

- (21) Prior to extending the frequency for the Integral Leakage Rate Testing described in TS 5.5.12, the licensee shall implement the modifications, that are modeled in the Fire PRA and described in Table S-2, Plant Modifications Committed, of Tennessee Valley Authority letter CNL-18-100, dated October 18, 2018; as supplemented by letter CNL-19-027, dated February 13, 2019.

(22) Maximum Extended Load Line Limit Analysis Plus (MELLLA+) Special Consideration

The licensee shall not operate the facility within the MELLLA+ operating domain with more than a 10°F reduction in feedwater temperature below the design feedwater temperature.

(23) Maximum Extended Load Line Limit Analysis Plus (MELLLA+) Implementation

Prior to the first implementation of MELLLA+, TVA shall perform reload safety analyses using codes that have been corrected for the errors described in TVA letter CNL-19-125, dated December 19, 2019.

- D. The UFSAR supplement, as revised, submitted pursuant to 10 CFR 54.21(d), shall be included in the next scheduled update to the UFSAR required by 10 CFR 50.71(e)(4) following the issuance of this renewed operating license. Until that update is complete, TVA may make changes to the programs and activities described in the supplement without prior Commission approval, provided that TVA evaluates such changes pursuant to the criteria set forth in 10 CFR 50.59 and otherwise complies with the requirements in that section.
- E. The UFSAR supplement, as revised, describes certain future activities to be completed prior to the period of extended operation. TVA shall complete these activities no later than June 28, 2014, and shall notify the NRC in writing when implementation of these activities is complete and can be verified by NRC inspection.
- F. All capsules in the reactor vessel that are removed and tested must meet the test procedures and reporting requirements of the most recent NRC-approved version of the Boiling Water Reactor Vessels and Internals Project (BWRVIP) Integrated Surveillance Program (ISP) appropriate for the configuration of the specimens in the capsule. Any changes to the BWRVIP ISP capsule withdrawal schedule, including spare capsules, must be approved by the NRC prior to implementation. All capsules placed in storage

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
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 Backup Stability Protection (BSP) Regions defined in the COLR.	Immediately
	<u>AND</u>	
	I.2 Implement the Automated BSP Scram Region using the modified APRM Flow Biased Simulated Thermal Power-High scram setpoints defined in the COLR.	12 hours
	<u>AND</u>	
	I.3 Initiate action to submit an OPRM report 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
	<u>AND</u>	
	J.2 Reduce operation to below the BSP Boundary defined in the COLR.	12 hours
	<u>AND</u>	
	J.3 NOTE LCO 3.0.4 is not applicable.	
	Restore required channel to OPERABLE status.	120 days
K. Required Actions and associated Completion Time of Condition J not met.	K.1 Reduce THERMAL POWER to < 18% RTP.	4 hours

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.1.10	Perform CHANNEL CALIBRATION.	184 days
SR 3.3.1.1.11	(Deleted).	
SR 3.3.1.1.12	Perform CHANNEL FUNCTIONAL TEST.	24 months
SR 3.3.1.1.13	<p>-----NOTE----- Neutron detectors are excluded.</p> <p>Perform CHANNEL CALIBRATION.</p>	24 months
SR 3.3.1.1.14	Perform LOGIC SYSTEM FUNCTIONAL TEST.	24 months
SR 3.3.1.1.15	Verify Turbine Stop Valve - Closure and Turbine Control Valve Fast Closure, Trip Oil Pressure - Low Functions are not bypassed when THERMAL POWER is $\geq 26\%$ RTP.	24 months
SR 3.3.1.1.16	<p>-----NOTE----- For Function 2.a, not required to be performed when entering MODE 2 from MODE 1 until 12 hours after entering MODE 2.</p> <p>Perform CHANNEL FUNCTIONAL TEST.</p>	184 days
SR 3.3.1.1.17	(Deleted).	

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 3.3.1.1.1 SR 3.3.1.1.3 SR 3.3.1.1.5 SR 3.3.1.1.6 SR 3.3.1.1.9 SR 3.3.1.1.14	≤ 120/125 divisions of full scale
	5(a)	3	H	SR 3.3.1.1.1 SR 3.3.1.1.4 SR 3.3.1.1.9 SR 3.3.1.1.14	≤ 120/125 divisions of full scale
b. Inop	2	3	G	SR 3.3.1.1.3 SR 3.3.1.1.14	NA
	5(a)	3	H	SR 3.3.1.1.4 SR 3.3.1.1.14	NA
2. Average Power Range Monitors					
a. Neutron Flux - High, (Setdown)	2	3(b)	G	SR 3.3.1.1.1 SR 3.3.1.1.6 SR 3.3.1.1.7 SR 3.3.1.1.13 SR 3.3.1.1.16	≤ 13% RTP
b. Flow Biased Simulated Thermal Power - High	1	3(b)	F	SR 3.3.1.1.1 SR 3.3.1.1.2 SR 3.3.1.1.7 SR 3.3.1.1.13 SR 3.3.1.1.16	≤ 0.61 W + 68.3% RTP and ≤ 120% RTP(c)(e)
c. Neutron Flux - High	1	3(b)	F	SR 3.3.1.1.1 SR 3.3.1.1.2 SR 3.3.1.1.7 SR 3.3.1.1.13 SR 3.3.1.1.16	≤ 120% RTP
(continued)					

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

(b) Each APRM channel provides inputs to both trip systems.

(c) [0.55 W + 65.5% - 0.55 Δ W] RTP when reset for single loop operation per LCO 3.4.1, "Recirculation Loops Operating."

(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)					
d. Inop	1,2	3(b)	G	SR 3.3.1.1.16	NA
e. 2-Out-Of-4 Voter	1,2	2	G	SR 3.3.1.1.1 SR 3.3.1.1.14 SR 3.3.1.1.16	NA
f. OPRM Upscale	≥18% ^(f)	3(b)	I	SR 3.3.1.1.1 SR 3.3.1.1.7 SR 3.3.1.1.13 SR 3.3.1.1.16	NA
3. Reactor Vessel Steam Dome Pressure - High ^(d)	1,2	2	G	SR 3.3.1.1.1 SR 3.3.1.1.8 SR 3.3.1.1.10 SR 3.3.1.1.14	≤ 1090 psig
4. Reactor Vessel Water Level - Low, Level 3 ^(d)	1,2	2	G	SR 3.3.1.1.1 SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≥ 528 inches above vessel zero
5. Main Steam Isolation Valve - Closure	1	8	F	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≤ 10% closed
6. Drywell Pressure - High	1,2	2	G	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≤ 2.5 psig
7. Scram Discharge Volume Water Level - High					
a. Resistance Temperature Detector	1,2	2	G	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≤ 50 gallons
	5(a)	2	H	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≤ 50 gallons

(continued)

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

(b) Each APRM channel provides inputs to both trip systems.

(d) During instrument calibrations, if the As Found channel setpoint is conservative with respect to the Allowable Value but outside its acceptable As Found band as defined by its associated Surveillance Requirement procedure, then there shall be an initial determination to ensure confidence that the channel can perform as required before returning the channel to service in accordance with the Surveillance. If the As Found instrument channel setpoint is not conservative with respect to the Allowable Value, the channel shall be declared inoperable. Prior to returning a channel to service, the instrument channel setpoint shall be calibrated to a value that is within the acceptable As Left tolerance of the setpoint; otherwise, the channel shall be declared inoperable.

The nominal Trip Setpoint shall be specified on design output documentation which is incorporated by reference in the Updated Final Safety Analysis Report. The methodology used to determine the nominal Trip Setpoint, the predefined As Found Tolerance, and the As Left Tolerance band, and a listing of the setpoint design output documentation shall be specified in Chapter 7 of the Updated Final Safety Analysis Report.

(f) Following Detect and Suppress Solution – Confirmation Density (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

-----NOTE-----

Single recirculation loop operation is prohibited in the MELLLA+ operating domain.

One recirculation loop may be in operation 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 3.3.1.1, "Reactor Protection System (RPS) Instrumentation," Function 2.b (Average Power Range Monitors Flow Biased Simulated Thermal Power - High), Allowable Value of Table 3.3.1.1-1 is reset for single loop operation;

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Requirements of the LCO not met.	A.1 Satisfy the requirements of the LCO.	24 hours
B. Operation in the MELLLA+ operating 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. <u>OR</u> No recirculation loops in operation.	C.1 Be in MODE 3.	12 hours

5.6 Reporting Requirements (continued)

5.6.4 (Deleted).

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 APLHGRs for Specification 3.2.1;
 - (2) The LHGR for Specification 3.2.3;
 - (3) The MCPR Operating Limits for Specification 3.2.2;
 - (4) The Manual Backup Stability Protection (BSP) Scram Region (Region I), the Manual BSP Controlled Entry Region (Region II), the modified APRM Flow Biased 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 RBM setpoints and applicable reactor thermal power ranges for each of the setpoints for Specification 3.3.2.1, Table 3.3.2.1-1.
- b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:
 - 1. XN-NF-81-58(P)(A) Revision 2 and Supplements 1 and 2, RODEX2 Fuel Rod Thermal-Mechanical Response Evaluation Model, Exxon Nuclear Company, March 1984.
 - 2. XN-NF-85-67(P)(A) Revision 1, Generic Mechanical Design for Exxon Nuclear Jet Pump BWR Reload Fuel, Exxon Nuclear Company, September 1986.
 - 3. EMF-85-74(P) Revision 0 Supplement 1(P)(A) and Supplement 2 (P)(A), RODEX2A (BWR) Fuel Rod Thermal-Mechanical Evaluation Model, Siemens Power Corporation, February 1998.

(continued)

5.6 Reporting Requirements (continued)

14. EMF-2245(P)(A) Revision 0, Application of Siemens Power Corporation's Critical Power Correlations to Co-Resident Fuel, Siemens Power Corporation, August 2000.
15. EMF-2361(P)(A) Revision 0, EXEM BWR-2000 ECCS Evaluation Model, Framatome ANP Inc., May 2001 as supplemented by the site-specific approval in NRC safety evaluation dated February 15, 2013, and July 31, 2014.
16. EMF-2292(P)(A) Revision 0, ATRIUM™-10: Appendix K Spray Heat Transfer Coefficients, Siemens Power Corporation, September 2000.
17. EMF-CC-074(P)(A), Volume 4, Revision 0, BWR Stability Analysis: Assessment of STAIF with Input from MICROBURN-B2, Siemens Power Corporation, August 2000.
18. BAW-10255(P)(A), Revision 2, Cycle-Specific DIVOM Methodology Using the RAMONA5-FA Code, AREVA NP, May 2008.
19. BAW-10247PA Revision 0, Realistic Thermal-Mechanical Fuel Rod Methodology for Boiling Water Reactors, AREVA NP, February 2008.
20. ANP-10298PA Revision 0, ACE/ATRIUM 10XM Critical Power Correlation, AREVA NP, March 2010.
21. ANP-3140(P) Revision 0, Browns Ferry Units 1, 2, and 3 Improved K-factor Model for ACE/ATRIUM 10XM Critical Power Correlation, AREVA NP, August 2012.
22. NEDC-33075P-A, GE Hitachi Boiling Water Reactor Detect and Suppress Solution – Confirmation Density, Revision 8, November 2013.

(continued)

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

- 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 PAM Report

When a report is required by Condition B or G 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 OPRM Report

When an OPRM report is required by Condition I of LCO 3.3.1.1, "RPS Instrumentation," the 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.



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

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-296

BROWNS FERRY NUCLEAR PLANT, UNIT 3


AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 293
Renewed License No. DPR-68

1. The Nuclear Regulatory Commission (the Commission) has found that:
 - A. The application for amendment by the Tennessee Valley Authority (the licensee) dated February 23, 2018, as supplemented by letters dated March 7, 2018; July 23, 2018; December 13, 2018; December 14, 2018; January 16, 2019 (2 letters), January 25, 2019; March 13, 2019; April 24, 2019; April 29, 2019; November 25, 2019; and December 19, 2019, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;
 - B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;
 - C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;
 - D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and
 - E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.
2. Accordingly, the license is amended by changes to the Renewed Facility Operating License (RFOL) and Technical Specifications as indicated in the attachment to this license amendment.

3. This license amendment is effective as of its date of issuance and shall be implemented within 1 year from the date of its approval.

FOR THE NUCLEAR REGULATORY COMMISSION

A handwritten signature in black ink, appearing to read 'Undine Shoop', is written over the typed name.

Undine Shoop, Chief
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Renewed Facility Operating
License and Technical Specifications

Date of Issuance: December 26, 2019

ATTACHMENT TO LICENSE AMENDMENT NO. 293

BROWNS FERRY NUCLEAR PLANT, UNIT 3

RENEWED FACILITY OPERATING LICENSE NO. DPR-68

DOCKET NO. 50-296

Replace the following pages of Renewed Facility Operating License No. DPR-68 with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

REMOVE

3
6f

INSERT

3
6f

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

REMOVE

3.3-3

3.3-6
3.3-7
3.3-8
3.4-1
3.4-2
5.0-24
5.0-24b
5.0-25

INSERT

3.3-3
3.3-3a
3.3-6
3.3-7
3.3-8
3.4-1
3.4-2
5.0-24
5.0-24b
5.0-25

- (3) Pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess, and use at any time any byproduct, source, and special nuclear material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;
- (4) Pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess, and use in amounts as required any byproduct, source, or special nuclear material without restriction to chemical or physical form for sample analysis or equipment and instrument calibration or associated with radioactive apparatus or components;
- (5) Pursuant to the Act and 10 CFR Parts 30 and 70, to possess but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

C. This renewed operating 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 3952 megawatts thermal.

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 293, are hereby incorporated in the renewed operating license. The licensee shall operate the facility in accordance with the Technical Specifications.

For Surveillance Requirements (SRs) that are new in Amendment 212 to Facility Operating License DPR-68, the first performance is due at the end of the first surveillance interval that begins at implementation of the Amendment 212. For SRs that existed prior to Amendment 212, including SRs with modified acceptance criteria and SRs whose frequency of performance is being extended, the first performance is due at the end of the first surveillance interval that begins on the date the surveillance was last performed prior to implementation of Amendment 212.

(16) Radiological Consequences Analyses Using Alternative Source Terms

TVA shall perform facility and licensing basis modifications to resolve the non-conforming/degraded condition associated with the Alternate Leakage Treatment pathway such that the current licensing basis dose calculations (approved in License Amendment Nos. 251/282 (Unit 1), 290/308 (Unit 2) and 249/267 (Unit 3)) would remain valid. These facility and licensing basis modifications shall be complete prior to initial power ascension above 3458.

(17) Prior to extending the frequency for the Integral Leakage Rate Testing described in TS 5.5.12, the licensee shall implement the modifications, that are modeled in the Fire PRA and described in Table S-2, Plant Modifications Committed, of Tennessee Valley Authority letter CNL-18-100, dated October 18, 2018; as supplemented by letter CNL-19-027, dated February 13, 2019.

(18) Maximum Extended Load Line Limit Analysis Plus (MELLLA+) Special Consideration

The licensee shall not operate the facility within the MELLLA+ operating domain more than a 10°F reduction in feedwater temperature below the design feedwater temperature.

(19) Maximum Extended Load Line Limit Analysis Plus (MELLLA+) Implementation

Prior to the first implementation of MELLLA+, TVA shall perform reload safety analyses using codes that have been corrected for the errors described in TVA letter CNL-19-125, dated December 19, 2019.

- D. The UFSAR supplement, as revised, submitted pursuant to 10 CFR 54.21(d), shall be included in the next scheduled update to the UFSAR required by 10 CFR 50.71(e)(4) following the issuance of this renewed operating license. Until that update is complete, TVA may make changes to the programs and activities described in the supplement without prior Commission approval, provided that TVA evaluates such changes pursuant to the criteria set forth in 10 CFR 50.59 and otherwise complies with the requirements in that section.
- E. The UFSAR supplement, as revised, describes certain future activities to be completed prior to the period of extended operation. TVA shall complete these activities no later than July 2, 2016, and shall notify the NRC in writing when implementation of these activities is complete and can be verified by NRC inspection.

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
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 Backup Stability Protection (BSP) Regions defined in the COLR.	Immediately
	<u>AND</u>	
	I.2 Implement the Automated BSP Scram Region using the modified APRM Flow Biased Simulated Thermal Power-High scram setpoints defined in the COLR.	12 hours
	<u>AND</u>	
	I.3 Initiate action to submit an OPRM report 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
	<u>AND</u>	
	J.2 Reduce operation to below the BSP Boundary defined in the COLR.	12 hours
	<u>AND</u>	
	J.3 <u>NOTE</u> LCO 3.0.4 is not applicable. Restore required channel to OPERABLE status.	120 days
K. Required Actions and associated Completion Time of Condition J not met.	K.1 Reduce THERMAL POWER to < 18% RTP.	4 hours

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.1.10	Perform CHANNEL CALIBRATION.	184 days
SR 3.3.1.1.11	(Deleted)	
SR 3.3.1.1.12	Perform CHANNEL FUNCTIONAL TEST.	24 months
SR 3.3.1.1.13	-----NOTE----- Neutron detectors are excluded. -----	
	Perform CHANNEL CALIBRATION.	24 months
SR 3.3.1.1.14	Perform LOGIC SYSTEM FUNCTIONAL TEST.	24 months
SR 3.3.1.1.15	Verify Turbine Stop Valve - Closure and Turbine Control Valve Fast Closure, Trip Oil Pressure - Low Functions are not bypassed when THERMAL POWER is $\geq 26\%$ RTP.	24 months
SR 3.3.1.1.16	-----NOTE----- For Function 2.a, not required to be performed when entering MODE 2 from MODE 1 until 12 hours after entering MODE 2. -----	
	Perform CHANNEL FUNCTIONAL TEST.	184 days
SR 3.3.1.1.17	(Deleted)	

RPS Instrumentation
3.3.1.1

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 3.3.1.1.1 SR 3.3.1.1.3 SR 3.3.1.1.5 SR 3.3.1.1.6 SR 3.3.1.1.9 SR 3.3.1.1.14	≤ 120/125 divisions of full scale
	5(a)	3	H	SR 3.3.1.1.1 SR 3.3.1.1.4 SR 3.3.1.1.9 SR 3.3.1.1.14	≤ 120/125 divisions of full scale
b. Inop	2	3	G	SR 3.3.1.1.3 SR 3.3.1.1.14	NA
	5(a)	3	H	SR 3.3.1.1.4 SR 3.3.1.1.14	NA
2. Average Power Range Monitors					
a. Neutron Flux - High, (Setdown)	2	3(b)	G	SR 3.3.1.1.1 SR 3.3.1.1.6 SR 3.3.1.1.7 SR 3.3.1.1.13 SR 3.3.1.1.16	≤ 13% RTP
b. Flow Biased Simulated Thermal Power - High	1	3(b)	F	SR 3.3.1.1.1 SR 3.3.1.1.2 SR 3.3.1.1.7 SR 3.3.1.1.13 SR 3.3.1.1.16	≤ 0.61 W + 68.3% RTP and ≤ 120% RTP(c)(e)
c. Neutron Flux - High	1	3(b)	F	SR 3.3.1.1.1 SR 3.3.1.1.2 SR 3.3.1.1.7 SR 3.3.1.1.13 SR 3.3.1.1.16	≤ 120% RTP
(continued)					

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

(b) Each APRM channel provides inputs to both trip systems.

(c) [0.55 W + 65.5% - 0.55 Δ W] RTP when reset for single loop operation per LCO 3.4.1, "Recirculation Loops Operating."

(e) With OPRM Upscale (Function 2.f) inoperable, the Automated BSP Scram Region setpoints are implemented in accordance with Action I of this Specification.

RPS Instrumentation
3.3.1.1

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)					
d. Inop	1,2	3(b)	G	SR 3.3.1.1.16	NA
e. 2-Out-Of-4 Voter	1,2	2	G	SR 3.3.1.1.1 SR 3.3.1.1.14 SR 3.3.1.1.16	NA
f. OPRM Upscale	≥18% ^(f)	3(b)	I	SR 3.3.1.1.1 SR 3.3.1.1.7 SR 3.3.1.1.13 SR 3.3.1.1.16	NA
3. Reactor Vessel Steam Dome Pressure - High ^(d)	1,2	2	G	SR 3.3.1.1.1 SR 3.3.1.1.8 SR 3.3.1.1.10 SR 3.3.1.1.14	≤ 1090 psig
4. Reactor Vessel Water Level - Low, Level 3 ^(d)	1,2	2	G	SR 3.3.1.1.1 SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≥ 528 inches above vessel zero
5. Main Steam Isolation Valve - Closure	1	8	F	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≤ 10% closed
6. Drywell Pressure - High	1,2	2	G	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≤ 2.5 psig
7. Scram Discharge Volume Water Level - High					
a. Resistance Temperature Detector	1,2	2	G	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≤ 50 gallons
	5(a)	2	H	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≤ 50 gallons

(continued)

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

(b) Each APRM channel provides inputs to both trip systems.

(d) During instrument calibrations, if the As Found channel setpoint is conservative with respect to the Allowable Value but outside its acceptable As Found band as defined by its associated Surveillance Requirement procedure, then there shall be an initial determination to ensure confidence that the channel can perform as required before returning the channel to service in accordance with the Surveillance. If the As Found instrument channel setpoint is not conservative with respect to the Allowable Value, the channel shall be declared inoperable. Prior to returning a channel to service, the instrument channel setpoint shall be calibrated to a value that is within the acceptable As Left tolerance of the setpoint; otherwise, the channel shall be declared inoperable. The nominal Trip Setpoint shall be specified on design output documentation which is incorporated by reference in the Updated Final Safety Analysis Report. The methodology used to determine the nominal Trip Setpoint, the predefined As Found Tolerance, and the As Left Tolerance band, and a listing of the setpoint design output documentation shall be specified in Chapter 7 of the Updated Final Safety Analysis Report.

(f) Following Detect and Suppress Solution - Confirmation Density (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

-----NOTE-----
Single recirculation loop operation is prohibited in the MELLLA+
operating domain.

One recirculation loop may be in operation 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 3.3.1.1, "Reactor Protection System (RPS) Instrumentation," Function 2.b (Average Power Range Monitors Flow Biased Simulated Thermal Power - High), Allowable Value of Table 3.3.1.1-1 is reset for single loop operation;

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Requirements of the LCO not met.	A.1 Satisfy the requirements of the LCO.	24 hours
B. Operation in the MELLLA+ operating 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. <u>OR</u> No recirculation loops in operation.	C.1 Be in MODE 3.	12 hours

5.6 Reporting Requirements (continued)

5.6.4 (Deleted).

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 APLHGRs for Specification 3.2.1;
 - (2) The LHGR for Specification 3.2.3;
 - (3) The MCPR Operating Limits for Specification 3.2.2;
 - (4) The Manual Backup Stability Protection (BSP) Scram Region (Region I), the Manual BSP Controlled Entry Region (Region II), the modified APRM Flow Biased 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 RBM setpoints and applicable reactor thermal power ranges for each of the setpoints for Specification 3.3.2.1, Table 3.3.2.1-1.
- b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:
 - 1. XN-NF-81-58(P)(A) Revision 2 and Supplements 1 and 2, RODEX2 Fuel Rod Thermal-Mechanical Response Evaluation Model, Exxon Nuclear Company, March 1984.
 - 2. XN-NF-85-67(P)(A) Revision 1, Generic Mechanical Design for Exxon Nuclear Jet Pump BWR Reload Fuel, Exxon Nuclear Company, September 1986.
 - 3. EMF-85-74(P) Revision 0 Supplement 1(P)(A) and Supplement 2 (P)(A), RODEX2A (BWR) Fuel Rod Thermal-Mechanical Evaluation Model, Siemens Power Corporation, February 1998.

(continued)

5.6 Reporting Requirements (continued)

12. ANF-1358(P)(A) Revision 3, The Loss of Feedwater Heating Transient in Boiling Water Reactors, Framatome ANP, September 2005.
13. EMF-2209(P)(A) Revision 3, SPCB Critical Power Correlation, AREVA NP, September 2009.
14. EMF-2245(P)(A) Revision 0, Application of Siemens Power Corporation's Critical Power Correlations to Co-Resident Fuel, Siemens Power Corporation, August 2000.
15. EMF-2361(P)(A) Revision 0, EXEM BWR-2000 ECCS Evaluation Model, Framatome ANP Inc., May 2001 as supplemented by the site-specific approval in NRC safety evaluations dated February 15, 2013, and July 31, 2014.
16. EMF-2292(P)(A) Revision 0, ATRIUM™-10: Appendix K Spray Heat Transfer Coefficients, Siemens Power Corporation, September 2000.
17. EMF-CC-074(P)(A), Volume 4, Revision 0, BWR Stability Analysis: Assessment of STAIF with Input from MICROBURN-B2, Siemens Power Corporation, August 2000.
18. BAW-10255(P)(A), Revision 2, Cycle-Specific DIVOM Methodology Using the RAMONA5-FA Code, AREVA NP, May 2008.
19. BAW-10247PA Revision 0, Realistic Thermal-Mechanical Fuel Rod Methodology for Boiling Water Reactors, AREVA NP, February 2008.
20. ANP-10298PA Revision 0, ACE/ATRIUM 10XM Critical Power Correlation, AREVA NP, March 2010.
21. ANP-3140(P) Revision 0, Browns Ferry Units 1, 2, and 3 Improved K-factor Model for ACE/ATRIUM 10XM Critical Power Correlation, AREVA NP, August 2012.
22. NEDC-33075P-A, GE Hitachi Boiling Water Reactor Detect and Suppress Solution – Confirmation Density, Revision 8, November 2013.

(continued)

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

- 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 PAM Report

When a report is required by Condition B or G 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 OPRM Report

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

ENCLOSURE 5
(NONPROPRIETARY)

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION
RELATED TO AMENDMENT NO. 310
TO RENEWED FACILITY OPERATING LICENSE NO. DPR-33
AMENDMENT NO. 333 TO RENEWED FACILITY OPERATING LICENSE NO. DPR-52
AND
AMENDMENT NO. 293 TO RENEWED FACILITY OPERATING LICENSE NO. DPR-68
TENNESSEE VALLEY AUTHORITY
BROWNS FERRY NUCLEAR PLANT, UNITS 1, 2, AND 3
DOCKET NOS. 50-259, 50-260, AND 50-296

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 with boldface double brackets as shown here [[]].

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

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TENNESSEE VALLEY AUTHORITY

BROWNS FERRY NUCLEAR PLANT, UNITS 1, 2, AND 3

DOCKET NOS. 50-259, 50-260, AND 50-296

1.0 INTRODUCTION

By letter to the U.S. Nuclear Regulatory Commission (NRC or the Commission) dated February 23, 2018 (Reference 1), as supplemented by letters dated March 7, 2018 (Reference 2); July 23, 2018 (Reference 3); December 13, 2018 (Reference 4); December 14, 2018 (Reference 5); January 16, 2019 (Reference 6); January 16, 2019 (Reference 7); January 25, 2019 (Reference 8); March 13, 2019 (Reference 9); April 24, 2019 (Reference 10); April 29, 2019 (Reference 11); November 25, 2019 (Reference 12); and December 19, 2019 (Reference 13), Tennessee Valley Authority (TVA or the licensee) submitted a license amendment request (LAR) for Browns Ferry Nuclear Plant (Browns Ferry or BFN), Units 1, 2 and 3. The proposed amendments would allow operation in the expanded Maximum Extended Load Line Limit Analysis Plus (MELLLA+) operating domain and use of the Detect and Suppress Solution - Confirmation Density (DSS-CD) stability solution.

The supplements dated July 23, 2018; December 13, 2018; December 14, 2018; January 16, 2019 (two letters), January 25, 2019; March 13, 2019; April 24, 2019; April 29, 2019; November 25, 2019; and December 19, 2019, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the NRC staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on June 5, 2018 (83 FR 26116).

2.0 REGULATORY EVALUATION

2.1 Background

General Information

BFN Units 1, 2, and 3 are boiling-water reactor (BWR) plants of the BWR/4 design with Mark-1 containments located in Limestone County, Alabama, approximately 30 miles west of Huntsville, Alabama. The site contains approximately 840 acres and is located on the north shore of Wheeler Lake at Tennessee River Mile 294.

The construction permits for BFN Units 1, 2, and 3 were issued by the Atomic Energy Commission (AEC) on May 10, 1967, for Units 1 and 2, and July 31, 1968, for Unit 3. The NRC originally licensed BFN Unit 1 on December 20, 1973; BFN Unit 2 on June 28, 1974; and BFN Unit 3 on August 18, 1976, for operation at the original licensed power level of 3,293 megawatts thermal (MWt).

By amendment dated March 6, 2007 (Reference 14), for BFN Unit 1, and amendments dated September 8, 1998 (Reference 15) for BFN Units 2 and 3, the NRC-approved power uprates of 5 percent, allowing the plant to be operated at the licensed power level of 3,458 MWt. By amendments dated August 14, 2017 (Reference 16), the NRC approved an extended power uprated (EPU) allowing an increase in the licensed power level to 3,952 MWt, which resulted in an increase of approximately 20 percent over the original licensed power level for BFN Units 1, 2, and 3.

Proposed MELLLA+ Operation

As discussed in Section 2.1 of the enclosure to the licensee's application (Reference 1), operation of BWRs requires that the reactivity balance be maintained to accommodate fuel burnup. BWR operators have typically two methods to maintain this reactivity balance, which include (1) control rod movements and (2) reactor recirculation core flow (CF) adjustments. Because of strong void reactivity feedback and its distributed effect through the reactor core, recirculation flow adjustments are the preferred reactivity control method. Operating at low CF conditions at rated power level also increases the fuel capacity factor through spectral shift. In addition, an increased flow region compensates for reactivity reduction due to fuel depletion during the operating cycle.

EPUs are implemented by extending the Maximum Extended Load Line Limit Analysis (MELLLA) operating domain up to EPU rated thermal power (RTP) levels. However, this reduces the available CF flow window at these levels. In addition, the increased core pressure drop limits recirculation flow capability. Consequently, EPU plants generally operate with a greatly reduced CF window and compensate for reactivity loss with control rod movement.

MELLLA+ increases the operating boundary to permit BFN operation at a current licensed thermal power (CLTP) of 3,952 MWt with a CF as low as 85 percent, thus adding a 14 percent flow-control window. By operating in the MELLLA+ domain, a significantly lower number of control rod manipulations is required than is currently required in the present operating domain. The licensee stated that lessening the number of rod manipulations represents a significant improvement in operating flexibility, as well as providing safer plant operation. Specifically, this minimizes the likelihood of fuel failures and reduces the likelihood of events initiated by reactor maneuvers required to achieve an operating condition where control rods can be withdrawn.

The licensee further stated that the MELLLA+ core operating domain expansion requires minor plant system modifications. It involves changes to the operating power/CF map, changes to a small number of instrument setpoints included in the proposed technical specification (TS) changes, and application of the DSS-CD stability solution.

2.2 The Licensee's Approach

Attachment 6, "NEDO-33877, Safety Analysis Report for Browns Ferry Nuclear Plant Units 1, 2, and 3 Maximum Extended Load Line Limit Plus," of the licensee's letter dated February 23, 2018 (Reference 1), contains the MELLLA+ Safety Analysis Report (M+SAR) for the BFN MELLLA+ LAR. The M+SAR provided the technical bases and the underlying safety analyses and evaluations performed specifically for BFN. The BFN MELLLA+ Safety Analysis Report (BFN M+SAR) for MELLLA+ follows the guidelines contained in the generic MELLLA+ Licensing Topical Report (M+LTR), NEDC-33006P-A (Reference 17).

The licensee stated in the enclosure of its LAR that the BFN MELLLA+ LAR is based on the following General Electric – Hitachi (GEH) Licensing Topical Reports (LTRs):

- NEDC-33006P-A, Revision 3, Maximum Extended Load Line Limit Analysis Plus (M+LTR) (Reference 17)
- NEDC-33075P-A, Revision 8, General Electric Boiling Water Reactor Detect and Suppress Solution - Confirmation Density (DSS-CD LTR) (Reference 18)
- NEDC-33173P-A, Revision 4, Applicability of General Electric (GE) Methods to Expanded Operating Domains (Methods LTR) (Reference 19)

BFN currently operates with AREVA fuel. It should be noted that AREVA nuclear reactor operations have been recently renamed "Framatome." For the purposes of this safety evaluation (SE), "AREVA" should be considered synonymous with "Framatome."

Attachment 7 to the LAR (Reference 1) contains the proprietary version of AREVA Report ANP-3551P, AREVA MELLLA+ Safety Analysis Report for BFN (AMSAR), and Attachment 8 provides the nonproprietary version of this report. AMSAR supplements the GEH M+SAR and summarizes the results of fuel-related safety analysis and evaluations performed specifically for BFN to justify the expansion of the CF operating domain and use of the DSS-CD stability solution for AREVA fuel. Attachments 9 through 32 to the LAR support AMSAR by providing the following fuel-related reports.

- ANP-3546, Browns Ferry Units 1, 2, and 3 LOCA Break Spectrum Analysis for ATRIUM 10XM Fuel (EPU MELLLA+)
- ANP-3547, Browns Ferry Units 1, 2, and 3 LOCA-ECCS Analysis MAPLHGR Limits for ATRIUM 10XM Fuel (EPU MELLLA+)
- ANP-3548, Browns Ferry Units 1, 2, and 3 LOCA-ECCS Analysis MAPLHGR Limits for ATRIUM-10 Fuel (EPU MELLLA+)
- ANP-3544 Browns Ferry EPU (120% OLTP) MELLLA+ Equilibrium Fuel Cycle Design

- ANP-3553, Browns Ferry Unit 3 Cycle 19 EPU (120% OLTP) MELLLA+LAR Reference Fuel Cycle Design
- ANP-3552, Browns Ferry Unit 3 Cycle 19 Representative Reload Analysis at EPU MELLLA+
- ANP-3568, Fuel Rod Thermal-Mechanical Evaluation for Browns Ferry Extended Power Uprate/MELLLA+
- ANP-3550, Evaluation of AREVA Fuel Thermal-Hydraulic Performance for Browns Ferry at EPU MELLLA+
- FS1-0029291/2, Browns Ferry Units 1, 2, and 3 EPU MELLLA+ MCPR Safety Limit Analysis with SAFLIM3D Methodology
- ANP-2860 Revision 2, Supplement 3, Browns Ferry Unit 1 – Summary of Responses to Request for Additional Information, Extension for Extended Power Flow Operating Domain
- ANP-3572, Browns Ferry EPU MELLLA+ 10 CFR 50.46c Evaluation
- ANP-3633, Browns Ferry EPU MELLLA+ CRDA Assessment with DG-1327 Criteria

2.3 Method of NRC 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 Review Standard for Extended Power Uprate (RS)-001 (Reference 20), relevant sections of the Standard Review Plan (SRP) (Reference 21), and the findings of the NRC staff's evaluation of the M+LTR (Reference 17).

The M+SAR (Attachment 6 to the LAR (Reference 1)) and AMSAR (Attachment 8 to the LAR (Reference 1)) follow the same structure and content as the M+LTR (Reference 17). The evaluations in the M+SAR and the AMSAR are either "generic" or "plant-specific."

- Generic assessments and plant-specific assessments are described in Sections 1.1.1 and 1.1.2 of the M+SAR, respectively. The generic assessments, as defined in the M+LTR, include generic bounding analyses, impacts that have 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 BFN.
- For the plant-specific reviews, the NRC staff seeks to determine whether the licensee's proposal meets the regulatory criteria, and for evaluations where calculations were necessary, that 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 RS-001, Revision 0. Although the MELLLA+ LAR is not an EPU LAR 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 to the LAR. The technical evaluation of the plant-specific studies

was based on the guidance of review in RS-001. In particular, the following reactor systems topics were reviewed in detail:

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

2.4 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"
 - a. 10 CFR 20.1101, "Radiation Protection Programs," requires that licensees use, to the extent practical, procedures and engineering controls to achieve occupational doses and doses to members of the public that are as low as is reasonably achievable (ALARA).
 - b. 10 CFR 20.1201, "Occupational Dose Limits for Adults," establishes occupational dose limits for adults and requires licensees to control the occupational dose to individual adults.
 - c. 10 CFR 20.1301, "Dose Limits for Individual Members of the Public," establishes dose limits for individual members of the public such that total effective dose equivalent (TEDE) does not exceed 100 mrem in a year. In addition, 10 CFR 20.1301(e) requires compliance with the Environmental Protection Agency's 40 CFR Part 190 dose limits for any member of the public in the general environment (i.e., 25 mrem to the whole body, 75 mrem to the thyroid, and 25 mrem to any other organ).
- 10 CFR 50.36, "Technical specifications," which contains regulatory requirements related to the contents of the TSs.

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 technical specifications 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:

Surveillance requirements are requirements relating to test, calibration, or inspection to assure that the necessary quality of systems and

components is maintained, that facility operation will be within safety limits, and that the limiting conditions of operation will be met.

- 10 CFR 50.36a, "Technical specifications on effluents from nuclear power reactors," 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, but still within the limits specified in 10 CFR 20.1301, and expects that the licensee will exert its best efforts to keep levels of radioactive effluents 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," insofar as it requires that plants be provided with the capability of controlling combustible gas concentrations in the containment atmosphere.
- 10 CFR 50.46, "Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors," which sets standards for the calculation of emergency core cooling system (ECCS) performance and acceptance criteria for that calculated performance.
- 10 CFR 50.62, "Requirements for reduction of risk from anticipated transients without scram (ATWS) events for light-water-cooled nuclear power plants," which requires licensees to provide the means to address an ATWS event, i.e., 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 Criterion (GDC) 20 of Appendix A.
- 10 CFR 50.62(c)(4), which requires that the standby liquid control (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 specified 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.
- 10 CFR Part 50, Appendix G, "Fracture Toughness," specifies fracture toughness requirements for ferritic materials of pressure-retaining components of the reactor coolant pressure boundary of light-water nuclear power reactors to provide adequate margins of safety during any condition of normal operation, including AOOs and system hydrostatic tests to which the pressure boundary may be subjected over its service lifetime.
- 10 CFR Part 50, Appendix I, "Numerical Guides for Design Objectives and Limiting Conditions for Operation to Meet the Criterion 'As Low as is Reasonably Achievable' for Radioactive Material in Light-Water-Cooled Nuclear Power Reactor Effluents," 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, "ECCS Evaluation Models," 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).

The BFN units were designed and constructed based on the proposed General Design Criteria (GDC) published by the AEC in the *Federal Register* (32 FR 10213) on July 11, 1967 (hereafter called "AEC draft GDC"). The AEC published the final rule that added Appendix A to 10 CFR Part 50, "General Design Criteria for Nuclear Power Plants," in the *Federal Register* (36 FR 3255) on February 20, 1971 (hereafter called "final GDC").

Differences between the draft GDC and final GDC included a consolidation from 70 to 64 criteria. As discussed in the NRC Staff Requirements Memorandum (SRM) for SECY-92-223, dated September 18, 1992 (Reference 22), the Commission decided not to apply the GDC to plants with construction permits issued prior to May 21, 1971. At the time of promulgation of Appendix A to 10 CFR Part 50, the Commission stressed that the final GDC were not new requirements and were promulgated to more clearly articulate the licensing requirements and practice in effect at that time. Each plant licensed before the final GDC were formally adopted was evaluated on a plant-specific basis determined to be safe and licensed by the Commission.

As discussed in Appendix A of the Updated Final Safety Analysis Report (UFSAR), the licensee has made changes to the facility over the life of the plant that may have invoked the GDC. The extent to which the final GDC have been invoked can be found in specific sections of the UFSAR and in other design and licensing basis documentation.

The NRC staff's review of the BFN MELLLA+ LAR and its supplements is based upon the final GDC (GDC) listed below. A cross-reference of the applicable GDC to the AEC draft GDC that forms the BFN licensing basis is also shown.

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

The contents of the above GDC are in AEC draft GDC 1.

- GDC 2, "Design bases for protection against natural phenomena," requires SSCs important to safety to be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunamis, and seiches without loss of capability to perform their safety functions. The design bases for these SSCs shall reflect: (1) appropriate consideration of the most severe of the natural phenomena that have been historically reported for the site and surrounding area, with sufficient margin for the limited accuracy, quantity, and period of time in which the historical data have been accumulated; (2) appropriate combinations of the effects of normal and accident conditions with the effects of the natural phenomena; and (3) the importance of the safety functions to be performed.

The contents of the above GDC are in AEC draft GDC 2.

- GDC 4, "Environmental and dynamic effects design bases," requires SSCs important to safety be protected against dynamic effects associated with flow instabilities and loads such as those resulting from water hammer.

The contents of the above GDC are in AEC draft GDC 40 and 42.

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

The contents of the above GDC are in AEC draft GDC 4.

- GDC 10, "Reactor design," requires that the reactor core and associated coolant, control, and protection systems be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences.

The contents of the above GDC are in AEC draft GDC 6.

- GDC 11, "Reactor inherent protection," requires that the reactor core be designed so that the net effect of the prompt inherent nuclear feedback characteristics tends to compensate for a rapid increase in reactivity.

The contents of the above GDC are in AEC draft GDC 8.

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

The contents of the above GDC are in AEC draft GDC 7.

- GDC 13, "Instrumentation and control," requires that instrumentation shall be provided to monitor variables and systems over their anticipated ranges for normal operation, for AOOs, and for accident conditions, as appropriate, to assure adequate safety, including those variables and systems that can affect the fission process, the integrity of the reactor core, the reactor coolant pressure boundary, and the containment and its associated systems. Appropriate controls shall be provided to maintain these variables and systems within prescribed operating ranges.

The contents of the above GDC are in AEC draft GDC 12 and 13.

- GDC 14, "Reactor coolant pressure boundary," requires the reactor coolant pressure boundary to be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture.

The contents of the above GDC are in AEC draft GDC 9.

- GDC 16, "Containment design," requires that the containment and associated systems be designed to establish an essentially leaktight 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.

The contents of the above GDC are in AEC draft GDC 10.

- GDC 19, "Control room," 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 roentgen equivalent man (rem) whole body, or its equivalent, to any part of the body for the duration of the accident.

The contents of the above GDC are in AEC draft GDC 11.

- GDC 20, "Protection System Functions," requires that the protection system be designed (1) to initiate automatically the operation of appropriate systems, including the reactivity control systems, to assure that specified acceptable fuel design limits are not exceeded as a result of AOOs and (2) to sense accident conditions and to initiate the operation of systems and components important to safety.

The contents of the above GDC are in AEC draft GDC 14 and 15.

- GDC 21, "Protection System Reliability and Testability," requires that the system be designed for high functional reliability and inservice 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.

The contents of the above GDC are in AEC draft GDC 19.

- GDC 22, "Protection System Independence," requires that the system be designed so that natural phenomena and operating, maintenance, testing, and postulated accident conditions do not result in loss of the protection function.

The contents of the above GDC are in AEC draft GDC 20 and 23.

- GDC 23, "Protection System Failure Modes," requires that the system be designed to fail to a safe state in the event of conditions such as disconnection, loss of energy, or postulated adverse environments.

The contents of the above GDC are in AEC draft GDC 26.

- GDC 24, "Separation of Protection and Control Systems," requires that interconnection of the protection and control systems be limited to assure safety in case of failure or removal from service of any single common protection or control channels or component.

The contents of the above GDC are in AEC draft GDC 22.

- GDC 25, "Protection System Requirements for Reactivity Control Malfunctions," requires that the reactor protection system be designed to assure that SAFDLs are not exceeded for any single malfunction of the reactivity control systems.

The contents of the above GDC are in AEC draft GDC 31.

- GDC 26, "Reactivity Control System Redundancy and Capability," requires that two independent reactivity control systems be provided with both systems capable of reliably controlling the rate of reactivity changes resulting from planned and normal power changes, including AOOs, so that SAFDLs are not exceeded.

The contents of the above GDC are in AEC draft GDC 27, 28, 29, and 30.

- GDC 28, "Reactivity Limits," 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 sufficiently disturb the core, its support structures, or other reactor vessel internals to significantly impair the capability to cool the core.

The contents of the above GDC are in AEC draft GDC 32.

- GDC 29, "Protection Against Anticipated Operational Occurrences," 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 AOOs.

The contents of the above GDC are in AEC draft GDC 19.

- GDC 31, "Fracture prevention of reactor coolant pressure boundary," requires that the reactor coolant pressure boundary shall be designed with sufficient margin to assure that when stressed under operating, maintenance, testing, and postulated accident conditions, (1) the boundary behaves in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized. The design shall reflect consideration of service temperatures and other conditions of the boundary material under operating, maintenance, testing, and postulated accident conditions and the uncertainties in determining (1) material properties; (2) the effects of irradiation on material properties; (3) residual, steady state and transient stresses; and (4) size of flaws.

The contents of the above GDC are in AEC draft GDC 35.

- GDC 33, "Reactor coolant makeup," 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 shall 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.

The contents of the above GDC are in AEC draft GDC 37, 39, 41, and 44.

- GDC 34, "Residual heat removal," requires that 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.

The contents of the above GDC are in AEC draft GDC 6.

- GDC 35, "Emergency core cooling," requires an emergency system to provide abundant emergency core cooling to transfer heat from the reactor core following any LOCA.

The contents of the above GDC are in AEC draft GDC 37, 41, and 44.

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

The contents of the above GDC are in AEC draft GDC 41.

- GDC 41, "Containment atmosphere cleanup," requires systems to control fission products, hydrogen, oxygen, and other substances that may be released into the reactor containment be provided as necessary to reduce, consistent with the functioning of other associated systems, the concentration and quality of fission products released to the environment following postulated accidents, and to 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 41 also requires that each system have suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities 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), its safety function can be accomplished, assuming a single failure.

The contents of the above GDC are in AEC draft GDC 70.

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

The contents of the above GDC are in AEC draft GDC 49.

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

The contents of the above GDC are in AEC draft GDC 51 and 57.

The following guidance documents were used in this review:

- US NRC, Office of Nuclear Reactor Regulation, "Review Standard for Extended Power Upgrades," RS-001, Revision 0, dated December 2003 (Reference 20).

- NUREG-0800, "Standard Review Plan [SRP] for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition" (Reference 21), hereinafter referred to as the SRP, and 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 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"
 - Branch Technical Position 7-19 of NUREG-0800 (Reference 21), Guidance for Evaluation of Diversity and Defense-in-Depth in Digital Computer-Based Instrumentation and Control Systems
- NUREG-0737, "Clarification of TMI 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," dated November 1980 (Reference 23)
- NRC Staff Requirements Memorandum (SRM) on SECY-93-087, dated July 21, 1993 (Reference 24), describes the position of the NRC regarding diversity and defense-in-depth. This SRM states that applicants using digital or computer-based technology shall assess the defense-in-depth and diversity of the proposed instrumentation and control system to demonstrate that vulnerabilities to common mode failures have been adequately addressed. The SRM also states, "In performing the assessment, the vendor or applicant shall analyze each postulated common-mode failure for each event that is evaluated in the accident analysis section of the SAR using best-estimate methods. The vendor or applicant shall demonstrate adequate diversity within the design for each of these events."

3.0 TECHNICAL EVALUATION

3.1 BFN M+SAR and AMSAR Section 1.0, "Introduction"

Section 1 of the M+SAR (Attachment 6 to the LAR (Reference 1)) describes the report approach, as well as the differences between generic and plant-specific assessments. Generic assessments are those analyses that can be disposed of 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 M+SAR, the licensee stated that it will provide fuel and cycle dependent analysis, including the plant-specific thermal limits assessment. [[

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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 M+SAR, the licensee submitted reload calculations for a representative MELLLA+ cycle based on BFN Unit 3 Cycle 19 in ANP-3552NP (Attachment 20 to the LAR (Reference 1)), which meets the intent of L&C 12.4. For additional information regarding L&Cs, see Appendix B of this SE.

Table 1-1 of the M+SAR and Table 1-1 of the AMSAR (Attachment 8 to the LAR (Reference 1)) list the GEH and AREVA computer codes used in the MELLLA+ analysis.

Figure 1-1 of the M+SAR (reproduced here as Figure 3.1-1) defines the MELLLA+ operating domain for BFN. [[

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Section 1.2.4 of the M+SAR describes the allowed operational enhancements, which are covered by the approved M+LTR Safety Evaluation Report (SER). The impact of these operational enhancements in MELLLA+ is considered in the M+SAR and AMSAR analyses. The following enhancements are allowed in MELLLA+ at BFN:

- Increased core flow (ICF)
- 10 °F feedwater (FW) temperature operational band*
- 3 percent main steam safety relief valve (MSRV) setpoint tolerance
- Up to 1 MSRV out of service
- Turbine bypass valves out of service
- End of cycle recirculation pump trip out-of-service
- Power load unbalance out of service
- 24-month cycle

*Note the 10 °F FW temperature operational band is governed by a license condition proposed by the licensee. This license condition is evaluated in Section 4.1 of this SE.

The following operational conditions are not generically allowed in the MELLLA+ domain, and the licensee is not requesting plant-specific approval to incorporate these:

- Feedwater heater out of service (FWHOOS)/final FW temperature reduction
- Single-loop operation (SLO)

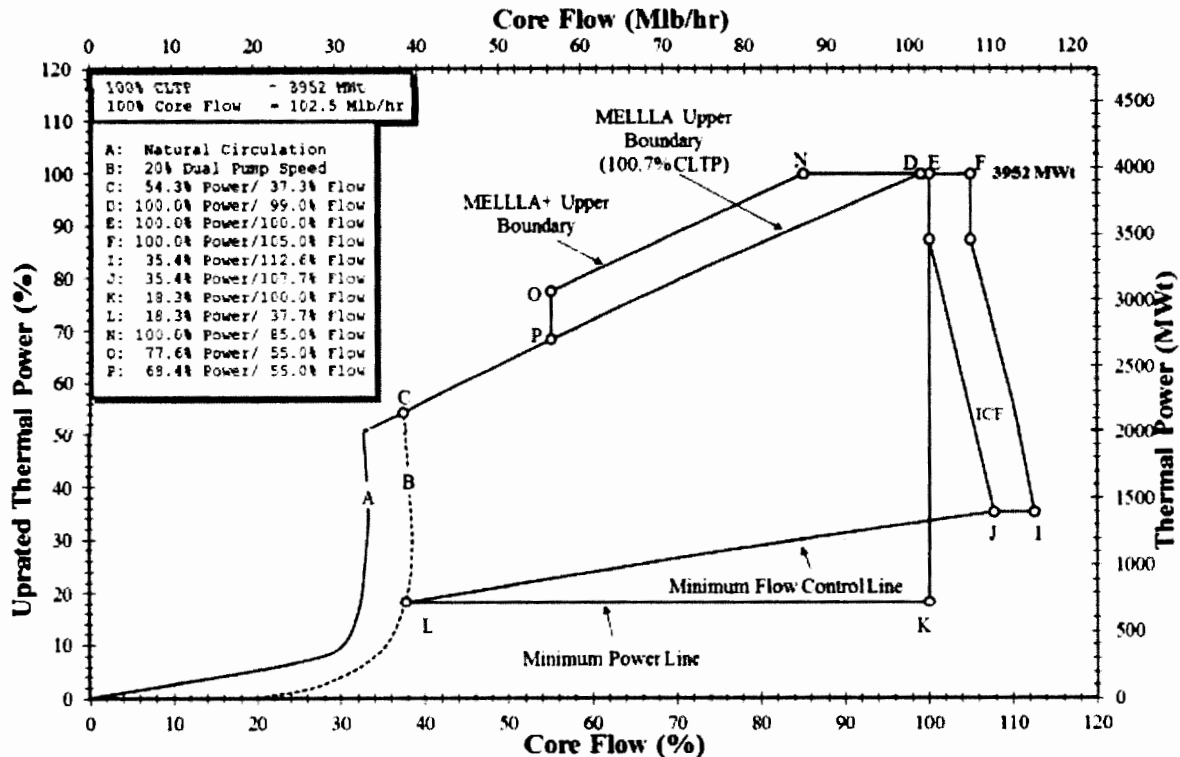


Figure 3.1-1 - Power/Flow Operating Map for BFN MELLLA+

3.2 BFN M+SAR and AMSAR Section 2.0, "Reactor Core and Fuel Performance"

3.2.1 BFN AMSAR 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, performance of the fuel system during normal operation, AOOs, and postulated accidents. The NRC's acceptance criteria are based on 10 CFR 50.46 and GDC 10, 26, and 35. Specific review criteria are contained in SRP Section 4.2 and other guidance provided in Matrix 8 of RS-001 (Reference 20).

Limitations and Conditions

The SER for the Methods LTR (Reference 19) and the SER for the M+LTR (Reference 17) contain limitations and conditions (L&Cs) pertaining to the fuel system design. The licensee addressed these L&Cs in Appendices A and B of the M+SAR and Appendices A and B of the AMSAR. The NRC staff evaluation for these limitations is provided in Appendices A and B of this SE.

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. The staff's evaluation of these analyses and the results are documented in this section.

Only AREVA fuel types will be resident in the BFN cores during MELLLA+ implementation. The predominant fuel type will be ATRIUM 10XM fuel design with any remaining ATRIUM-10 fuel type in non-limiting locations on or near the periphery. Therefore, this section on fuel design will be limited to potential impact of the MELLLA+ extended power/flow operating domain (EPFOD) on the ATRIUM 10XM fuel design.

The ATRIUM 10XM fuel design is comprised of a 10 by 10 array of fuel rods with a square internal water channel that displaces a [] 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, [] 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-3568NP, Revision 2 (Attachment 22 to the LAR (Reference 1)), for the ATRIUM 10XM fuel transition that began in BFN Unit 2 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 American Society of Mechanical Engineers (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. The NRC staff's review of fuel design performance, and structural design of the fuel assembly and the fuel channel found that the design meets all mechanical compatibility and strength requirements for operation under the MELLLA+ domain at BFN.

The fuel design analysis assesses fuel rod performance at EPU conditions with operation into the MELLLA+ operating domain that are assumed to first occur in Cycle 19 of BFN Unit 3. This unit is considered as a proxy for all other units. A first batch of ATRIUM 10XM inserted in Cycle 18 was at pre-EPU power and within the MELLLA operating domain. The second batch of ATRIUM 10XM fuel was inserted in Cycle 19. For Cycle 19, as well as subsequent cycles, the thermal power is assumed to be 120 percent of original licensed thermal power (OLTP), and operation is assumed to be in the MELLLA+ operating domain. Performance for cycles beyond Cycle 19 is assessed using an equilibrium cycle comprised exclusively of ATRIUM 10XM fuel assemblies under EPU conditions and the MELLLA+ operating domain.

Fuel design analysis is performed according to the generic fuel thermal and mechanical design criteria contained in ANF-89-98 (P)(A), Revision 1, and Supplement 1 (Reference 25), along with the design criteria provided in RODEX4 fuel thermal-mechanical (T-M) topical report (Reference 26). The fuel rod design is nearly identical to the design used for the first U.S. ATRIUM 10XM lead fuel assemblies, the reload fuel currently supplied to two other U.S. BWR/4 reactors, and first reloads of ATRIUM 10XM fuel in BFN Unit 2 Cycle 19, Unit 3 Cycle 18, and Unit 1 Cycle 12. These ATRIUM 10XM fuel assemblies are loaded with pellets composed of either [[

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The fuel rod analyses (ANP-3568P, Revision 2) cover normal operating conditions and AOOs. The fuel centerline temperature analysis (overheating of fuel) and cladding strain analysis consider slow transients at rated operating conditions. Other fuel-rod related topics such as overheating of cladding, cladding rupture, fuel rod mechanical fracturing, rod bow, axial irradiation growth, cladding embrittlement, violent expulsion of fuel, and fuel ballooning are evaluated as part of the respective fuel assembly structural analysis, thermal hydraulic (TH) analyses, or LOCA analyses, and are reported in the documents submitted as part of the MELLLA+ LAR.

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 T-M 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 BFN TS 5.6.5; this includes cycle-specific confirmation that the ATRIUM 10XM fuel design limits for BFN established using the approved RODEX4 methodology remain applicable for each reload cycle. Reload

evaluations for MELLLA+ operating cycles will use MELLLA+ specific core configurations for the depletion calculations consistent with Safety Evaluation Report for NEDC-33006P-A, Revision 3 (M+SER) L&C 12.3.e (see Appendix B of this SE).

Table 3-2 through Table 3-4 of ANP-3568P, Revision 2 (Attachment 21 to the LAR (Reference 1)), provide results from the fuel rod design analysis. This summary of fuel rod design evaluation includes internal hydriding, cladding collapse, overheating of fuel pellets, stresses and strain limits, cladding fatigue, cladding oxidation, hydriding and crud buildup, rod internal pressure, and plenum spring design. The design criteria for the various design parameters and the methodologies used for the evaluation are acceptable for the fuel rod thermal and mechanical design analysis.

The NRC staff reviewed the impact on the fuel system of the proposed MELLLA+ operating domain extension based on the analyses provided by the applicant for normal operation, AOOs, and infrequent and special events. The complete NRC staff evaluation of these results is documented in Section 3.9, "Reactor Safety Performance Evaluations." As stated in that evaluation, operation at the lower MELLLA+ flows will not have any impact on the response for anticipated transients because all AOOs analyzed are limiting at the 105 percent CF condition. Furthermore, the applicant analyses demonstrate that, with the proposed BFN 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 of fuel on operation with the more restrictive setpoints requested by the licensee at the lower MELLLA+ flows is minimal.

Conclusions for Section 3.2.1

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 and GDC 10, 26, and 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 BFN AMSAR 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 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 SER for the Methods LTR (Reference 19) and the SER for the M+LTR (Reference 17) contain L&Cs pertaining to the fuel system design. The licensee addressed these limitations in Appendices A and B of the M+SAR and Appendices A and B of the AMSAR. The details of the NRC staff evaluation for these limitations are discussed in Appendices A and B of this SE.

Technical Evaluation: AREVA Fuel Thermal-Hydraulic Performance

ANP-3550P (Attachment 25 to the LAR (Reference 1)) provides the results of TH analysis to demonstrate that the AREVA ATRIUM 10XM fuel is hydraulically compatible with resident ATRIUM-10 fuel at EPU MELLLA+ conditions. ATRIUM 10XM and ATRIUM-10 fuel assemblies are geometrically different, but hydraulically the two designs are compatible at EPU MELLLA+ operation. [[

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Even though the TH design analyses were performed on a generic fuel design basis, TH analyses are reperformed on a cycle-specific basis due to differences in reactor and operating cycles. The TH design criteria are summarized as:

- Relative thermal margin analyses in accordance with TH methodology (see Appendix E for XCOBRA code discussion) and approved ACE/TRIUM 10XM critical power ratio (CPR) correlation has concluded that TRIUM 10XM fuel will not adversely affect the thermal margin performance of the TRIUM-10 fuel.
- Fuel design operation shall be such that fuel centerline melting will not occur during normal operation and AOOs.
- Rod bow magnitude shall be accounted for in establishing thermal margin performance.
- Bypass flow (leakage flow through lower tie plate the (LTP) flow holes) analysis has shown that the slight decrease in bypass flow will not significantly impact bypass exit subcooling.
- Core stability, which is strongly dependent on core power, CF, and core power distribution, is evaluated on a cycle-specific basis. The core stability performance of

new fuel designs is found to be equivalent to, or better than, the existing AREVA fuel design.

- ASME pressure vessel code requirements with respect to overpressurization are satisfied.
- Under accident conditions, the assemblies are found to remain engaged in the fuel support.

Table 3.3 of ANP-3550P provides hydraulic characterization comparison between ATRIUM-10 and ATRIUM 10XM fuel designs from XCOBRA-based TH models.

Section 2 of ANP-3551NP (AMSAR, Attachment 8 to the LAR (Reference 1)), Revision 0, documents reactor core and fuel performance that includes fuel design and operation; core design and fuel thermal monitoring thresholds; thermal limits assessment; safety limit minimum critical power ratio; operating limit minimum critical power ratio; minimum average planar heat generation rate limits; linear heat generation rate limits; power-to-flow ratio; and reactivity characteristics such as hot excess reactivity, strong rod out shutdown margin, and standby liquid control system (SLCS) shutdown margin.

The licensee performed representative plant-specific applicable evaluations for reactor core and fuel performance and reactivity characteristics. Results from hydraulic characterization indicate that these fuel designs are hydraulically compatible at BFN for the entire range of the licensed EPU MELLLA+ power-to-flow operating map. The loss coefficients have been modified from the test data to [[

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Reactor Core Design and Fuel Thermal Monitoring Threshold

As a result of MELLLA+ implementation, the maximum licensed power level and the fuel design do not change. The CLTP remain at the power level of 3,952 MWt, and BFN will continue to use the AREVA ATRIUM 10XM fuel design. As a result, the average bundle power and average core power density remain unchanged. Therefore, there is no change required in the fuel thermal monitoring threshold.

Since the MELLLA+ EPFOD allows for higher power to flow conditions, the M+LTR recognizes that this may cause the range of void fraction, axial and radial power shape, and control rod positions in the core to change slightly. In spite of these changes, the individual fuel bundles are required to remain within the allowable thermal limits, which are calculated and included in the cycle-specific core operating limits report (COLR) that supports the plant operation during that cycle.

Operation at MELLLA+ EPFOD will increase the core average void fraction when compared to operation at the same power level in the MELLLA operating domain. This potential increase in the core average void fraction is driven by the corresponding increase in the in-channel power-to-flow ratio. The potential for bypass region voiding may increase with MELLLA+ EPFOD operation due to the lower total CF. [[

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A significant impact of voiding in the bypass region would be the impact on the local power range monitor (LPRM) reading. However, a confirmatory evaluation described in Section 3.1 of ANP-3551 (AMSAR, which is Attachment 8 to the LAR (Reference 1)) has concluded that there is no impact to the LPRM response.

Section 2.1 of ANP-3551NP provides documentation for core design and fuel monitoring parameters for each cycle exposure statepoint consistent with L&C 9.24 of GEH Methods LTR (Reference 19). The NRC staff has reviewed the documentation and found that BFN meets the intent of the M+SER with respect to core design and the fuel monitoring threshold.

Safety Limit Minimum Critical Power Ratio (SLMCPR)

The safety limit minimum critical power ratio (SLMCPR) is calculated as the minimum critical power ratio (CPR) value that guarantees that 99.9 percent of the fuel rods in the core will not experience boiling transition under normal operation and anticipated occurrences. The licensee calculates the SLMCPR based on the cycle-specific core loading pattern for each reload core and reports the results in the RSAR for each reload core. For this BFN MELLLA+ application, the licensee provided an SLMCPR demonstration analysis at MELLLA+ conditions and justification of use of the SLMCPR methodology currently implemented at BFN at MELLLA+ conditions.

The methodology currently in the BFN licensing basis is the generically approved SAFLIM3D methodology (Reference 28). This statistically-based methodology employs a Monte Carlo process that perturbs key input parameters (uncertainties) used in the MCPR calculation to confirm the SLMCPR criterion. The licensee provided the fuel and plant-related uncertainties in Table 1 of FS1-0029291 Revision 1.0 (Attachment 23 to the LAR (Reference 1)).

The licensee provided the results of the demonstration SLMCPR analysis for an equilibrium core of ATRIUM 10XM fuel for MELLLA+ at BFN in Table 2 of FS1-0029291 Revision 1.0 (Attachment 23 to the LAR (Reference 1)). The licensee evaluated the [[

]] statepoints in the operating domain. Consistent with the restrictions in Section 2.2.1.1 of the M+SER, the licensee imposed the SLO flow uncertainties to the [[

]] statepoints to account for possible errors in the flow measurement at the higher void fraction conditions expected in the MELLLA+ region. For all conditions evaluated, the number of rods in boiling transition is lower than 0.1 percent, with the highest number at the [[]]] with 0.093 percent of rods predicted to experience boiling transition, as presented in Table 2 of FS1-0029291, Revision 1.0, (Attachment 23 to the LAR (Reference 1)).

The NRC staff reviewed the demonstration analysis to determine if it meets GDC 10 and 12, if the analysis appropriately followed the M+LTR and SAFLIM3D methodology, and if the results are reasonable for BFN. The staff reviewed the applicability of the SAFLIM3D methodology at MELLLA+ conditions in Appendix E of this SE.

The NRC staff determined the licensee appropriately followed the methodology by comparing the demonstration analysis, FS1-0029291, Revision 1.0, to the SAFLIM3D LTR (Reference 28). Specifically, the NRC staff observed that the licensee incorporated the ACE/TRIUM 10XM critical power correlation and the channel bowing effects, which were relatively new features in

the SAFLIM3D methodology. There are no L&Cs in the SAFLIM3D for the licensee to follow explicitly. Because of these reasons, the NRC staff determined that the licensee appropriately followed the SAFLIM3D methodology.

The M+LTR requires that the MELLLA+ statepoints (100 percent EPU power/maximum CF, 100 percent EPU power/minimum CF, maximum core power at minimum MELLLA+ flow) be evaluated to determine the most limiting SLMCPR. Additionally, the methodology requires the higher SLO uncertainties be included in the SLMCPR analysis at the 100 percent EPU power/minimum CF statepoint and the maximum core power at minimum MELLLA+ flow. Table 3.2-1 below provides the BFN-specific MELLLA+ statepoints and associated uncertainty and whether they are two loop or single loop dependent, as used in the SLMCPR analysis.

Table 3.2-1: Recirculation Loop Dependent Uncertainties Used in BFN MELLLA+ SLMCPR Analysis

Statepoint		Uncertainty		
Power (%)	Flow (%)	Assembly Radial Peaking (%)	Nodal Power (%)	Total Core Flow (%)
[[]]	2.5
[[]]	6.0
[[]]	6.0

¹ [[

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² [[

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The NRC staff determined the licensee appropriately followed the M+LTR for the SLMCPR analysis since the licensee analyzed the required statepoints and used the required uncertainty for the statepoints. Note: The Methods LTR contains additional L&Cs (9.4 and 9.5), which are addressed below.

The licensee presented the demonstration analysis in Table 2 of FS1-0029291, Revision 1.0 (Attachment 23 to the LAR (Reference 1)). The analysis supports a two-loop operation (TLO) SLMCPR of 1.06 and an SLO SLMCPR of 1.07. The limiting statepoint is the [[
]] At this statepoint, 0.0935 percent of rods was predicted in boiling transition, which is less than the acceptance criterion of 0.1 percent. Therefore, the analysis meets the criteria in GDC 10 and 12 since the results meet the SLMCPR fuel design limit under normal operation and anticipated occurrences.

The SER for the Methods LTR (Reference 19) contains two L&Cs that require adders to the SLMCPR analysis results. L&C 9.4 requires a 0.02 adder to the cycle-specific SLMCPR value for EPU operation. L&C 9.5 requires a 0.03 adder to the cycle-specific SLMCPR for MELLLA+ operation. These conservative adders were included to address concerns in methodology that supported the Methods LTR. The licensee is requesting approval without either of these SLMCPR adders since the SAFLIM3D methodology was not part of the Methods LTR approval.

BFN is approved (Reference 16) for BFN EPU operation using the SAFLIM3D methodology. The NRC staff required no additional adder as part of its review. Implementation of MELLLA+ will not impact the approval of EPU operation. Therefore, the NRC staff determined that the 0.02 SLMCPR adder is for EPU operation (L&C 9.4) and since the adder was not required when the NRC approved BFN EPU operation, it is also not necessary for MELLLA+.

The 0.03 SLMCPR adder in L&C 9.5 of the Methods LTR was based, in part, on the ability of that code to predict the power distribution in the core and the limited data used to validate the code predictions. For this application, the licensee demonstrated its methodology's ability to predict the power distribution and provided additional data to qualitatively validate the prediction.

The licensee uses the CASMO-4/MICROBURN-B2 methodology (Reference 29) to calculate the power distribution and associated uncertainties used in the SLMCPR analysis (SAFLIM3D methodology). The power distribution uncertainties determined by CASMO-4/MICROBURN-B2 and passed to SAFLIM3D are the nodal and radial power uncertainty. These uncertainties are dependent, in part, upon the traversing in-core probe (TIP) measurement uncertainty, LPRM measurement uncertainty, and calculated uncertainty from the CASMO-4/MICROBURN-B2 methodology.

The base nodal and radial measured power uncertainties are found in Section 9.7 of the evaluation and validation of the CASMO-4/MICROBURN-B2 methodology. The nodal and radial uncertainties for an equilibrium core of ATRIUM 10XM fuel for BFN at MELLLA+ conditions are found in Table 1 of FS1-0029291, Revision 1.0 (Attachment 23 to the LAR (Reference 1)). Compared to the base nodal and radial uncertainties, the equilibrium BFN MELLLA+ uncertainties are higher because they include additional uncertainty for items such as equipment out of service.

To justify that an SLMCPR adder is not necessary, the licensee provided data to demonstrate CASMO-4/MICROBURN-B2 methodology's ability to predict the core power distribution. The licensee provided 13 cycles of 2D TIP uncertainty (δT_{ij}) data for BFN Units 1, 2, and 3 in ANP-2860NP, Revision 2, Supplement 3NP, Revision 2 (Attachments 27 (proprietary) and 28 (nonproprietary) to the LAR (Reference 1)). This uncertainty is a comparison of the measured and code predicted bundle power. The average 2D TIP uncertainty is approximately $\pm 0.5\%$. This is less than the $\pm 1.0\%$ 2D TIP uncertainty determined in Section 9.7 of CASMO-4/MICROBURN-B2 methodology. The licensee also provided 2D and 3D TIP data, which were presented for other plants. The staff determined that none of the TIP data show a discernible trend in uncertainty with respect to power-to-flow ratio, core average void fraction, or power. However, the BFN data only extend as high as approximately 42 MWt/Mlb/hr. Although the data for the other plants included power-to-flow ratios up to 52 MWt/Mlb/hr, few data were obtained above 42 MWt/Mlb/hr. For BFN, 42 MWt/Mlb/hr encompasses a large portion of the MELLLA+ domain, with 52 MWt/Mlb/hr being exceeded in only a small corner of the MELLLA+ domain, which will not typically be entered during normal cycle operation.

To supplement the TIP data, the licensee provided measured versus calculated LPRM statistics for BFN and a BWR/6 plant. $\pm 0.5\%$

$\pm 0.5\%$ The staff observed no discernible trend in LPRM data uncertainty relative to power-to-flow ratio. The LPRMs are periodically calibrated to the TIP data during operation; therefore, measured vs. calibrated LPRM statistics have an indirect correspondence with measured power distribution uncertainties. In its evaluation of ANP-10300P (Reference 30), the NRC staff concluded that the use of LPRM uncertainty to assess a qualitative change in bundle power distribution uncertainty is reasonable, provided that the LPRM uncertainty shows a qualitatively similar trend as the measured TIP uncertainties. In the data provided for the BFN MELLLA+ application, the NRC staff observed no discernible trend in either TIP or LPRM uncertainties as a function of power-to-flow ratio, including for the data points above 42 MWt/Mlb/hr.

The NRC staff determined that the approved power distribution uncertainties CASMO-4/MICROBURN-B2 methodology used in the SAFLIM3D analysis is applicable for BFN in the MELLLA+ operating domain because of the following:

- There is no adverse discernable trend in the TIP data for BFN or other plants presented.
- There is no adverse discernable trend in the LPRM data for BFN or other plants presented.
- There is significant margin in the BFN TIP uncertainty compared to that in uncertainties CASMO-4/MICROBURN-B2 and that used in the SALFLIM3D analysis.

Since the NRC staff determined that the CASMO-4/MICROBURN-B2 power distribution uncertainties are acceptable for BFN MELLLA+ operation, no additional SLMCPR penalty (i.e., L&C 9.5) is needed for BFN MELLLA+ operation.

Operating Limit Minimum Critical Power Ratio (OLMCPR)

The operating limit minimum critical power ratio (OLMCPR) is calculated by adding the change in MCPR (i.e., delta-CPR) due to the limiting AOO event to the SLMCPR. The OLMCPR for BFN is determined on a cycle-specific basis from the results of the reload transient analysis, which is documented in the RSAR.

Section 3.9.1 of this SE discusses the demonstration AOO analyses that were performed for BFN MELLLA+ and the limiting delta-CPR value from these analyses. The justification for that AOO methodology is reviewed in Appendix E of this SE.

AREVA Critical Power Ratio (CPR) Correlations

All AREVA critical heat flux (CHF) and CPR correlations are approved by the NRC staff to be applicable over specified ranges of assembly operating conditions. 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 31) and ANP-3140P (Reference 32). The applicable critical power correlation for ATRIUM-10 fuel is the Siemens Power Corporation B (SPCB) critical power correlation, EMF-2209PA, Revision 3 (Reference 33). When any of the BFN units included a mixed core of ATRIUM-10 and ATRIUM 10XM fuel, the critical power was only evaluated for the ATRIUM 10XM assemblies using the ACE critical power correlation and the critical power for ATRIUM-10 assemblies was evaluated using the SPCB critical power correlation (Attachment 18 to the LAR (Reference 1)). The NRC staff finds this acceptable because these fuel types are similar, which is acceptable 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 BFN MELLLA+ presented in ANP-2860NP, Revision 2, Supplement 3P, Revision 2 (Attachment 28 to the LAR (Reference 1)), in Appendix E of this SE and concludes that the use of the ACE correlation is acceptable for the ATRIUM 10XM fuel in BFN MELLLA+. Applicability

of SPCB at EPU conditions for ATRIUM-10 fuel was previously addressed in ANP-2860NP, Revision 2, and remains valid for EPFOD operating conditions.

Void Fraction Correlations

Quantification of the void-quality correlation used in the AREVA methodologies was presented in Section 3.1 of ANP-2860NP, Revision 2, Supplement 3NP, Revision 2, using ATRIUM-10 measurements. [[

]] 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-2860NP, Revision 2, Supplement 3NP, Revision 2 (Attachment 28 to the LAR (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 TH conditions in the BFN 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 BFN MELLLA+ application.

Qualification of the void-quality correlations used in AREVA methodologies using ATRIUM 10XM measurements was presented in Section 2.2 of ANP-2860NP, Revision 2, Supplement 1NP. Figure 3-4 of ANP-2860NP (Attachment 28 to the LAR (Reference 1)), Revision 2, Supplement 3NP, Revision 2, compares the predicted to measured void fraction correlation to FRIGG and ATRIUM-10 test data. The NRC staff has determined that good agreement was achieved between predicted and measured void fraction with measured ATRIUM 10XM void fractions up to [[]].

3.2.3 BFN AMSAR 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, 11, 12, 13, 20, 25, 26, and 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 Safety Evaluation Report for NEDC-33173P-A, Revision 4, Applicability of GE Methods to Expanded Operating Domains LTR (Methods SER) (Reference 19) and the M+SER (Reference 17) contain L&Cs pertaining to the nuclear and fuel design. The licensee addressed these

limitations in Appendices A and B of the M+SAR and AMSAR. The details of the NRC staff evaluation for these limitations are discussed in Appendices A and B of this SE.

Technical Evaluation

Analysis for the representative core design is documented in ANP-3553NP (Attachment 20 to the LAR (Reference 1)) for Unit 3 Cycle 19. This representative core design consists of 352 fresh ATRIUM 10XM fuel assemblies, 304 irradiated ATRIUM 10XM fuel assemblies, and 108 irradiated ATRIUM-10 assemblies. The Cycle 19 fresh batch of 352 assemblies with a batch average enrichment of $[[\quad]]$ was determined to meet the requirements for the operation of the plant at the EPU MELLLA+ operating condition. At the time of initial MELLLA+ implementation, the BFN units will be operating with a mixed core of ATRIUM-10 and 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-3553NP (Attachment 20 to the LAR (Reference 1)). The cycle design calculations demonstrate adequate hot excess reactivity and cold shutdown margin throughout the cycle. ANP-3553NP 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 $[[\quad]]$ provides pin power and exposure distributions and 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 TH models define the flow distribution among the assemblies. Models for the calculation of CPR, linear heat generation rate (LHGR), and maximum average planar linear heat generation rate (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 occur 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 good agreement for the entire exposure range of plant operation.

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 29) 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 CF indicates 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 assembly gamma scan data were 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 9 by 9 assemblies and AREVA ATRIUM-10 fuel assemblies in the reactor designated as KWU-S.

Pin-by-pin gamma scan data are used for verification of the local peaking factor uncertainty. Quad Cities Nuclear Power Station measurements presented in the LTR EMF-2158(P)(A) (Reference 29) 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 showed 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 demonstrated 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-2860NP, Revision 2, Supplement 3NP, Revision 2 (Attachment 28 to the LAR (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. Since the neutronic and TH conditions predicted for the MELLLA+ condition 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 ANP-3553NP (Attachment 20 to the LAR (Reference 1)) indicates that MELLLA+ operation in the power/flow map, shown in Figure 3.1-1 of this SE, is within the range of the original methodology approval for assembly power and exit void fraction.

Fuel Assembly Design

For BFN 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 weight (wt) percent U-235. Descriptions of fuel enrichments on both a lattice and assembly basis for the first reload of ATRIUM 10XM fuel in BFN are listed in Table 3.1 of ANP-3553NP.

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

ANP-3553NP (Attachment 20 of LAR (Reference 1)) documents the reference cycle, which determined a SLC minimum shutdown margin of 2.63 $\Delta k/k$ at beginning of cycle (BOC), assuming a short exposure basis for the previous cycle. This is greater than the 0.88 $\Delta k/k$ acceptance criteria applied by AREVA for its methods.

The hot shutdown boron weight (HSBW) is calculated on a plant- and cycle-specific using the fuel designs loaded into each core (e.g., ATRIUM 10XM in the case of BFN). These plant- and cycle-specific calculations confirm that HSBW remains effective to shut down the BFN core under MELLLA+ initial conditions.

BFN Units 1, 2, and 3 increased the Boron 10 enrichment from 19.8 atom percent to 94 atom percent as part of plant modifications for EPU. There are no new changes to the SLC system design since the increased Boron-10 enrichment for EPU. Since the reactor pressure is not changed and the SLC system boron inventory shutdown margin is evaluated for the representative core in the BFN MELLLA+ LAR that resulted in acceptable margin, the NRC staff finds the requirements of 10 CFR 50.62(c)(4), and the intent of GDC 26 will continue to be satisfied.

Conclusion for Reactivity Characteristics

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 representative 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. Therefore, the NRC staff finds SLC system changes acceptable for the proposed operating domain extension.

3.2.4 BFN M+SAR Section 2.4, "Stability"

BFN currently operates with the Option III stability solution. To operate in the MELLLA+ region, BFN will be implementing a DSS-CD stability solution. The DSS-CD solution is designed to identify the power oscillation upon inception and initiate control rod insertion (scram) to terminate the oscillations prior to any significant amplitude growth.

Regulatory Basis

Acceptance criteria pertaining to BWR stability are based on GDC 10, 12, 13, 20, and 29, as well as Generic Letter (GL) 94-02, which concerns the installation of long-term stability (LTS) to satisfy GDC 10 and 12. SRP 15.9 (Reference 21) defines acceptance criteria 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 17) and associated SER approved the use of DSS-CD (Reference 18) as a long-term stability solution in the MELLLA+ operating domain. The relevant L&Cs in the M+LTR, which are addressed in Appendix B of the BFN M+SAR, are:

- M+SER L&C 12.2, which specifies the compliance with the L&Cs in the DSS-CD LTR SER;
- M+SER 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+SER L&C 12.3g, which specifies the use of an approved stability protection method and approved backup stability method for MELLLA+ operation; and
- M+SER L&C 12.7, which specifies a non-manual backup stability protection system for operation in the MELLLA+ domain.

The SER for the DSS-CD LTR (Reference 18) contains the following L&Cs relevant to DSS-CD implementation for operation in MELLLA+, which are addressed in Appendix C of the M+SAR:

- DSS-CD SER L&C 5.1, which specifies the use of GEH Option III hardware or a hardware review for non-GEH hardware;
- DSS-CD SER 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 SER 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 SER L&C 5.4, which requires plant licensees to ensure that the DSS-CD trip function is applicable in their plant licensing bases, including the optional BSP trip function, if it is to be installed.

There is one L&C in the Methods SER (Reference 19) pertaining to the nuclear and fuel design. The licensee addressed this limitation in Appendix A of the M+SAR.

- Methods SER 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 to meeting the requirements of GDC 10 and 12. Therefore, it is necessary for BWRs to implement an LTS solution that has the capability of automatically suppressing the instabilities. The LTS solution currently implemented in BFN is Option III. Option III is not approved for use in the MELLLA+ operating domain as stated in the M+SER (Reference 17). Therefore, BFN will implement the DSS-CD long-term stability solution (Reference 18), which was approved in the M+SER and DSS-CD SER 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 CDA, which identifies oscillations and their characteristics by a period-based algorithm (PBA) and requires a sufficient number, or a "confirmation density," of oscillation power range monitor (OPRM) cells to generate an OPRM channel trip signal, [[

]]. In this manner, power oscillations are detected and suppressed prior to challenging the SAFDLs.

[[

]]

The DSS-CD solution was generically demonstrated to be able to automatically detect and suppress instabilities with approved procedures provided 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, it was not necessary for the NRC staff to review the DSS-CD design itself to ensure that it can suppress reactor instabilities. The staff focused its review on adequate implementation of the DSS-CD solution for the plant and operating conditions at BFN in the MELLLA+ domain.

BFN 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 stated 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 perform a one-time, plant-specific analysis to extend the applicability to include the ATRIUM 10XM fuel.

The potential exists for the first MELLLA+ cycle in BFN to include some legacy ATRIUM-10 fuel [[

]] The NRC staff issued SRXB request for additional information (RAI)-6 (Reference 8) to request more

information on the acceptability of implementing DSS-CD with the possible presence of ATRIUM-10 fuel in the first MELLLA+ cycle. In its RAI response, [[

]]

The stated purpose [[in the approved DSS-CD methodology is to confirm that sufficient MCPR margin is maintained from inception to suppression of anticipated oscillations, considering the specific fuel loading and plant configuration for BFN M+SAR and including transition cycles if these are present at the time of MELLLA+ implementation. For BFN, the first MELLLA+ operational cycle will consist of a transition from ATRIUM-10 fuel to ATRIUM 10XM fuel. Table 6-5 of the DSS-CD LTR, which defines required TRACG cases for various fuel transition scenarios, [[

]]

The intent of performing the transition cycle analyses is to ensure that safety limits are not violated during these cycles following DSS-CD implementation. The NRC staff evaluated the acceptability of omitting the ATRIUM-10 transition cycle on this basis. To perform this evaluation, the NRC staff considered two primary factors. First, the NRC staff sought to determine whether the limiting MCPR margin could occur in ATRIUM-10 fuel bundles, in which case the MCPR margin could not be acceptably calculated by analysis of ATRIUM 10XM fuel only. In this regard, the NRC staff evaluated the licensee's SRXB-RAI 6 response (Reference 8) and concludes with reasonable assurance that the limiting MCPR margin will not occur in the peripheral ATRIUM-10 bundles, which would be located on or near the periphery of the core, during the last transition cycle. This is based on the result that showed these locations were not limiting during the BFN MELLLA+ equilibrium cycle and the NRC staff's expectation that ATRIUM-10 would exhibit reasonably similar MCPR margins as ATRIUM 10XM due to the similarity of these fuel types.

The NRC staff's second consideration was to determine whether the last transition cycle would be expected to exhibit significantly different overall stability behavior compared to the equilibrium cycle; for example, significantly different harmonic mode shape or other large-scale differences such that the conclusions from the equilibrium cycle analyses could not be reasonably applied to the last transition cycle. With respect to this topic, the NRC staff's experience is consistent with the licensee's statement that the oscillations are driven not by the low-powered bundles located on or near the periphery of the core, but by the higher-powered bundles, particularly those near the harmonic peak locations (for out-of-phase oscillations) or fundamental mode peak locations (for core-wide oscillations). Additionally, the relative similarity between ATRIUM-10 and ATRIUM 10XM and the limited number of potential ATRIUM-10 bundles in the transition cycle provided the NRC staff with further assurance that the overall stability behavior will not be largely affected by the presence of ATRIUM-10 fuel in the transition cycle. For these reasons, the NRC staff concludes that the licensee's omission of transition cycle analyses with ATRIUM-10 fuel is acceptable for the BFN MELLLA+ application.

[[

]]

The extension of the applicability envelope, as discussed in Section 6.0 and Tables 6.3 and 6.5 of the DSS-CD LTR, [[

]]

The licensee proposed an MCPR margin [[

]] 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 staff concludes that the licensee can sufficiently analyze the impact of future core designs with respect to LTS.

DSS-CD Time Period Lower Limit Change

In addition to the extension of the generic DSS-CD applicability envelope, the licensee is also requesting an increase to the time period lower limit (T_{min}), which is designated as a fixed parameter in the DSS-CD LTR. This parameter constitutes the minimum time period value, which may be used to generate successive confirmation counts in the PBA. If oscillations were to occur with a period less than the time period lower limit, no confirmation counts would be generated, and the PBA 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 oscillations in BFN, including MELLLA+ operation.

In its RAI response (Reference 8), [[

]] The minimum period value determined using this method was 1.51 seconds, which provides a 0.31 second margin to the proposed T_{\min} value of 1.2 seconds.

[[

]] Because this value is more than 0.2 seconds above the proposed T_{\min} of 1.2 seconds, the staff concludes that a T_{\min} of 1.2 seconds follows the intent of the DSS-CD LTR in providing sufficient 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 Methods LTR (Reference 19), "The existing GE thermal-hydraulic stability models reasonably and adequately model the magnitude and period of industry thermal-hydraulic instability events." [[

]]

Based on these considerations, the staff concludes that a T_{\min} of 1.2 seconds, which will continue to be a [[], provides adequate safety protection against anticipated TH oscillations in BFN MELLLA+ [[

]].

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 limiting conditions for operation (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 determined they support implementation of the DSS-CD methodology, and therefore, the changes are acceptable.

The proposed TS changes include implementation of automated backup stability protection (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 3 Cycle 19 of BFN in the Representative Reload Analysis (ANP-3552P, Attachment 17 to the LAR (Reference 1)). The NRC staff reviewed these BSP regions and concluded that they were determined in accordance with the DSS-CD LTR, and therefore, are 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+SER L&Cs 12.3.g and 12.7.

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

- TS 3.3.1.1, Reactor Protection System (RPS) Instrumentation

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

- I.1 Initiate action to implement the Manual Backup Stability Protection (BSP) Regions defined in the COLR). (Completion Time: Immediately)

AND

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

AND

- I.3 Initiate action to submit an OPRM report 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 18), and these changes are consistent with 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 of Appendix A (hereafter "sample BWR4 TSs of Appendix A") of the approved LTR. The sample TSs within LTR Appendix A are applicable to all three BFN units because these plants are GE Type 4 BWR plants. These required actions are, therefore, acceptable.

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

Required Action J.1 is changed from one action to three actions, as follows:

- J.1 Initiate action to implement the Manual BSP Regions defined in the COLR. (Completion Time: Immediately)

AND

- J.2 Reduce operation to below the BSP Boundary defined in the COLR.
(Completion Time: 12 hours)

AND

- J.3 Note: LCO 3.0.4 is not applicable.

Restore required channel to OPERABLE status. (Completion Time: 120 days)

This change addresses the situation where the required action and associated completion time of Condition I are not being met. The NRC staff determines that these proposed required actions are also the same actions provided in the sample BWR-4 TSs of Appendix A to the approved LTR and are consistent with Section 8, "Effect on Technical Specifications," of the approved LTR (hereafter "Section 8 of the approved LTR") (Reference 18). These required actions are, therefore, acceptable.

- TS 3.3.1.1, Required Action K.1

A new Required Action K.1 is added 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 percent of rated thermal power, and this action must be completed within 4 hours. The NRC staff concludes that this proposed required action is consistent with the action provided in the sample BWR-4 TSs of Appendix A to the approved LTR (Reference 18). Reducing power level to less than 18 percent RTP will place the plant into a condition to which LCO 3.3.1.1 does not apply for OPRM upscale functions because Function 2.f of Table 3.3.1-1 is only required at power levels greater than or equal to 18 percent RTP. This is consistent with Section 8 of the approved LTR. These required actions are, therefore, acceptable.

- TS 3.3.1.1, Surveillance Requirement (SR) 3.3.1.1.17

This SR is deleted. The licensee stated that this requirement is no longer required by the proposed DSS-CD solution. 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 18). This change is consistent with the sample BWR-4 TSs of Appendix A and with Section 8 of the approved LTR. This deletion is also reflected in proposed Function 2.f of Table 3.3.1.1-1. The NRC staff concludes that the deletion of this SR is acceptable.

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

The allowable value for Function 2.b in Table 3.3.1.1-1, "Flow Biased Simulated Thermal Power – High" is changed from $\leq 0.55W + 65.5$ percent reactor thermal power (RTP) to $\leq 0.61W + 68.3$ percent RTP. In addition, a note (e) is added 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 core power level and recirculation drive flow. These revised setpoints were calculated in accordance with M+ SAR Section 5.3.1 and are

consistent with Section 5.3.1 of the approved M+LTR. The OPRM functions are described in the TS Basis Section 2.f, "Oscillation Power Range Monitor (OPRM) Upscale," contained within Attachment 3 to the LAR (Reference 1). The changes to the OPRM upscale function settings and the addition of note (e) are consistent with the sample BWR-4 TSs and TS Bases of Appendix A and with Section 8 of the approved LTR. The NRC staff concludes that these changes are, therefore, acceptable.

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

The specified condition associated with Function 2.f of Table 3.3.1.1-1 is changed from "Mode 1" to greater than or equal to (\geq) 18 percent 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 pass completely through the DSS-CD armed region.

SR 3.3.1.1.17, which required a periodic verification that the OPRM is not bypassed when the average power range monitor (APRM) simulated thermal power is \geq 23 percent and recirculation flow is less than ($<$) 60 percent is deleted. This SR is no longer required because DSS-CD functions automatically arm when pre-determined conditions are met.

Footnote (e), "Refer to COLR for OPRM period-based detection algorithm (PBDA) setpoint limits," was deleted due to implementation of DSS-CD in accordance with the approved LTR.

The change to the specified condition for Function 2.f is consistent with the requirement that the DSS-CD must be operable above a power level 5 percent below the lower RTP boundary of the DSS-CD armed region. Since the lower boundary of the DSS-CD armed region is 23 percent, as defined by the MCPR threshold power level in TS 3.2.2, the NRC staff concludes that this revised condition of \geq 18 percent RTP is acceptable. The NRC staff also concludes that the addition of footnote (f) is consistent with the sample BWR-4 TSs of Appendix A and with Section 8 of the approved LTR and is, therefore, acceptable.

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

Item a.4 of this TS is replaced to reflect new COLR setpoint requirements associated with the DSS-CD reactor trip function. The NRC staff concludes that this change is consistent with the sample BWR-4 TSs of Appendix A and with Section 8 of the approved LTR (Reference 18) and is, therefore, acceptable.

Subsection b of this TS is revised to reflect the change in the approved analytical method associated with the DSS-CD methodologies. The PBDA will no longer be credited in the safety analysis. A new reference to the DSS-CD LTR is included with this change. The NRC staff confirmed that the correct reference to approved methodology was provided; therefore, this change is acceptable.

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

A new TS is added to stipulate when a report required by Condition I, Required Action 1.3, of LCO 3.3.1.1, RPS Instrumentation, shall be submitted and what the contents of this report shall be.

The NRC staff concludes that this new TS is consistent with the sample BWR-4 TSs of Appendix A and with Section 8 of the approved LTR and is, therefore, acceptable.

3.2.5 BFN M+SAR Section 2.5, "Reactivity Control"

Reactivity control is addressed generically following the approach in the M+LTR. Core reactivity is controlled by positioning neutron absorbing control rods within the reactor and scrambling the reactor by rapidly inserting control rods into the core. No change is made to the control rods or drive system due to MELLLA+.

Control Rod Scram

The hydraulic control unit accumulators supply the initial scram pressure and, as the scram continues, the reactor becomes the primary source of pressure to complete the scram. [[

]] The BFN reactor dome pressure is 1,050 pounds per square inch absolute (psia) (1,035 pounds per square inch gauge (psig)) and does not change as a result of MELLLA+ operating domain expansion. Therefore, the generic disposition is applicable to BFN such that [[

]]

Control Rod Drive Positioning, Cooling, and Integrity

As far as control rod positioning and cooling are concerned, [[

]] As a result, there is no increase in temperature, and the small drop in drive water pressure is compensated for by automatic adjustment of the CRD system flow control valve. Therefore, there is no impact of MELLLA+ on the CRD positioning and cooling functions.

Consistent with the generic disposition described above, the reactor coolant temperature does not increase.

The licensee stated in M+SAR (Attachment 6 to the LAR (Reference 1)) that the postulated abnormal operating conditions for the CRD design assume a failure of the CRD system pressure regulating valve that applies the maximum pump discharge pressure to the CRD mechanism internal components. This postulated abnormal pressure bounds the ASME reactor overpressure limit. The MELLLA+ operating domain expansion does not affect the CRD pump discharge pressure and the corresponding stress for the limiting CRD components. Because of the postulated abnormal conditions for the CRD design that bound the local loads and stresses experienced by the CRD system, the NRC staff determined that no further evaluation of CRD integrity is required as a result of MELLLA+ operation at the BFN units.

3.2.6 BFN AMSAR Section 2.6, "Additional Limitations and Conditions Related to Reactor Core and Fuel Performance"

This section is discussed in AMSAR (Attachment 8 to the LAR (Reference 1)) Section 2.6. The continued applicability of AREVA methods to MELLLA+ EPFOD operation is addressed for BFN in ANP-2860NP, Revision 2, Supplement 3NP, Revision 2 (Attachment 28 to the LAR

(Reference 1)). This evaluation concludes that the NRC-approved AREVA methodology remains within its approval basis and continues to meet all applicable SER L&Cs for the expanded operating domain. When the SAFLIM3D safety methodology was introduced in BFN TS 5.6.5.b for TS Amendment No. 285 for Unit 1, Amendment No. 311 for Unit 2, and Amendment No. 270 for Unit 3 (Reference 35), the following license condition was documented in SER Section 4.0:

The fuel channel bow standard deviation component of the channel bow model uncertainty used by ANP-10307PA, AREVA MCPR Safety Limit Methodology for Boiling Water Reactors (i.e., TS 5.6.5.b.10) to determine the Safety Limit Minimum Critical Power Ratio shall be increased by the ratio of channel fluence gradient to the nearest channel fluence gradient bound of the channel measurement database, when applied to channels with fluence gradients outside the bounds of the measurement database from which the model uncertainty is determined.

This license condition addresses a staff concern that few channels may exhibit fluence gradients exceeding the measurement values. This license condition is not affected by MELLLA+ EPFOD implementation and will continue to be effective.

A regulatory commitment was added during the addition of the RODEX4 fuel T-M methodology to Section 5.6.5.b of the BFN TSs:

When using AREVA topical report, BAW-10247PA, "Realistic Thermal-Mechanical Fuel Rod Methodology for Boiling Water Reactors," Revision 0, February 2008 to determine core operating limits, the fuel cladding peak oxide thickness calculated by the RODEX4 corrosion model will be limited to less than the proprietary value defined in Section 3.2.7 of AREVA report, ANP-3159P, Revision 0, dated October 2012.

This commitment is not affected by MELLLA+ EPFOD implementation and will continue to be implemented as part of the reload analysis process.

3.3 BFN M+SAR and AMSAR Section 3.0, "Reactor Coolant and Connected Systems"

3.3.1 BFN M+SAR Section 3.1, "Nuclear System Pressure Relief and Overpressure Protection"

3.3.1.1 BFN M+SAR Section 3.1.1, "Flow-Induced Vibration"

As discussed in M+SAR Section 3.1.1, the licensee confirmed that the generic disposition in the M+LTR for the flow-induced vibration (FIV) topic is applicable for BFN. Specifically, MELLLA+ operation does not increase main steam line (MSL) flow. As such, there is no effect on the FIV experienced by piping and safety relief valves (SRVs) during normal operation.

The NRC staff concludes that the generic M+LTR disposition is applicable to BFN because FIV of piping and the SRVs is unaffected by operation in the MELLLA+ operating domain.

3.3.1.2 BFN M+SAR and AMSAR Section 3.1.2, "Overpressure Relief Capacity"

The pressure relief system prevents overpressurization of the nuclear system during AOOs, the plant ASME upset overpressure protection event, and postulated ATWS events. The licensee stated that the limiting overpressure event is the main steam isolation valve closure with scram on high flux. The peak RPV bottom head pressure (1,350 psig) remains less than the ASME limit of 1,375 psig. The peak RPV dome pressure (1,319 psig) remains less than the ASME limit of 1,325 psig.

BFN determined that the ASME overpressure event met the acceptance criteria. Additionally, the ATWS analysis discussed in Section 9.3.1 of the AMSAR concludes that no increase in the number of 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 analyses.

3.3.2 BFN M+SAR Section 3.2, "Reactor Vessel"

The licensee, in M+SAR Section 3.2, confirmed that the generic disposition in the M+LTR for the reactor vessel structural evaluation topic is applicable to BFN. Specifically, MELLLA+ operation does not change any of the parameters (reactor operating pressure, FW flow or steam flow rate, or other applicable mechanical loads) that affect the stress or fatigue for the reactor vessel components. As such, there is no change to the stress or fatigue for reactor vessel components. Therefore, the M+LTR concludes that there is no change in the stress or fatigue for the reactor vessel components as a result of MELLLA+.

Further, plant-specific evaluations for EPU were performed by the licensee for FW nozzle fatigue life and effects of environmental fatigue on the RPV components that demonstrated BFN compliance with the applicable ASME Code requirements for a 60-year plant life. These evaluations are applicable for the MELLLA+ operating domain expansion because there are no changes in operating parameters such as operating pressure, FW temperature and flow, reactor coolant chemistry parameters, or applicable mechanical loads.

The NRC staff concludes that the generic M+LTR disposition is applicable to BFN because the stress and fatigue of reactor vessel components is unaffected by operation in the MELLLA+ operating domain.

3.3.2.1 BFN M+SAR Section 3.2.1, "Reactor Pressure Vessel Fracture Toughness"

Regulatory Evaluation

Appendix G of 10 CFR Part 50, "fracture toughness," requirements are to ensure the structural integrity of the ferritic components of the RCPB during any condition of normal operation, including AOOs and hydrostatic tests. Regulatory requirements for RPV Charpy upper shelf energy (USE) and pressure-temperature (P-T) limits are specified in Sections IV.A.1 and IV.A.2 of the rule. The NRC staff's detailed regulatory and technical evaluations related to the RPV Charpy USE and P-T limit requirements and their application to BFN Units 1, 2, and 3 are provided in Section 2.1.2 of the NRC staff's SE for the August 14, 2017, issuance of Amendment Nos. 299, 323, and 283 (Reference 16), which authorized implementation of the

EPU for BFN Units 1, 2, and 3. All NRC guidance documents and NRC-approved methodologies that are currently implemented for demonstrations of compliance with Charpy USE and P-T limits requirements in 10 CFR Part 50, Appendix G, as described in Section 2.1.2 of the staff's SE for the EPU, remain applicable for the BFN Units 1, 2, and 3 MELLLA+ application.

Technical Evaluation

For ferritic materials that are part of the RPV beltline region, 10 CFR Part 50, Appendix G, specifies that the values of the reference temperature (RT_{NDT}) and Charpy USE must account for the effects of neutron radiation, including the results of the RPV surveillance program required by 10 CFR Part 50, Appendix H.

To account for the effects of neutron embrittlement on RPV beltline materials, Regulatory Guide (RG) 1.99, Revision 2 (Reference 36), specifies methods for calculating projected values of percentage drop in Charpy USE and adjusted RT_{NDT} (ART) due to accumulated neutron fluence exposure. As shown in the RG, for a given set of RPV beltline material properties, including copper (Cu) content, nickel (Ni) content, and initial (unirradiated) RT_{NDT} , the projected value of ART increases as a function of projected neutron fluence exposure, directly resulting in a rightward shift of the P-T limits for the limiting RPV beltline materials to more bounding values. Specifically, higher values of projected neutron fluence exposure result in greater embrittlement of RPV beltline materials. It is, therefore, necessary to operate the RCS at higher temperatures and/or lower pressures (i.e., below and/or to the right of the shifted P-T limits) to ensure that the RPV beltline material fracture toughness and pressure loadings are maintained at the required levels to protect the RPV from non-ductile fracture. Conversely, for a given set of licensed P-T limits and RPV material properties, a decrease in the projected neutron fluence exposure (due to a change in the analyzed rate of neutron fluence accumulation for the RPV beltline materials), will result in the current licensed P-T limits (which were previously calculated for higher fluence values), providing greater safety margin for the protection of the RPV against non-ductile fracture in accordance with 10 CFR Part 50, Appendix G requirements.

With respect to Charpy USE, the projected percentage drop in Charpy USE increases as a function of projected neutron fluence exposure, as shown in RG 1.99, Revision 2. For cases where initial (unirradiated) Charpy USE values are available for RPV beltline materials, projected values of Charpy USE decrease with increasing neutron fluence. For many BWRs, including BFN Units 1, 2, and 3, measured values of initial USE are not available for the RPV beltline materials, and therefore, these plants implement the NRC-approved generic equivalent margin analyses (EMAs) provided in EPRI BWR Vessel and Internals Project (BWRVIP) Topical Report BWRVIP-74-A (Reference 37), (Reference 38), and (Reference 39)), to demonstrate compliance with Charpy USE requirements in 10 CFR Part 50, Appendix G. The BWRVIP-74-A report provides EMA acceptance criteria for plant-specific projections of percentage drop in Charpy USE, which are calculated per RG 1.99, Revision 2. As with P-T limits, higher projected values of neutron fluence result in less margin for application of the generic BWRVIP-74-A EMAs, whereas lower projected values of neutron fluence result in greater margin for application of these EMAs.

Plant-Specific Impact of the MELLLA+ on RPV Beltline Fracture Toughness

For the BFN Units 1, 2, and 3 MELLLA+ application, the NRC staff identified that the projected neutron fluence decreases relative to the CLTP neutron fluence for the CLTP set of RPV beltline materials; the CLTP RPV beltline materials were previously evaluated for the EPU, which was

approved with the issuance of Amendment Nos. 299, 323, and 283 (Reference 16). For MELLLA+ conditions, the staff confirmed that additional RPV plates (intermediate shell plates) were appropriately added to the set of RPV beltline materials for BFN Units 2 and 3 [I

]]. The staff verified that the addition of these plates to the set of RPV beltline materials for MELLLA+ does not impact the calculation of any RPV beltline ART values for defining the bounding P-T limits, nor does it impact the limiting RPV beltline materials for application of the BWRVIP-74-A EMA acceptance criteria for MELLLA+ conditions. Other than the intermediate shell plates for BFN Units 2 and 3, the staff verified that all other RPV beltline materials at BFN show a decrease in projected neutron fluence for MELLLA+.

The NRC staff also identified that the RPV beltline material property inputs into the fracture toughness analyses are consistent with those approved for the 2017 EPU amendments; therefore, they are acceptable. The staff confirmed that there have been no BWRVIP integrated surveillance program (ISP) surveillance capsules pulled for the BFN units since the issuance of the EPU amendments for BFN in August 2017; thus, all applicable ISP surveillance data remains consistent with what the staff reviewed for its approval of the 2017 EPU.

The NRC staff reviewed the impact the RPV beltline neutron fluence decrease on the CLTP analyses related to RPV fracture toughness requirements summarized above. The staff noted that the licensee's M+SAR includes plant-specific calculations of projected ART and calculations of projected percentage drop in USE based on the lower neutron fluence for the CLTP beltline materials. The staff verified that the projected values of ART and projected percentage drop in USE for MELLLA+ conditions were correctly determined in accordance with applicable regulatory positions in RG 1.99, Revision 2. The staff confirmed that MELLLA+ results in a decrease in these projected embrittlement parameters based on the decrease in neutron fluence for the CLTP beltline materials. Since lower projected values of ART would result in less bounding P-T limits for RPV beltline materials, the staff determined that the decrease in the projected ART values associated with MELLLA+ for the CLTP beltline materials provides assurance that the current BFN Units 1, 2, and 3 TS P-T limits would be more bounding and provide greater safety margin for MELLLA+ conditions. Similarly, the staff determined that the decrease in the projected percentage drop in USE for the RPV beltline materials due to the lower projected neutron fluence provides assurance of greater safety margin for the application of generic EMA acceptance criteria from BWRVIP-74-A. Therefore, the NRC staff finds that the current TS P-T limits and application of BWRVIP-74-A EMAs will remain valid for meeting regulatory requirements in 10 CFR Part 50, Appendix G, for MELLLA+ conditions at BFN Units 1, 2, and 3.

Conclusion to Fracture Toughness

The NRC staff concluded that all CLTP analyses related to RPV fracture toughness requirements in 10 CFR Part 50, Appendix G, would continue to remain valid for the remaining terms of the renewed facility operating licenses for BFN Units 1, 2, and 3 for MELLLA+ conditions.

3.3.2.2 BFN M+SAR Section 3.2.2, "Reactor Internals Structural Evaluation for Faulted Conditions"

The structural evaluation of reactor internals for faulted conditions is based on the applicable loads, load combinations, and service conditions consistent with the plant design basis for each

component. [[

]].

The M+SAR thus showed that operation in the MELLLA+ domain does not increase the load. Therefore, the NRC staff concludes that the reactor internals are structurally acceptable for faulted conditions.

3.3.3 BFN M+SAR Section 3.3, "Reactor Internals"

3.3.3.1 BFN M+SAR Section 3.3.1, "Reactor Internal Pressure Differences"

Fuel Assembly Lift Forces

As discussed in M+SAR (Attachment 6 to the LAR (Reference 1)) Section 3.3.1.1, the licensee confirmed that the generic disposition in the M+LTR for the fuel assembly lift forces topic is applicable to BFN. 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 CF. [[

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There is no increase in maximum CF in the MELLLA+ operating domain. As such, the lift forces for normal, upset, emergency, and faulted conditions in the MELLLA+ operating domain remain bounded by the current licensed operating domain.

The NRC staff concludes that the generic M+LTR disposition is applicable to BFN because there is no increase in maximum CF. Therefore, operation in the MELLLA+ domain is bounded by current plant operation with respect to fuel assembly lift forces.

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

The licensee confirmed that the generic treatment in the M+LTR for the RIPDs topic is applicable to BFN. 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 CF. Maximum CF is reduced in the MELLLA+ operating domain. As such, the RIPDs for normal, upset, emergency, and faulted conditions in the MELLLA+ operating domain remain bounded by the current licensed operating domain.

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

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

The faulted acoustic and flow-induced loads in the RPV annulus on jet pump, core shroud, access hole cover, and core shroud support resulting from the recirculation line break LOCA have been considered in the BFN evaluation. [[

]]

The NRC staff concludes that operation in the MELLLA+ domain expansion does not increase the loads in the RPV annulus on the jet pumps, core shroud, access hole cover, and core shroud support.

3.3.3.2 BFN M+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. [[

]]

Reactor Internals Structural Evaluation for Faulted Conditions

The structural evaluation of reactor internals for faulted conditions is based on the applicable loads, load combinations, and service conditions, consistent with the plant design basis for each component. [[

]]

The M+SAR thus showed that operation in the MELLLA+ domain does not increase the load. Therefore, the NRC staff concludes that the reactor internals are structurally acceptable for faulted conditions.

3.3.3.3 BFN M+SAR 3.3.3, "Steam Separator and Dryer Performance"

M+SAR (Attachment 6 to the LAR (Reference 1)) Section 3.3 addresses steam separator and steam dryer performance. The evaluation of steam separator and dryer performance at MELLLA+ conditions by the licensee indicates an increase in moisture carryover (MCO) will occur. The BFN plant-specific evaluation concluded that the performance of the steam dryer and separator remains acceptable (i.e., $MCO \leq 0.10$ wt%) and the dryer skirt remains covered at L4, the low water level alarm in the MELLLA+ region.

The steam dryer is a safety-significant component that is required to maintain its structural integrity so that no loose parts are generated. Any loose parts from the steam dryer can affect safety-related components in performing their safety function. In Section 3.4.2 of the M+SAR (Attachment 6 to the LAR (Reference 1)), the steam dryer is addressed. [[

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In response to an RAI (Reference 4), TVA provided the minimum alternating stress ratio (MASR) at the limiting or maximum stress location due to high cycle fatigue for the replacement steam dryers (RSDs) for the combined EPU and MELLLA+ operating conditions. Here, MASR is the ratio of alternating stress amplitude of 13,600 psi at 1,011 cycles from the ASME Code for austenitic stainless steel to maximum computed alternating stress intensity in the steam dryer.

[[

]] The analysis results for both the PBLE01 and PBLE02 methods were summarized in the final EPU reports (GEH Reports: NEDC-33894P, Revision 0, and NEDC-33895P, Revision 0, Enclosures 1 and 2 to the TVA letter dated October 24, 2018 (Reference 40).

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Core Flow Adjustment Factor		
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The minimum alternating stress ratio (MASR) for RSD at the maximum stress location is shown below.

Minimum Alternating Stress Ratio (MASR) for RSD				
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]]

The NRC staff concludes that the MASR > 1.0 meets the acceptance criteria, and therefore, the RSDs for the BFN units will maintain their structural integrity at the combined EPU and MELLLA+ operating conditions.

3.3.4 BFN M+SAR Section 3.4, "Flow-Induced Vibration"

3.3.4.1 BFN M+SAR Section 3.4.1, "FIV Influence on Piping"

As discussed in M+SAR (Attachment 6 to the LAR (Reference 1)) Section 3.4.1, the licensee confirmed that the generic disposition in the M+LTR for the FIV influence on the piping topic is applicable to BFN. Specifically, there are no increases in the flow rates in the recirculation system piping, main steam (MS) piping, and FW piping as a result of operation in the MELLLA+ operation, as compared to current plant operation. As such, there is no increase in FIV in these piping systems, including safety-related thermowells and probes.

The NRC staff concludes that the generic M+LTR disposition is applicable to BFN because operation in the MELLLA+ domain is bounded by current plant operation with respect to FIV influence on piping.

3.3.4.2 BFN M+SAR Section 3.4.2, "Flow Induced Vibration Influence on Reactor Internals"

As discussed in M+SAR (Attachment 6 to the LAR (Reference 1)) Section 3.4.2, the licensee confirmed that the generic disposition in the M+LTR for the FIV influence on reactor internals topic is applicable to BFN. Specifically, the MELLLA+ operating domain does not result in increased core and recirculation flow. There is also 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 disposition is applicable to BFN because operation in the MELLLA+ domain is bounded by current plant operation with respect to FIV influence on reactor internals. Note: The NRC staff evaluation with respect to steam dryer performance is contained in SE Section 3.3.3.3.

3.3.5 BFN M+SAR Section 3.5, "Piping Evaluation"

3.3.5.1 BFN M+SAR Section 3.5.1, "Reactor Coolant Pressure Boundary Piping"

3.3.5.1.1 BFN M+SAR Section 3.5.1.1, "Main Steam and Feedwater Piping Inside Containment"

As discussed in M+SAR (Attachment 6 to the LAR (Reference 1)) Section 3.5.1.1, the licensee confirmed that the generic disposition in the M+LTR for the MS and FW piping inside containment topic is applicable to BFN. Specifically, MS and FW system temperatures, flows, and pressures at MELLLA+ operating conditions are bounded by the current plant operation temperatures, flows, and pressures. The piping and support loads due to MS isolation valve (MSIV) closure and turbine stop valve (TSV) closure evaluated by the EPU analysis are bounded by MELLLA+ conditions.

The NRC staff concludes that the generic M+LTR disposition is applicable to BFN because operation in the MELLLA+ domain is bounded by current plant operation with respect to MS and FW piping and the MS connected branch piping (reactor core isolation cooling (RCIC) and high pressure reactor injection (HPCI) steam lines) inside containment.

3.3.5.1.2 BFN M+SAR Section 3.5.1.2, "Reactor Recirculation and Control Rod Drive Systems"

As discussed in M+SAR (Attachment 6 to the LAR (Reference 1)) Section 3.5.1.2, the licensee confirmed that the generic disposition in the M+LTR for the reactor recirculation and CRD systems topic is applicable to BFN. Specifically, there is no change in the maximum operating system temperatures, pressures, and flows in the MELLLA+ operating domain. As such, piping stresses, support loads, and RPV nozzle loads are unaffected by MELLLA+.

The NRC staff concludes that the generic M+LTR disposition is applicable to BFN because operation in the MELLLA+ domain is bounded by current plant operation with respect to the reactor recirculation and CRD systems.

3.3.5.1.3 BFN M+SAR Section 3.5.1.3, "Other RCPB Piping Systems"

As discussed in M+SAR (Attachment 6 to the LAR (Reference 1)) Sections 3.5.1.3.1 through 3.5.1.3.5, the licensee confirmed that the generic disposition in the M+LTR for the RCPB piping systems topic is applicable to BFN. Specifically, the temperatures, flows, and pressures for these systems (RPV head vent line, MS SRV discharge lines, reactor water cleanup (RWCU), and safety-related thermowells) 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 the generic M+LTR disposition is applicable to BFN because operation in the MELLLA+ domain is bounded by current plant operation with respect to the other RCPB piping systems.

3.3.5.2 BFN M+SAR Section 3.5.1.4, "Other-Than-Category 'A' RCPB Material"

The licensee stated that as required by M+LTR SER (Reference 41) L&C 12.9, the other-than-Category "A" materials that exist in the RCPB piping are evaluated based on NUREG-0313, Revision 2 (Reference 42), "Technical Report on Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping."

The licensee stated that the BFN RCPB weldments have been categorized and inspected in accordance with NUREG-0313 and BWRVIP-75-A (Reference 43), "BWR Vessel and Internals Project, Technical Basis for Revisions to Generic Letter 88-01 Inspection Schedules," Electrical Power Research Institute (EPRI), TR-1012621, dated October 2005.

The NRC staff notes that for intergranular stress corrosion cracking (IGSCC) to occur in the RCPB piping, three conditions must be present: (1) a susceptible material, (2) the presence of residual or applied tensile stress, and (3) an aggressive environment. The licensee has applied several IGSCC mitigation processes to reduce the RCPB components' susceptibility to IGSCC.

To address item (1), the licensee indicated that the design of the RCPB piping has been modified to reduce the amount of installed IGSCC susceptibility material. Table 3-8 of M+SAR (Attachment 6 to the LAR (Reference 1)) lists the materials used in the RCPB piping. The licensee stated that to address IGSCC in RCPB piping, BFN was designed, fabricated, and constructed by one of three methods: (1) corrosion resistant materials, (2) solution heat treatment, or (3) clad with resistant materials. The licensee stated that for the weldments where these three processes were not used, it applied stress improvement processes to reduce IGSCC susceptibility. The licensee stated that stress improvement processes and original construction processes used for IGSCC resistance are not affected by MELLLA+. The NRC staff finds that there is no change due to MELLLA+ on the susceptibility of RCPB piping material to IGSCC.

To address item (2), the licensee stated that implementation of MELLLA+ will not increase temperature or flow within the RCPB piping and does not affect the other susceptibility factors associated with IGSCC. The NRC staff finds that if temperature and flow will not increase, the pipe stresses will not increase, and thereby, the RCPB piping will not be affected.

To address item (3), the licensee has implemented a hydrogen water chemistry (HWC) program with noble metals, which reduces the potential for IGSCC of RCPB components in BFN. The

licensee indicated that all BFN units have implemented the NobleChem™ (NMCA) process, including the annual on-line NobleChem™ injection process. The licensee used the NobleChem™ processes in conjunction with HWC injection to FW to achieve IGSCC mitigation of reactor piping and internals at lower FW hydrogen addition rates than would be required for mitigation strategies that employ an HWC program only (e.g., moderate HWC). The licensee indicated that implementation of these programs at the BFN units after MELLLA+ implementation will continue to be performed in accordance with the recommendations of the applicable EPRI BWRVIP guidelines and experience reports (i.e., BWRVIP-190 (Reference 44), “BWR Vessel and Internals Project, BWR Water Chemistry Guidelines – 2014 Revision,” EPRI, 2008; BWRVIP-118 (Reference 45), “BWR Vessel and Internals Project, NMCA Experience Report and Applications Guidelines, 2003 Revision,” EPRI TR-1003193, 2003; BWRVIP-159 (Reference 46), “BWR Vessel and Internals Project, HWC/NMCA Experience Report and NMCA Applications Guidelines,” EPRI TR-1013397, 2006; and BWRVIP-219 (Reference 47), “BWR Vessel and Internals Project, Technical Basis for On-Line NobleChem™ Mitigation and Effectiveness Criteria for Inspection Relief,” EPRI TR-1019071, 2009).

In addition to the programs to address the three factors that cause IGSCC, the licensee has implemented a program to monitor IGSCC mitigation based on catalyst loading, electrochemical potential, and hydrogen and oxygen concentration. The licensee has implemented an inservice inspection (ISI) program for RCPB piping in accordance with the requirements of the ASME Code, Section XI, and coupled with the augmented program for reactor coolant piping based on GL 88-01 (Reference 48), “NRC Position on IGSCC In BWR Austenitic Stainless Steel Piping,” dated January 25, 1988, NUREG-0313, and BWRVIP-75-A.

The NRC staff finds that the licensee has implemented programs to mitigate and manage the potential for IGSCC. The NRC staff further determines that the existing ISI program is adequate to monitor any potential IGSCC effects on the RCPB piping. The NRC staff further determines that plant operation under MELLLA+ conditions will not likely increase IGSCC within the RCPB because the licensee has implemented appropriate degradation management programs to address any potential effects on the structural integrity of RCPB materials under MELLLA+ conditions. The NRC staff finds that RCPB materials will continue to be acceptable following implementation of the proposed MELLLA+ and will continue to meet the requirements of GDC 1, 2, 14, and 31; and 10 CFR Part 50, Appendix G. Therefore, the staff concludes that the proposed MELLLA+ is acceptable with respect to RCPB materials.

3.3.5.3 BFN M+SAR Section 3.5.2, “Balance-of-Plant Piping”

3.3.5.3.1 BFN M+SAR Section 3.5.2.1, “Main Steam and Feedwater (Outside Containment)”

As discussed in M+SAR (Attachment 6 to the LAR (Reference 1)) Section 3.5.2.1, the licensee confirmed that the generic disposition in the M+LTR for the MS and FW outside containment topic is applicable to BFN. Specifically, MS and FW system temperatures, flows, and pressures at MELLLA+ operating conditions are bounded by the current plant operation temperatures, flows, and pressures. There is no change in stress and fatigue evaluations for the piping or associated supports.

In addition, the FW piping outside containment susceptibility to erosion/corrosion does not increase since the FW flow does not increase.

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

3.3.5.3.2 BFN M+SAR Section 3.5.2.2.1, "Other BOP Piping Systems – RCIC, HPCI, CS, and RHR"

As discussed in M+SAR (Attachment 6 of (Reference 1)) Section 3.5.2.2.1, the licensee confirmed that the generic disposition in the M+LTR for the other balance-of-plant (BOP) piping systems RCIC, HPCI, core spray (CS), and residual heat removal (RHR) systems) topic is applicable to BFN. 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 BFN 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 the generic M+LTR disposition is applicable to BFN because operation in the MELLLA+ domain is bounded by current plant operation with respect to these BOP piping systems.

3.3.5.3.3 BFN M+SAR Section 3.5.2.2.2, "Other BOP Piping Systems - Offgas System, Containment Air Monitoring, and Neutron Monitoring System"

As discussed in M+SAR (Attachment 6 of (Reference 1)) Section 3.5.2.2.2, the licensee confirmed that the generic disposition in the M+LTR for the other BOP piping systems (offgas system and neutron monitoring system) topic is applicable to BFN. Specifically, there is no change in the BFN reactor operating pressure or power level at MELLLA+ operating conditions.

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

3.3.6 BFN M+SAR Section 3.6, "Reactor Recirculation System"

3.3.6.1 System Evaluation

As discussed in M+SAR (Attachment 6 of (Reference 1)) Section 3.6.1, the licensee confirmed that the generic disposition in the M+LTR for the reactor recirculation system (RRS) evaluation topic is applicable to BFN. Specifically, for BFN, there are no increases in RRS temperature, pressure, or flow rates for MELLLA+ operation, as compared to current plant operation. RRS system temperature for the current licensed operating domain is 529.8 °F; in the MELLLA+ operating domain, it is 525.7 °F. RRS system pressure for the current licensed operating domain and in the MELLLA+ operating domain is 1,050 pounds per square inch atmospheric (psia). For the proposed amendments, TS 3.4.1, "Recirculation Loops Operating," would be revised to prohibit SLO in the MELLLA+ domain.

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

3.3.6.2 Net Positive Suction Head

As stated in UFSAR (Reference 49) Section 4.3, the reactor recirculation system (RRS) pumps are single-stage, variable speed, vertical, centrifugal pumps. The pump is capable of stable and satisfactory performance while operating continuously at any speed corresponding to a power supply frequency range of 11.5 to 57.5 Hertz. The FW flowing into the reactor vessel annulus during operation cools the fluid passing to the recirculation pumps, thus increasing the net positive suction head (NPSH) beyond that provided by the pump location below the reactor vessel water level.

The M+LTR (Reference 17) states that the [[

]], the NRC staff finds that the RRS current NPSH analysis is acceptable.

3.3.6.3 Single Loop Operation

The M+LTR states that SLO is not allowed in the MELLLA+ operating domain. In Section 3.6.3 of the M+SAR (Attachment 6 to the LAR (Reference 1)), TVA states that consistent with the generic disposition in M+LTR, SLO is not allowed in the MELLLA+ operating domain, and therefore, is not affected by the MELLLA+ domain expansion. The licensee identified BFN SLO limitations in proposed TS 3.4.1.

The NRC staff concludes that BFN meets all M+LTR dispositions for SLO. Limitations proposed in TS 3.4.1 on BFN SLO are discussed in Section 4.2 of this SE.

3.3.7 BFN M+SAR Section 3.7, "Main Steam Line Flow Restrictors"

The generic disposition M+LTR of the "Main Steam Line Flow Restrictors" states that there is no increase in MS flow as a result of the MELLLA+ operating domain expansion. [[
]] and no further evaluation of this topic is required.

The licensee in M+SAR (Attachment 6 to the LAR (Reference 1)) Section 3.7 confirmed that there is no increase in BFN MS flow as a result of MELLLA+ operating domain expansion. The numerical values showing that MS flow does not increase as a result of MELLLA+ operation.

The NRC staff concludes that, since there is no expected change in MS flow, FW flow, CF or operating pressure for BFN MELLLA+, this generic resolution of M+LTR is applicable to BFN units.

3.3.8 BFN M+SAR Section 3.8, "Main Steam Isolation Valves"

The generic disposition of the MS isolation valve (MSIV) topic in the M+LTR states that there is no increase in MS pressure, flow, or pressure drop as a result of the MELLLA+ operating domain expansion. [[

]] and no further evaluation of this topic is required.

The licensee stated in M+SAR (Attachment 6 to the LAR (Reference 1)) Section 3.8 that there is no increase of MS pressure for the current licensed operating domain and in the MELLLA+ operating domain. The calculated pressure drop across MSIVs for the MELLLA+ operating domain is slightly less than the current calculated pressure drop. Therefore, the licensee confirmed that the generic M+LTR disposition of the "Main Steam Isolation Valves" is applicable to BFN units.

The NRC staff concludes that, since there is no expected change in MS flow, FW flow, CF or operating pressure for BFN MELLLA+, this generic resolution is acceptable.

3.3.9 BFN M+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.

3.3.9.1 BFN M+SAR Section 3.9.1, "System Hardware"

The licensee in M+SAR (Attachment 6 of (Reference 1)) Section 3.9.1 confirmed that consistent with the generic disposition in M+LTR there are no changes to the BFN RCIC system hardware as a result of MELLLA+.

Therefore, the NRC staff concluded that BFN meets all M+LTR dispositions for the RCIC system hardware.

3.3.9.2 BFN M+SAR Section 3.9.2, "System Initiation"

The generic disposition of the RCIC System Initiation topic in the M+LTR states that there are no changes to the normal reactor operating pressure, decay heat, or SRV setpoints as a result of MELLLA+ operating domain expansion.

The licensee in M+SAR (Attachment 6 of (Reference 1)) Section 3.9.2 confirmed that there are no changes to the normal reactor operating pressure, decay heat, or MSRVS setpoints as a result of MELLLA+ operating domain expansion. Therefore, BFN meets all M+LTR dispositions for the RCIC system initiation.

The NRC staff concludes that since BFN meets M+TLR, no further evaluation of this topic is required.

3.3.9.3 BFN M+SAR Section 3.9.3, "Net Positive Suction Head"

As stated in UFSAR (Reference 49) Section 4.7.5, the RCIC system pump is a steam turbine-driven pump capable of delivering 600 gallons per minute (gpm) makeup water to the

RPV. The water supply for the RCIC system comes from the condensate storage tank (CST), with a secondary supply from the suppression pool. The turbine-pump assembly is located below the level of the CST and below the minimum water level in the suppression pool to ensure a positive NPSH margin for the pump. The turbine steam supply comes from the MS line from the RPV. The exhaust steam from the turbine is discharged to the suppression pool. From the standpoint of available NPSH, the CST suction source is less limiting, because greater static head over that provided by the suction from the suppression pool is available, and secondly the CST is not subject to the heat addition from reactor blowdown and decay heat. During RPV isolation with the reactor shutdown, the suppression pool is heated up by the MSRVs steam and the RCIC turbine steam discharge. In the MELLLA+ operation, the maximum core power and decay heat are not increased. Consequently, the increase in the suppression pool temperature is unaffected so that its currently calculated temperature of 140 °F for which adequate NPSH is available do not change. The remaining parameters affecting the available NPSH that are: (a) minimum pressure of 14.4 psia above the suppression pool or the CST, (b) pump flow rate, and (c) pump suction piping configuration also do not change for the MELLLA+ operation.

The NRC staff, therefore, finds that the MELLLA+ operation does not affect the current evaluation of the RCIC system pump NPSH because the suppression pool temperature and the remaining parameters (a), (b), and (c) mentioned above that could affect the available NPSH do not change.

3.3.9.4 Conclusion for Section 3.3.9

The NRC staff reviewed the effect of the proposed MELLLA+ operating domain on the functional capability of the RCIC system. The system 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 the intent of GDC 4, 5, 33, and 54 continue to be satisfied.

3.3.10 BFN M+SAR Section 3.10, "Residual Heat Removal System"

The RHR system is designed to restore and maintain the reactor coolant inventory following a LOCA and remove reactor decay heat following reactor shutdown for normal, transient, and accident conditions. The RHR system is typically a low-pressure system that takes over the shutdown cooling function when the RCS temperature is reduced. The RHR system operates in various modes: low pressure coolant injection (LPCI) mode, suppression pool and containment spray cooling modes, shutdown cooling mode, steam condensing mode, and fuel pool cooling assist. 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 in LPCI mode as discussed in Section 3.4.2.6 of this SE. The RHR system in suppression pool and containment spray cooling and shutdown cooling modes meet the M+LTR generic disposition, since for BFN [[

]] and [[

]].

As discussed in Section 6.3.1 of the M+SAR (Attachment 6 to the LAR (Reference 1)), [[
]] for the BFN units in the RHR in fuel pool cooling assist mode.

3.3.11 BFN M+SAR Section 3.11, "Reactor Water Cleanup System"

3.3.11.1 System Performance

The generic disposition of the RWCU System Performance topic in the M+LTR describes that the MELLLA+ operating domain expansion does not change the pressure or fluid thermal conditions experienced by the RWCU system. Operation in the MELLLA+ operating domain does not increase the quantity of fission products, corrosion products, and other soluble and insoluble impurities in the reactor water. Reactor water chemistry is within fuel warranty and TS limits on effluent conductivity and particulate concentration, and thus, no changes will be made in water quality requirements.

The licensee in M+SAR (Attachment 6 to the LAR (Reference 1)) Section 3.11.1 stated that for BFN, there is no increase in the quantity of fission products, corrosion products, and other soluble and insoluble impurities in the reactor water. Also, there is no significant change in the FW line temperature, pressure, or flow rate. FW line temperature for the current licensed operating domain and in the MELLLA+ operating domain is approximately 394.5 °F (upstream of the RWCU return).

The NRC staff concludes that because there is no increase in the quantity of fission products, corrosion products, and other soluble and insoluble impurities in the reactor water, no further evaluation of this topic is required, and BFN meets all M+LTR dispositions for RWCU system performance.

3.3.11.2 Containment Isolation

The RWCU Containment Isolation topic in the M+LTR describes that the RWCU system is a normally operating system with no safety-related functions other than containment isolation.

[[]] because there is no change in the FW line pressure, temperature, and flow rate.

The licensee in M+SAR (Attachment 6 of (Reference 1)) Section 3.11.2 stated that consistent with the generic disposition discussed above for BFN, there is no significant change in the FW line temperature, pressure, or flow rate. FW line temperature for the current licensed operating domain and in the MELLLA+ operating domain is approximately 394.5 °F (upstream of the RWCU return).

Therefore, the NRC staff concludes that since FW system resistance and operating conditions do not change and the pressure at the RWCU/FW system interface does not change for RWCU return lines and there is no change to RWCU inlet conditions, the BFN units meets the generic disposition for RWCU containment isolation.

3.4 BFN M+SAR and AMSAR Section 4.0, "Engineering Safety Features"

3.4.1 BFN M+SAR Section 4.1, "Containment System Performance"

3.4.1.1 BFN M+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 do not exceed their design limits with the proposed change. The analysis covers the LOCA blowdown period during which the maximum drywell pressure and differential pressure between drywell and wetwell occur. This analysis is affected by any change in the mass flow rate and/or enthalpy of the break fluid. The M+SAR (Attachment 6 of (Reference 1)) Section 4.1.1 states:

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The BFN current licensing basis is the EPU. The licensing basis for the short-term containment pressure and temperature response analysis is for the limiting design-basis (DB) LOCA based on a large double-ended guillotine break of a recirculation suction line, hereinafter called recirculation suction line break (RSLB). This break resulted in the maximum peak containment pressure. This analysis is more limiting relative to the main steam line break LOCA for the containment pressure response. In this analysis, the licensee defined a Design (D) Case and a Bounding (B) Case, for the short-term analysis. These cases used the same initial drywell and wetwell pressure of 2.6 psig and 1.5 psig respectively (bounding normal operating pressures), but with different initial drywell temperatures. The initial drywell temperatures used are 70 °F for Case D and 130 °F for Case B. The licensee stated that the Case D initial drywell temperature of 70 °F is below the lowest initial drywell temperature during normal operation and is, therefore, conservative for demonstrating the maximum containment pressure response at the CLTP. The purpose of the Case D was to perform a conservative analysis to demonstrate that a DB LOCA initiated under the CLTP conditions would not challenge the containment design pressure of 56 psig. The licensee developed the Case B initial drywell temperature of 130 °F with a statistical basis to conservatively calculate the containment pressure response due to a DB LOCA at the CLTP (EPU power level). This temperature represents a lower statistical bound of the 5-year historical drywell operating temperature during normal plant operation. The containment pressure response determined from the DB LOCA using Case B conservative initial conditions is used to determine a conservative value of peak containment accident pressure (P_a) for 10 CFR Part 50 Appendix J containment leakage rate testing.

For the MELLLA+ operating domain, the current DB LOCA analysis would be affected by the change in the mass and energy (M&E) of the break fluid. As stated in Section 1.2.4 of the M+SAR (Attachment 6 to the LAR (Reference 1)), SLO and final FW temperature reduction are not allowed in the MELLLA+ operating domain. Also as stated in Section 1.2.4 of the M+SAR, a 10 °F FW operational temperature band is allowed implying that a feedwater temperature reduction of more than 10 °F is not permitted. Using the same initial pressure and temperature as in the current Cases D and B, the licensee performed the short-term containment response for the DB LOCA at 102.0 percent CLTP, 85.0 percent CF condition, and 79.6 percent CLTP, 55 percent CF conditions, which corresponds to points N and O in Figure 1-1 of the M+SAR

respectively. These points encompass the MELLLA+ operating domain. The results indicated that the peak containment pressure for the point N (102 percent CLTP, 85.0 percent CF) is the more limiting condition in the MELLLA+ domain. Table 4-1 in the M+SAR shows the comparison of results of peak containment pressure and temperature for Points N and O of MELLLA+ domain to the limiting case in the CLTP analysis for Cases B and D. These drywell pressure results for Cases B and D show that the peak short-term DB LOCA pressures for the MELLLA+ operating domain are bounded by peak pressures obtained for the CLTP DB LOCA and are below the containment design pressure of 56.0 psig. The drywell temperature results for Case B show that the peak drywell temperatures from the short-term DB LOCA for the MELLLA+ operating domain is the same for point N and bounded for point O by the peak drywell temperature at the CLTP. Based on the results, the NRC staff finds that the short-term drywell temperature and pressure response in the MELLLA+ operating domain are less limiting than the response in the currently licensed operating domain at CLTP.

In the CLTP analysis for containment equipment qualification (EQ), the small steam line break (SSLB) is the limiting event. The long-term SSLB LOCA analysis results in a significantly higher peak drywell atmosphere temperature of 336.9 °F. The elevated drywell atmosphere also lasts for a longer duration than produced by the short-term RSLB LOCA drywell temperature. In SRXB-C-RAI 1, the licensee was requested to provide the results of the drywell gas temperature response and the peak drywell shell temperature for the SSLB LOCA in the MELLLA+ operating domain. The licensee was also requested to state the analysis method and those inputs and assumptions that differed from the current analysis and justify the changes that reduced conservatism. In response to SRXB-C-RAI 1 (Reference 6), the licensee stated that the SSLB does not result in a change in the break M&E because there are no changes in the following parameters in the MELLLA+ operating conditions: (a) break sizes, (b) reactor thermal power (RTP), and (c) reactor pressure. Therefore, the drywell gas temperature and the peak drywell shell temperature responses for the current SSLB LOCA analysis remains the same for MELLLA+ operating domain.

UFSAR (Reference 49) Section 5.2.5, and BFN TS addresses the containment leakage testing requirements in accordance with 10 CFR 50, Appendix J, Option B. The Type A, B, and C tests that measure the containment integrated leak rate, containment penetration leak rates, and containment isolation valve leak rates, respectively, are performed using the TS value for P_a , which is the peak containment pressure during a design-basis LOCA. The LOCA peak drywell pressure results in the MELLLA+ operating domain show that P_a for BFN units bounded by the current TS value of 49.1 psig. Therefore, the operation in the MELLLA+ domain does not affect the leak rate testing requirements for containment penetrations and isolation valves or the containment integrated leak rate testing.

The NRC staff finds that containment peak pressure and temperature do not exceed their design limits with the proposed change, and that the short-term drywell pressure, gas temperature, and drywell shell temperature responses in the MELLLA+ operating domain are bounded by the peak drywell pressure, gas temperature, and the drywell shell temperature in the current analysis.

3.4.1.1.1 BFN M+SAR Section 4.1.1.1, "Long-Term Suppression Pool Cooling Temperature Response"

The M+LTR (Reference 17) disposition for the long-term suppression pool cooling temperature response is as follows:

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In M+SAR (Attachment 6 of (Reference 1)) Section 4.1.1.1, the licensee provided an evaluation of the long-term suppression pool temperature under the CLTP conditions operating in the MELLLA+ operating domain. The licensee stated that the sensible heat, decay heat, and the initial condition for the analysis in the MELLLA+ operating domain do not change from the current analysis; [[

]]

The NRC staff finds the licensee's evaluation of the long-term suppression pool temperature in the MELLLA+ operating domain acceptable because it is consistent with the generic disposition in the M+LTR.

Regarding the remaining long-term containment response key results, which are drywell atmosphere temperature and drywell shell temperature response, wetwell temperature and pressure response, and steam bypass capability, the licensee stated these do not change in the MELLLA+ operating domain. In SRXB-C-RAI 2, the licensee was requested to provide reasons why the above parameters are unaffected in the MELLLA+ operating domain. In response (Reference 6) to SRXB-C-RAI 2, the licensee stated that all long-term containment analysis response results remain unaffected because the containment initial conditions, RTP, sensible heat and decay heat do not change for the long-term analysis. The current containment steam bypass capability analysis remains unaffected because it depends on the RTP and the containment response to steam line breaks, which are unchanged.

The NRC staff finds that the long-term containment response and the steam bypass capability remain unaffected because the RTP, containment initial conditions, and the break sizes do not change in the MELLLA+ operating domain.

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 would rapidly reduce the containment pressure and temperature following a LOCA and maintain them at acceptable levels.

3.4.1.2 BFN M+SAR Section 4.1.2, "Containment Dynamic Loads"

3.4.1.2.1 BFN M+SAR Section 4.1.2.1, "Loss-of-Coolant Accident Loads"

The M+LTR (Reference 17) 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 50), for Mark I containments approved by NRC in NUREG-0661 (Reference 51). The BFN plant-specific loads

are defined in the plant unique load definition (PULD) report NEDO-24580, Revision 2 (Reference 52).

Vent Thrust Loads

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]] In SRXB-C-RAI 3, the licensee was requested to explain the method of calculating the MELLLA+ vent thrust loads. In response (Reference 6) to SRXB-C-RAI-3, the licensee stated that the DB LOCA short-term drywell-to-wetwell maximum differential pressure and mass flow during the vent clearing are obtained from the NRC-accepted M3CPT code (Reference 53) output. These values, along with the geometrical dimensions of the vent system, are substituted in the equations provided in Section 4.2.1 of the LDR for calculating the [[

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These calculations are performed using Microsoft Excel and are equivalent to hand calculations.

The NRC staff finds the current vent thrust loads are unaffected because they are based on the short-term containment response analysis results of the NRC-accepted M3CPT code [[]].

Pool Swell Loads

The BFN plant-specific pool swell loads are defined in NEDO-24580, Revision 2, which are based on the quarter scale test facility plant unique test report NEDE-21944-P, Volume 1 (Reference 54). These loads depend on the [[]], and the load definition is based on a DB LOCA, which is the double-ended guillotine RSLB LOCA. The current LOCA M&E release was calculated by the GEH LAMB code (Reference 55) and used as input to NRC-accepted M3CPT for calculating short-term containment pressure response and containment pressurization rate. The licensee stated that [[

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The NRC staff finds that the current pool swell load definition would be applicable in the MELLLA+ operating domain because the [[

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Condensation Oscillation Loads

The DB LOCA 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 suppression pool swell phenomena. The LDR provides the basis for the generic Mark I CO load definition. This load definition was developed from the data in the full scale test facility (FSTF) test report NEDE-24539-P (Reference 56). The tests are bounding for all US Mark I plants, including BFN.

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The NRC staff accepts that the current CO loads are not affected in the MELLLA+ domain because the [[

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Chugging Loads

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

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Regarding the chugging load tests, the licensee stated that the Mark I containment test program [[

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The current chugging loads are based on the LDR which, in turn, are in accordance with the FSTF tests (Reference 56). 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 bounds all Mark I containment plants, including the BFN MELLLA+ operating conditions.

The NRC staff accepts that the current LOCA [[

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The results of the BFN 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 that the LOCA

containment dynamic loads defined for the BFN units are not affected in the MELLLA+ operating domain.

Based on the above evaluation, under the MELLLA+ operating domain, the containment 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.

3.4.1.2.2 BFN M+SAR Section 4.1.2.2, "Subcompartment Pressurization"

The annular region between the outside vertical wall of the RPV and the inside of the sacrificial shield wall (SSW) is a containment subcompartment to be analyzed for differential pressure loads due to high energy line breaks (HELBs). The SSW is a cylindrical structure surrounding the RPV that provides thermal and radiation shielding. It is designed to withstand the maximum differential pressure that would develop across the wall as a result of any HELB between the RPV and the SSW.

As stated in Section 4.1 of the M+LTR (Reference 17), MELLLA+ operating conditions may require plant-specific subcompartment pressurization evaluation because there is a possibility of a change in the thermal-hydraulic conditions in the reactor vessel. The evaluation of the differential pressure across SSW and the SSW doors is provided below.

Sacrificial Shield Wall

Consistent with current analysis presented in Section 2.6.2 of NEDC-33860P, Revision 1 (Reference 57), for calculating the maximum differential pressure across the SSW, the licensee analyzed the double-ended break flow from the limiting 4-inch jet-pump instrument line at the safe-end weld joint located in the annulus. The calculations assumed a limiting subcooled break mass flux in the MELLLA+ operating domain, considering a minus 10 °F variation in nominal FW temperature for conservative M&E release. The results shown in M+SAR Table 4-4 show a maximum differential pressure of 3.0 psid across the SSW in the limiting MELLLA+ operating condition, which is bounded by the 3.6 psid maximum differential pressure (Section 2.6.2 in (Reference 57)) at the limiting critical break mass flux in the current analysis. The design limit for the SSW differential pressure is 19.0 psid.

The NRC staff finds the licensee's evaluation of the SSW annulus differential pressure developed due to HELB under the MELLLA+ operating condition acceptable because the licensee used conservative inputs that showed a significantly high margin between the calculated value and the design limit.

Sacrificial Shield Wall Doors

As stated in TVA response (Reference 58) to SCVB-RAI 14, the acceptance criteria for the SSW doors is that they do not fail and become missiles in the primary containment. Using the current methodology documented in that TVA response, the licensee analyzed the jet impingement force and drag load on the SSW penetration doors due to the break flow from the 4-inch jet pump instrument line with the reactor at the MELLLA+ statepoints N and O of Figure 1-1 in M+SAR. The calculation considered a minus 10 °F variation in the nominal FW temperature, which conservatively increases the fluid subcooling at the break location. The results for the limiting jet impingement force shows an increase of less than 0.1 percent (from 12,605 foot pound-force (lbf) in the current analysis to 12,615 lbf in the MELLLA+ domain) (as

provided in Table 4-5a of M+SAR). To determine the actual force applied to the door assembly, the licensee's calculation of the jet area showed that the jet radius for all break conditions is less than the distance from the jet pump nozzle centerline to the edge of the hole in the SSW door. Consequently, the jet would not impinge upon the SSW or the door assembly, and therefore, the actual force would be zero. The result of limiting SSW drag load for the 4-inch instrument line break in the MELLLA+ domain remains unchanged from the load at 102 percent of the CLTP (Table 4-5b in M+SAR). The NRC staff agrees that current structural analysis of the SSW penetration door remains unaffected because the limiting loads are unchanged in the MELLLA+ operating domain.

Based on the above evaluation, the NRC staff finds that the containment subcompartment continues to meet the requirements of GDC 4 because the dynamic loads due to LOCA subcompartment pressurization in the MELLLA+ operating domain are bounded by the current loads.

3.4.1.2.3 BFN M+SAR Section 4.1.2.3, "MSRV Piping - Containment Dynamic Loads"

The MSRV and its piping loads depend on its safety/relief valve (SRV) setpoints, reactor sensible heat and the decay heat. 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.

3.4.1.2.4 BFN M+SAR Section 4.1.2.4, "MSRV Containment Dynamic Loads"

The generic disposition in the M+LTR (Reference 17) Section 4.1 states:

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For BFN, the generic disposition 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 disposition in the M+LTR is applicable, which states that operation in the MELLLA+ domain does not affect the SRV discharge loads on the containment.

3.4.1.3 BFN M+SAR Section 4.1.3, "Containment Isolation"

The generic disposition in the M+LTR Section 4.1.3 states:

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plant-specific evaluation is required to demonstrate the adequacy of the containment isolation system.

As stated in Section 3.4.1.1 above, the MELLLA+ containment pressure and temperature response are bounded by the current pressure and temperature response. The NRC staff finds that further evaluation of the containment isolation systems is not required. The current evaluation is acceptable in the MELLLA+ operating domain.

3.4.1.4 BFN M+SAR Section 4.1.4, "Generic Letter 89-10"

The evaluation under GL 89-10, Supplement 3 (Reference 59), "Consideration of the Results of NRC-Sponsored Tests of Motor-Operated Valves" would be affected by the containment pressure and temperature under DBA conditions. [[

]] The licensee also confirmed that other parameters such as environment temperature during normal conditions and under HELB conditions, with the exception of RWCU system HELBs that could potentially affect the safety-related motor-operated valves (MOVs), are not changed in the MELLLA+ operating domain. The licensee's analysis for the area temperatures resulting from the RWCU system HELBs at MELLLA+ are bounded by the current maximum environment temperatures for GL 89-10 MOVs. Therefore, the NRC staff finds that the current evaluation under GL 89-10 is acceptable in the MELLLA+ operating domain.

3.4.1.5 BFN M+SAR Section 4.1.5, "Generic Letter 89-16"

The licensee has installed hardened containment vent systems (HCVSs) in all BFN units in response to GL 89-16 (Reference 60). The HCVS would function to relieve the wetwell gas space pressure to the atmosphere during a severe accident sequence resulting from a transient followed by a total loss of decay heat removal. In response to NRC Order EA-13-109 (Reference 61), the licensee has modified Units 1, 2, and 3 HCVS for a capacity of 1 percent of the rated CLTP of 3,952 MWt as described in the licensee's letter to the NRC dated June 7, 2019 (Reference 62). The vent capacity is consistent with the guidance in Nuclear Energy Institute (NEI) 13-02 (Reference 63), endorsed by NRC interim staff guidance (Reference 64). Since the reactor thermal power does not change in the MELLLA+ operating domain, the NRC staff finds the modified HCVS capacity under the CLTP condition for BFN Unit 1, 2, and 3 is acceptable in the MELLLA+ operating condition.

3.4.1.6 BFN M+SAR Section 4.1.7, "Generic Letter 95-07"

The evaluation under GL 95-07 (Reference 65), "Pressure Locking and Thermal Binding of Safety-Related Power-Operated Gate Valves," would be affected by the containment pressure and temperature under DBA conditions. The MELLLA+ operating domain does not impact the current evaluation under GL 95-07 [[

]] The NRC staff finds the current evaluation under GL 95-07 acceptable in the MELLLA+ operating domain.

3.4.1.7 BFN M+SAR Section 4.1.7, "Generic Letter 96-06"

The evaluation under GL 96-06 (Reference 66), "Assurance of Equipment Operability and Containment Integrity during Design-Basis Accident Conditions," 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 [[

]] The NRC staff finds the current evaluation under GL 96-06 acceptable in the MELLLA+ operating domain.

3.4.2 BFN M+SAR Section 4.2, "Emergency Core Cooling System"

Section 4.2.1, "High Pressure Coolant Injection"; Section 4.2.3, "Core Spray"; Section 4.2.4, "Low Pressure Coolant Injection"; and Section 4.2.5, "Automatic Depressurization System," of the M+SAR (Attachment 6 of (Reference 1)) Section 4.2, "Emergency Core Cooling Systems," are addressed generically following the approach in the M+LTR. M+LTR Section 4.2.2, "High Pressure Core Spray," is not applicable to BFN. The NRC staff reviewed the licensee's justification for use of the generic disposition to ensure BFN MELLLA+ plant conditions and associated analysis fall within the bounds of generic disposition. The NRC staff concludes that, since there is no expected change in operating pressure, decay heat, and SRV setpoints for BFN MELLLA+, the generic disposition is acceptable and the Sections 4.2.1 through 4.2.5 M+SER evaluation is applicable to this application. The ECCS performance is also discussed in Section 3.4.3 of this SE.

3.4.2.6 BFN M+SAR Section 4.2.6, "ECCS Net Positive Suction Head"

The RHR system in its low-pressure coolant injection (LPCI) mode and the core spray (CS) system are part of the ECCS. The RHR system in its suppression pool cooling or containment spray mode is used for containment heat removal. The events in which these systems perform these safety functions are: (a) LOCA, (b) SBO, (c) NFPA-805 fire, (d) stuck open relief valve, (e) loss of RHR normal shutdown cooling, and (f) shutdown and cooldown of the non-accident units during a DB LOCA (in a single unit), concurrent with the loss-of-offsite power (LOOP) and the loss of an emergency diesel generator (EDG). The RHR and CS pumps that draw water from the suppression pool require adequate NPSH be available for their acceptable operation. Assuming atmospheric pressure above the suppression pool surface (i.e. zero credit for the containment accident pressure (CAP)), the available NPSH for these pumps depends on the vapor pressure at the suppression pool temperature, static head at the pump inlet, and head loss in the suction strainer and the inlet piping, which depend on the pump flow rate and the inlet piping configuration. The MELLLA+ operating domain does not result in an increase in core power and decay heat or heat addition to the suppression pool during these events; therefore, the suppression pool temperature response is not affected. There are no physical changes in the RHR and CS system suction strainer and piping or arrangement that could impact static head at the pump inlet and the head loss in the inlet piping. The pump flow rate is also not affected in the MELLLA+ operating domain because there is no change in operator actions for throttling the RHR and CS flows during these events. In the current licensing basis, the licensee does not credit CAP for calculating the available NPSH for these pumps, and no CAP credit is needed for the MELLLA+ operation. Therefore, the NRC staff finds that the MELLLA+ operation does not affect the current NPSH evaluation of the RHR and CS pumps during the above events.

3.4.2.6.1 BFN M+SAR Section 4.2.6.1, "ECCS NPSH During ATWS"

Consistent with the current licensing basis, only RHR pumps are used for the suppression pool cooling during an ATWS event in the MELLLA+ operating domain. Therefore, the RHR pump NPSH is evaluated for the ATWS event in the MELLLA+ operating domain.

Without CAP credit, the NPSH available (NPSHA) at the suction inlet of the RHR pump depend on the suppression pool temperature response, static head at the pump inlet, and head loss in the suction strainer and the inlet piping. The NPSH margin is the difference between NPSHA and the NPSH required (NPSHR) at a given flow rate at the pump installation site, denoted as NPSHR effective (NPSHReff). The value of the NPSHReff is the sum of the pump vendor provided NPSHR 3 percent (NPSHR3%) and the uncertainty due to its field installation. The NPSHR3% is defined by the Hydraulic Institute standard as the tested value of the NPSH at the pump inlet, coincident with a 3-percent reduction in pump dynamic head at a given flow during the shop testing of the pump. NRC guidance in Enclosure 1 of SECY-11-0014 (Reference 67) allows assumption of zero percent uncertainty for special (non-design-basis) events that include ATWS.

The licensee performed the MELLLA+ ATWS analyses for the containment response using the NRC-approved code ODYN to determine the heat addition to the suppression pool from MSRV flow and STEMP to determine the suppression pool heatup due to energy input from the MSRVs. ODYN and STEMP are the current licensing basis codes for this application as documented in LDR NEDO-21888, Revision 2 (Reference 50).

The analysis for the pump suction static head used the transient water level height above the pump suction centerline obtained from the suppression pool temperature response.

The head loss in the suction strainer and the pump inlet piping depend on the system flow rate and the suction piping configuration. For conservatism, the licensee assumed a system flow rate that meets or exceeds the RHR pump flow requirements for operation during an ATWS event. The licensee increased the flow by a factor of $1/\sqrt{0.97}$ (1.015) to account for the reduction in pump flow rate associated with a 3-percent reduction in pump total developed head. In SRXB-C RAI-5, the licensee was requested to clarify the use of factor 1.015 by which the RHR pump flow is increased. In its response (Reference 6), the licensee stated that this factor accounts for the reduction in pump flow rate associated with a 3-percent reduction in pump total developed head. The NRC staff finds the licensee's response acceptable because it conforms to the NRCs guidance in Sections 6.3.6 and 6.6.2 of SECY-11-0014, Enclosure 1 (Reference 67).

For calculation of the head loss in the suction piping, the hydraulic analysis modeled the entire torus, torus ring header, and suction piping network and determined the strainer and suction piping friction losses for each RHR pump. The suction strainer debris loading is not considered because high energy fluid release from the RPV to the suppression pool is from the MSRV through its discharge line and T-quenchers below the suppression pool water level. There are no unpiped MSRVs discharging directly into the drywell. For conservatism, the licensee used the largest calculated RHR pump suction piping and strainer friction loss in the determination of NPSHA.

The most limiting ATWS event with respect to the suppression pool temperature response is determined to be the ATWS LOOP at the end-of-cycle (EOC) event. In this event, two RHR pumps and heat exchangers are used for suppression pool cooling. The licensee's NPSH analysis show the least NPSH margin of 15.1 feet for the ATWS LOOP EOC event and 16.6 feet for the ATWS pressure regulator failure open (PRFO) EOC event, without crediting CAP.

According to the guidance in SECY-11-0014, Enclosure 1, Section 6.3.3, for minimizing erosion the RHR pump should not operate for more than a limited time in the zone of maximum erosion. The margin ratio NPSHA/NPSHR3% is between 1.2 and 1.6 in this zone. The licensee's

calculation shows the ratio is 1.9 for the ATWS LOOP EOC event and 2.0 for the ATWS PRFO EOC event. The ratio being always greater than 1.6 is consistent with guidance in Enclosure 1 of SECY-11-0014.

In SRXB-C-RAI 6, the licensee was requested to clarify Note 2 in Table 4-6 of the M+SAR (Attachment 6 to the LAR (Reference 1)). The note states that the MELLLA+ "ATWS non-LOOP PRFO EOC" and "ATWS LOOP at EOC" results for peak suppression pool temperature are lower than the EPU results due to the use of more current ATWS analysis. The staff requested details on revised ATWS modeling and its differences/comparison with the EPU modeling, including the methodologies used. In response to SRXB-C-RAI 6 (Reference 6), the licensee stated the change is the result of an update to the reactor level control modeling procedure in ODYN, implemented for the MELLLA+ ATWS PRFO and ATWS main steam isolation valve closure (MSIVC) events. This update to the reactor level control modeling procedure had previously been implemented for the BFN CLTP LOOP event ATWS analysis (Reference 57).

Table 3.4.2-1 below shows the ATWS peak suppression pool temperature (SPT) results for CLTP (Reference 57) and M+SAR (Attachment 6 to the LAR (Reference 1)) for the LOOP, PRFO, and MSIVC events. It is noted that the LOOP results are slightly higher for MELLLA+, and the PRFO and MSIVC results are significantly lower for MELLLA+ than the results for CLTP. The change in the PRFO and MSIVC results is an unexpected trend and is the result of an update to the reactor level control modeling procedure in ODYN, which was implemented for the BFN MELLLA+ PRFO and MSIVC event ATWS analysis. This update to the reactor level control modeling procedure had previously been implemented for the BFN CLTP LOOP event ATWS analysis.

Table 3.4.2-1: BFN ATWS Peak SPT Results for EPU and MELLLA+ for LOOP, PRFO, and MSIVC Events

Event	CLTP (EPU) Peak SPT (°F)	MELLLA+ Peak SPT (°F)
LOOP	173.3	174.5
PRFO	171.8	164.4
MSIVC	171.8	164.0

The NRC staff finds the NPSH analysis of the RHR pump in the MELLLA+ ATWS event acceptable because by using conservative input parameters for calculating the NPSHA and NPSH margin, a significant amount of NPSH margin is present. The RHR pump operating time in the zone of maximum erosion is zero, which is consistent with the guidance in SECY-11-0014, Enclosure 1.

The NRC staff finds that the NPSH analysis for ATWS in the MELLLA+ operating domain is consistent with the M+LTR NRC SER (Reference 17) L&C 12.17 regarding evaluation of the RHR system performance during the long-term cooling phase of an ATWS in terms of available NPSH. The NRC staff also finds that the conditions in L&C 12.18d regarding equipment operability in terms of NPSH are met.

3.4.2.6.4. Section 4.2.6 Summary

The M+SAR sections reviewed in this SE are acceptable for BFN Units 1, 2, and 3 in the CLTP MELLLA+ operating domain. The following are the NRC staff's conclusions regarding the conformance to the requirements of the applicable GDC in the MELLLA+ operating domain:

The requirements of GDC 4 will continue to be met. The proposed change does not affect the protection of containment SSCs against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids that may result from equipment failures and from events and conditions outside the nuclear power unit.

The requirements of GDC 16 will continue to be met. The proposed change does not compromise the effectiveness of the containment in serving as a barrier to fission product release following an accident or special events. The safety analyses performed in support of this LAR demonstrate that acceptable containment performance will be maintained post-LOCA and special events. In addition, the containment will continue to be tested as specified in 10 CFR Part 50, Appendix J, and TS requirements.

The requirements of GDC 38 will continue to be met. The containment heat removal system will continue to perform its safety-related function to rapidly reduce the containment pressure and temperature following a DB LOCA and maintain them at acceptably low levels.

The requirements of GDC 50 will continue to be met. The overall function of the containment system will be maintained following a DB LOCA. The existing design-basis limits regarding post-accident containment pressure and temperature will not be exceeded. In addition, the containment design leakage rate as specified in the TSs will not be exceeded. The input assumptions inherent in the calculated margin of the overall containment system continue to remain valid.

3.4.3 BFN AMSAR Section 4.3, "Emergency Core Cooling System Performance"

The design of the ECCS is to provide protection against postulated LOCAs caused by ruptures in the primary system piping. The ECCS performance under all LOCA conditions and the analysis models must satisfy the requirements of 10 CFR 50.46 and 10 CFR Part 50, Appendix K, for this MELLLA+ application. The licensee used the NRC-approved EXEM BWR 2000 evaluation model for the LOCA analysis. The evaluation model is comprised of the following codes and methods:

- EMF-2361(P)(A), Revision 0 (Reference 68), "EXEM BWR-2000 ECCS Evaluation Model"
- XN-CC-33(A), Revision 1 (Reference 69), "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 (Reference 70), "Exxon Nuclear Company ECCS Cladding Swelling and Rupture Model"
- XN-NF-81-58(P)(A), Revision 2, and Supplements 1 and 2 (Reference 71), "RODEX2 Fuel Rod Thermal-Mechanical Response Evaluation Model"

The methodology used is unchanged from that used in the EPU analysis. The analysis was performed using an equilibrium core of ATRIUM-10 XM fuel at EPU and MELLLA+ conditions. There are four primary sections in the LAR addressing ECCS performance:

- Section 4.3 of the AMSAR (Attachments 7 (proprietary) and 8 (nonproprietary) to the LAR (Reference 1))
- Browns Ferry LOCA Break Spectrum Analysis, ANP-3546P, Revisions 0 (Attachments 11 (proprietary) and 12 (nonproprietary) to the LAR (Reference 1))
- Browns Ferry LOCA-ECCS Analysis MAPLHGR Limits for ATRIUM 10XM Fuel, ANP-3547P, Revision 0 (Attachments 13 (proprietary) and 14 (nonproprietary) to the LAR (Reference 1))
- Browns Ferry LOCA-ECCS Analysis MAPLHGR Limits for ATRIUM-10 Fuel, ANP-3548P, Revision 0 (Attachments 15 (proprietary) and 16 (nonproprietary) to the LAR (Reference 1))

Two major steps were performed in the LOCA analysis: (1) a break spectrum analysis to determine the limiting break and limiting single failure and 2) a MAPLHGR analysis to demonstrate the MAPLHGR limit is adequate to ensure compliance with 10 CFR 50.46(b).

The NRC staff reviewed the BFN MELLLA+ LOCA analysis to ensure the methodology was applied appropriately, there were no unexpected results in the MELLLA+ operating domain, and the results meet the 10 CFR 50.46 acceptance criteria. As more fully discussed below, the NRC staff concluded that the methodology was applied appropriately, there were no unexpected results in the MELLLA+ operating domain, and the analytical results meet the 10 CFR 50.46 acceptance criteria.

Break Spectrum Analysis

The break spectrum analysis was completed under EPU and MELLLA+ operating conditions for an equilibrium ATRIUM 10XM core. In Section 2.0 of ANP-3546NP (Attachment 12 to the LAR (Reference 1)), the licensee stated that the break spectrum results are applicable to ATRIUM 10XM fuel and other co-resident fuel assemblies since the thermal-hydraulic characteristics will be similar. The NRC staff questioned what other co-resident fuel types were bounded by the ATRIUM 10XM fuel. In response to SRXB-RAI 10 (Reference 6), the licensee stated the only other co-resident fuel type covered by this analysis was ATRIUM-10. The staff determined that the thermal-hydraulic characteristics between ATRIUM-10 and ATRIUM 10XM are similar enough such that only one break spectrum analysis is needed. Also, this approach was previously approved in the EPU analysis, and operation in MELLLA+ will not invalidate that approach in EPU.

A number of potential limiting single-failure events were evaluated in the break spectrum analysis, as documented in ANP-3546NP (Attachment 12 to the LAR (Reference 1)). The licensee performed each break size and single failure at CFs of [[]] to bound the range of allowable CFs at 100 percent EPU power. The licensee also evaluated the [[]]

]]. The limiting single failure for two recirculation loop operation and recirculation line breaks is in Table 3.4.3-1 below.

Table 3.4.3-1: BFN Limiting LOCA Break Characteristics and Results

Limiting LOCA Break Characteristics and Results	
Location	Recirculation discharge pipe
Type/size	Split/0.23 ft ²
Single failure	Battery (DC) power, board A
Axial power shape	Top-peaked
Initial State	102% power, [[]]
PCT	2030 °F

Additionally, the licensee evaluated single recirculation pump operation and breaks in the non-recirculation line. All these other runs were bounded by the results in Table 3.4.3-1 above. Since the peak cladding temperature (PCT) is relatively high compared to the 2,200 °F acceptance criteria in 10 CFR 50.46, the NRC staff requested additional sensitivity studies at 100 percent EPU power at a point between 85 and 100 percent flow. In SRXB-RAI 9 response (Reference 6), the licensee provided results of a sensitivity for full power at [[]].

]]. The predicted PCT at this statepoint is [[]], which is less than the value in Table 3.4.3-1.

The limiting break characteristics and results are the same as in the EPU analysis. The NRC staff determined this result is not unexpected. Variations in CF at the same power level would primarily impact the early PCT peaks due to potentially earlier boiling transition; however, a second later in transient, PCT peak is limiting for BFN. This second peak is more sensitive to plant ECCS configuration, single failure, and break size, whereas CF has a much smaller impact.

The NRC staff determined that the break spectrum analysis considered the appropriate break sizes, break locations, single failures, and initial statepoints (i.e., 100 percent EPU power and various CFs). The licensee provided appropriate coverage for calculating a number of postulated LOCAs of different sizes, locations, and other properties sufficient to provide assurance that the most severe postulated LOCAs are calculated. Therefore, the NRC staff determined that the break spectrum results are acceptable.

MAPLHGR Analysis

Using the limiting break size and location identified in ANP-3546P, the licensee performed an MAPLHGR analysis for both ATRIUM 10XM and ATRIUM-10 fuel in ANP-3547NP (Attachment 14 to the LAR (Reference 1)) and ANP-3548NP (Attachment 16 of LAR (Reference 1)), respectively. The purpose of this analysis is to specify the MAPLHGR limit versus exposure for each fuel type and demonstrate compliance with the 10 CFR 50.46(b) acceptance criteria. The NRC staff compared the MELLLA+ LOCA analysis to the EPU LOCA analysis and determined that the same approach was used. Therefore, the staff concluded that the licensee has appropriately applied the methodology for the MAPLHGR analysis.

MAPLHGR values were developed as a function of exposure for ATRIUM 10XM and ATRIUM-10 in Figure 2.1 of both ANP-3547P and ANP-3548P (Attachments 14 and 16 to the

LAR (Reference 1), respectively). The limiting results for the analysis are shown in Table 3.4.3-2 below.

Table 3.4.3-2: BFN Limiting LOCA MAPLHGR Analysis Results

Fuel	Parameter	Results ¹	Exposure (GWd/MTU)
ATRIUM-10	PCT	2086 °F	0
	Local Cladding Oxidation (LCO)	3.29%	25
ATRIUM 10XM	PCT	2008 °F	0
	LCO	2.16%	25

¹ All runs resulted in less than 1 percent core wide oxidation.

These results meet the 10 CFR 50.46 acceptance criteria and demonstrate the adequacy of the BFN ECCS, and therefore, the NRC staff finds them acceptable.

Additionally, the NRC staff reviewed the key figures of merit of the limiting break (Figure 5.1 through Figure 5.19 in ANP-3547P for ATRIUM 10XM and Figure 5.1 through Figure 5.19 in ANP-3548P for ATRIUM-10). Since the break spectrum results were the same as the EPU results, and the MAPLHGR limits are unchanged from the EPU analysis, the NRC staff compared these results to the EPU results. The NRC staff did not identify any unexpected behavior in the MELLLA+ results in the comparison to the EPU results. Therefore, the staff determined that these results are acceptable.

Effects of Fuel Thermal Conductivity Degradation

The licensee analyzed the impact of fuel thermal conductivity degradation (TCD). The RODEX2 code used in the LOCA analysis does not account for the effects of TCD. To assess the TCD impact, the licensee evaluated the impacts of TCD (i.e., differences in stored energy) with RODEX4 code, which accounts for TCD. The licensee then adjusted the RODEX2 input based on the differences between the RODEX4 code results and used it as input in the LOCA calculations. The results in Table 5.2 of both ANP-3547P and ANP-3548P estimate the impacts of TCD. This approach to address TCD was 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.

Applicable Limitation and Conditions

The NRC staff reviewed the application to ensure that the methodology was not impacted by BFN MELLLA+ conditions, the appropriate initial conditions were used, the methodology was adequately applied, and to ensure compliance with the L&Cs. The following L&Cs from the M+LTR are applicable to the BFN LOCA analysis:

- 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

The NRC staff determined that all L&Cs were adequately addressed (or not applicable as appropriate) as discussed in Appendices A and B of this SE.

Section 3.4.3, Conclusions

The NRC staff reviewed the BFN MELLLA+ LOCA analysis to ensure the methodology was applied appropriately, there were no unexpected results in the MELLLA+ operating domain, and the results met the 10 CFR 50.46 acceptance criteria. As discussed in this section, the staff determined that the licensee adequately applied the methodology, particularly for the break spectrum analysis and the MAPLHGR analysis, no unexpected results were observed, and the results meet the 10 CFR 50.46 acceptance criteria.

Subsequent to the NRC staff review as described above, TVA informed the NRC that Framatome identified two errors in the computer codes that were used for analyses associated with AREVA fuels, which may have impacted the BFN MELLLA+ LAR. By letter dated December 19, 2019 (Reference 13), TVA provided a description of these errors and their impacts on the MELLLA+ analyses.

The errors include [[

]]

The NRC staff evaluated the impact of the two errors in the codes and determined that the 10 CFR 50.46 criteria is not challenged, and the errors do not significantly impact the steady

state, transient, and accident analyses. Additionally, the licensee plans to implement a license condition that would ensure the use of corrected codes during the reload safety analyses. In addition, in the letter dated December 19, 2019, TVA proposed a license condition for the implementation of MELLLA+ for each unit (see Section 4.1 of this SE for specific information).

3.4.4 BFN M+SAR Section 4.4, "Main Control Room Atmosphere Control System"

The licensee confirmed that the generic M+LTR treatment of the iodine intake topic is applicable to BFN. Specifically, the licensee stated that there is no change in the BFN source term or release rates as a result of MELLLA+ operating domain expansion. [[

]]

The NRC staff concludes that the generic M+LTR treatment is applicable to BFN because operation in the MELLLA+ operating domain is bounded by the current plant operation with respect to the main control room atmosphere control system.

Based on the above evaluation, the requirement of GDC 19 continues to be met because the main control room atmosphere control system, which 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.

3.4.5 BFN M+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. The licensee confirmed that the generic M+LTR treatment is applicable to BFN.

In the MELLLA+ operating domain, the [[

]]

In the MELLLA+ operating domain, the [[

]]

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 requirement of GDC 41 as it relates to reducing 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.

3.4.6 BFN M+SAR Section 4.6, "Main Steam Isolation Valve Leakage Control System"

BFN does not have an MSIV leakage control system.

3.4.7 BFN AMSAR Section 4.7, "Post-LOCA Combustible Gas Control System"

The NRC revised 10 CFR 50.44, "Combustible gas control for nuclear power reactors," in September 2003. The revised rule eliminates the requirements for maintaining hydrogen and oxygen control equipment associated with a DBA and relaxes the requirements for hydrogen and oxygen monitoring in containment. Based on the adoption of revised 10 CFR 50.44, the Mark I containments, such as BFN, are required to be inerted during normal operation to prevent an uncontrolled combination of hydrogen and oxygen in the containment during severe post-accident conditions. The requirement of GDC 41 as it relates to the control of hydrogen or oxygen concentration in the containment following postulated accidents is to assure that containment integrity is maintained.

The licensee in Section 4.7 of AMSAR (Attachment 8 of (Reference 1)) stated that the combustible gas control system at BFN is designed to ensure an inert atmosphere in the drywell and wetwell is maintained after a postulated LOCA by injecting nitrogen into the drywell and wetwell to keep the oxygen concentration below 5 percent by volume. The licensee added that the two primary factors that would impact the combustible gas control system are the cladding mass and the reactor thermal power, which are not impacted by extended power/flow operating domain (EPFOD). The licensee concluded that MELLLA+ has no effect on the post-LOCA combustible gas control system. Therefore, the NRC staff finds the combustible gas control system at BFN meets the requirements of 10 CFR 50.44 and GDC 41 under the MELLLA+ domain.

3.5 BFN M+SAR and AMSAR Section 5.0, "Instrumentation and Control"

The following is a summary of the licensee's generic MELLLA+ dispositions for BFN for the topics in Section 5.0 of the M+SAR (Attachments 5 (proprietary) and 6 (nonproprietary) to the LAR (Reference 1)) and the AMSAR (Attachments 7 (proprietary) and 8 (nonproprietary) to the LAR (Reference 1)).

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

Section 5.1 of the BFN M+SAR (Attachment 6 to the LAR (Reference 1)) describes changes to process parameters resulting from the MELLLA+ operating domain expansion and their effects on instrument performance. These changes include APRMs, intermediate range monitors (IRMS) and source range monitors (SRMs), LPRMs, rod block monitors (RBMs), rod worth minimizers (RWMs), and TIPs.

In the M+SAR (Attachment 6 in the (Reference 1)), the licensee stated that expansion into the MELLLA+ domain does not change the maximum core power for BFN, and therefore, the changes to the instrumentation and control (I&C) systems can be generically dispositioned by the M+LTR (Reference 17). The NRC staff reviewed the licensee's justification provided in the M+SAR and determined, as discussed in the following sections, that the generic disposition for the I&C systems is acceptable.

Changes to the TSs that resulted from the expansion into the MELLLA+ domain are described in Section 3.5.3 of this SE.

3.5.1.1 BFN M+SAR Section 5.1.1, "Average Power Range, Intermediate Range, and Source Range Monitors"

In BFN M+SAR Section 5.1.1, the licensee confirmed that the generic disposition in the M+LTR for the APRMs, IRMs, and SRMs topic is applicable to BFN. The APRM output signals are calibrated to read 100 percent at the CLTP. [[

]] Using normal plant surveillance procedures, the IRMs may be adjusted to ensure adequate overlap with the APRMs.

Based on the information provided in the M+SAR, the NRC staff concludes that the generic M+LTR disposition is applicable to BFN because there is no change in the core power as a result of MELLLA+ operating domain expansion.

3.5.1.2 BFN M+SAR Section 5.1.2, "Local Power Range Monitors"

In M+SAR Section 5.1.2, the licensee stated that there is no change in the neutron flux experienced by the LPRMs resulting from operating in the MELLLA+ domain. As such, [[

]] Therefore, the generic disposition in the MELLLA+ LTR for the LPRMs topic is applicable to BFN.

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

3.5.1.3 BFN M+SAR Section 5.1.3, "Rod Block Monitors"

The RBM uses LPRM instrumentation inputs that are combined and referenced to an APRM channel. [[

]]

The NRC staff concludes that the generic M+LTR disposition is applicable to BFN because [[

]]

3.5.1.4 BFN M+SAR and AMSAR Section 5.1.4, "Rod Worth Minimizer"

The RWM supports the operator by enforcing rod patterns until reactor power has reached appropriate levels. [[

]] In the M+SAR (Attachment 6 to the LAR (Reference 1)), the licensee referred to the AMSAR (Attachment 8 to the LAR (Reference 1)) for this evaluation, which was provided in Attachments 7 (proprietary) and 8 (nonproprietary) to the LAR. In the AMSAR, Section 5.1.4, the licensee evaluated the generic disposition of the RWM in the M+LTR. In this evaluation, the licensee confirmed that the RWM system at BFN is consistent with the generic disposition of the M+LTR.

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

3.5.1.5 BFN M+SAR Section 5.1.5, "Traversing Incore Probes"

In the M+SAR (Attachment 6 to the LAR (Reference 1)) Section 5.1.5, the licensee confirmed that the generic disposition in the M+LTR for the TIPs topic is applicable to BFN. There is no change in neutron flux experienced by the TIPs by MELLLA+ operation.

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

3.5.2 BFN M+SAR Section 5.2, "BOP Monitoring and Control"

In BFN M+SAR Section 5.2, the licensee indicated that operation in the MELLLA+ domain has no effect on the BOP instrumentation and control systems described in Sections 5.2.1 through 5.2.6 of the M+SAR, and therefore, the generic disposition in the M+LTR for these systems is applicable to BFN. 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 disposition is applicable to BFN because BOP monitoring and control devices are unaffected by operation in the MELLLA+ operating domain.

3.5.3 BFN M+SAR Section 5.3, "Technical Specification Instrumentation Setpoints"

Section 5.3 of the BFN M+SAR describes changes to instrumentation setpoints resulting from the MELLLA+ operating domain expansion. The affected instrumentation setpoints are associated with the APRM flow-biased scram and rod block functions. These setpoints were calculated in accordance with the methodology approved in Amendment Nos. 257, 296, and 254 for BFN Units 1, 2, and 3, respectively, dated September 14, 2006 (Reference 72).

To support operations within the MELLLA+ region, BFN is implementing a DSS-CD safety function. This function is being performed in accordance with the approved GEH Boiling Water Reactor Detect and Suppress Solution – Confirmation Density Licensing Topical Report (Reference 18). In addition, implementing the MELLLA+ domain requires changes to the operating power/CF map and changes to a small number of instrument setpoints included in the proposed TS changes.

The following TS changes related to instrumentation are proposed in the LAR. These changes support implementation of the DSS-CD approach to automatically detect and suppress neutronic/thermal-hydraulic instabilities.

TS 3.3.1.1, Reactor Protection System (RPS) Instrumentation

- TS 3.3.1.1, Required Action I

Required Action I.1 is being changed from a single action to initiate an alternate method of detecting and suppressing TH instability oscillation to three separate actions as follows:

- I.1 Initiate action to implement the Manual Backup Stability Protection (BSP) Regions defined in the COLR. (Completion Time: Immediately)

AND

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

AND

- I.3 Initiate action to submit an OPRM report in accordance with Specification 5.6.7. (Completion Time: Immediately)

The NRC staff compared these required actions to those in the approved DSS-CD LTR "GE Hitachi Boiling Water Reactor Detect and Suppress Solution – Confirmation Density Licensing Topical Report" (Reference 18) and determined these changes are consistent with 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 TSs of Appendix A of the approved LTR. The sample TSs within LTR Appendix A are applicable to BFN Units 1, 2, and 3 because they are GE Type 4 BWR plants. These required actions are, therefore, acceptable.

- TS 3.3.1.1, Required Action J

Required Action J.1 is being changed from one action to three actions, as follows:

- J.1 Initiate action to implement the Manual BSP Regions defined in the COLR. (Completion Time: Immediately)

AND

- J.2 Reduce operation to below the BSP Boundary defined in the COLR. (Completion Time: 12 hours)

AND

- J.3 Note: LCO 3.0.4 is not applicable.

Restore required channel to OPERABLE status. (Completion Time: 120 days)

This change addresses the situation where required action and associated completion time of Condition I are not being met. The NRC staff determined that these proposed required actions are also the same actions provided in the sample BWR-4 TSs of Appendix A to the approved LTR and are consistent with Section 8, "Effect on Technical Specifications," of the approved DSS-CD LTR. These required actions are, therefore, acceptable.

- TS 3.3.1.1, Condition K (new)

A new required Condition K and Required Action K.1 are being added to address the situation where the completion time of Condition J is not met. The condition is:

- K. Required Actions and associated Completion Time of Condition J not met.

The required action is:

- K.1 Reduce THERMAL POWER to < 18% RTP. (Completion Time: 4 hours)

Reducing power level to less than 18 percent RTP will place the plant into a condition in which LCO 3.3.1.1 does not apply for OPRM upscale functions because Function 2.f of TS Table 3.3.1-1 is only required at power levels greater than or equal to 18 percent RTP. This is consistent with Section 8, "Effect on Technical Specifications," of the approved DSS-CD LTR (Reference 18). The NRC staff also determined that this proposed required action is consistent with the action provided in the sample BWR-4 TSs of Appendix A to the approved DSS-CD LTR. This modification is, therefore, acceptable.

- TS 3.3.1.1, Surveillance Requirement (SR) 3.3.1.1.17

This SR is being deleted. 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 DSS-CD LTR (Reference 18). This change is consistent with the sample BWR-4 TSs of Appendix A and with Section 8, "Effect on Technical Specifications," of the approved LTR. The NRC staff agrees deletion of this SR is acceptable.

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

The allowable value for Function 2.b, "APRM Flow Biased Simulated Thermal Power – High," in Table 3.3.1.1-1, "Flow Biased Simulated Thermal Power – High," is being changed from " $\leq 0.55W + 65.5\%$ Reactor Thermal Power (RTP)" to " $\leq 0.61W + 68.3\%$ RTP." In addition, a new note (e) is added to implement automated backup stability region setpoints when Function 2.f is inoperable. The new note is:

- (e) With OPRM Upscale (Function 2.f) inoperable, the Automated BSP Scram Region setpoints are implemented in accordance with Action I of this Specification.

The proposed allowable value reflects the changed curve for determining the simulated thermal power (STP) trip setpoint based on core power level and recirculation drive flow. In the LAR, the licensee stated that the revised setpoint was calculated in accordance with the methodology approved in Amendment Nos. 257, 296, and 254 for BFN Units 1, 2, and 3, respectively, dated September 14, 2006 (Reference 72). Therefore, the NRC staff did not review the setpoint methodology.

The proposed new note will address changes to the allowable value resulting from the implementation of the automated BSP scram region.

The change to the allowable value is consistent with Section 5.3.1 of the approved M+LTR (Reference 15). The changes to the OPRM upscale function settings and the addition of note (e) are consistent with the sample BWR-4 TSs and TS Bases of Appendix A and with Section 8, "Effect on Technical Specifications," of the Approved DSS-CD LTR (Reference 18). The NRC staff agrees that these changes are, therefore, acceptable.

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

This function is revised to set the new operability power level for OPRM upscale and to add a new note due to implementation of the DSS-CD stability solution. Specifically, the applicable mode associated with Function 2.f of Table 3.3.1.1-1 is being changed from "Mode 1" to \geq "18% RTP." For BFN, the DSS-CD should be operable above 18 percent, which corresponds to the power level set in the DSS-CD LTR (Reference 18).

A new footnote (f) is being 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. This note states:

- (f) Following Detect and Suppress Solution – Confirmation Density (DSS-CD) implementation, DSS-CD is not required to be armed while in the DSS-CD Armed Region during the first reactor startup and 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.

Current footnote (e) will be removed to reflect the change from PBDA to CDA. The existing requirement is no longer applicable since the PBDA is no longer credited in the safety analysis.

The reference to SR 3.3.1.1.17 will be deleted from the SRs column due to implementation of DSS-CD, which is automatically armed.

These changes to Function 2.f are consistent with the requirement that the DSS-CD must be operable above a power level 5 percent below the lower RTP boundary of the DSS-CD armed region (i.e., 18 percent for BFN). These changes are consistent with the sample BWR-4 TSs of Appendix A and with Section 8, "Effect on Technical Specifications," of the approved LTR. The NRC staff finds these changes acceptable.

TS 3.4.1, Recirculation Loops Operating

- LCO 3.4.1

A new note is added to the LCO to prohibit operation in the MELLLA+ region with a single recirculation loop (SLO) in operation. 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 LCO reinforces the requirement that two recirculation loops with matched flows shall be in operation when the reactor is operating in the MELLLA+

region. This revised LCO is acceptable because the licensee is not requesting to operate the plant in the MELLLA+ domain during SLOs. The new note is:

-----NOTE-----

Single recirculation loop operation is prohibited in the MELLLA+ operating domain.

A new Condition B with Action B.1 is being added to LCO 3.4-1 and a completion time to TS 3.4.1 to reflect the fact that SLO is prohibited in the MELLLA+ region (Reference 4) and to require immediate exit from the MELLLA+ domain if one SLO is in operation. The new condition is:

- B. Operation in the MELLLA+ operating domain with a single recirculation loop in operation.

The new required action is:

- B.1 Initiate action to exit the MELLLA+ operating domain. (Completion Time: Immediately)

The existing Condition B is renumbered to Condition C, and the reference to "Condition A" is changed to "Condition A or B."

These changes are consistent with the premise that single recirculating loop operation is not allowed when the reactor is in the MELLLA+ operating domain. These changes are consistent with the DSS-CD LTR (Reference 18), and therefore, are acceptable.

TS 5.6.5, Core Operating Limits Report

- TS 5.6.5, Subsection a

Revise Administrative Control TS 5.6.5 subsection a.(4) to reflect implementation of DSS-CD stability solution. The revised subsection a.(4) is:

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

Item a.(4) of this TS is being replaced to reflect new COLR setpoint requirements associated with the DSS-CD reactor trip function. This change is consistent with the sample BWR-4 TSs of Appendix A and with Section 8, "Effect on Technical Specifications," of the approved DSS-CD LTR (Reference 18) and is, therefore, acceptable.

- TS 5.6.5, Subsection b

Add reference to the GE NEDC-33075P-A, Revision 8, due to implementation of DSS-CD. The revised subsection b is:

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

A new reference to the DSS-CD LTR (Reference 18) is included to provide the generic licensing basis for DSS-CD applications. The NRC staff confirmed the correct reference to the approved methodology was provided; therefore, this change is acceptable.

TS 5.6.7, Oscillation Power Range Monitor Report (new)

A new section is added to this TS to stipulate when an OPRM report required by Required Action I.3 of TS 3.3.1.1, RPS Instrumentation, be submitted and what the contents of this report will be.

This new TS is consistent with the Sample BWR-4 TSs of Appendix A and with Section 8 of the approved DSS-CD LTR and is, therefore, acceptable.

3.5.3.1 BFN M+SAR Section 5.3.1, "APRM Flow-Biased Scram"

In Section 5.3.1 of the M+SAR, the licensee stated that [[

]] In addition, the licensee confirmed that the evaluation of APRM flow-biased STP scram setpoints is consistent with the methods described for plant-specific evaluations of this topic in the M+LTR. Section 3.1 of this SE provides the technical evaluation for the proposed allowable value for the APRM flow-biased scram.

The NRC staff concludes that the generic M+LTR disposition is applicable to BFN because [[

]] In addition, the NRC staff notes that since BFN cannot operate with SLO in the MELLLA+ domain, the licensee did not modify the SLO setpoint for this function.

3.5.3.2 BFN M+SAR Section 5.3.2, "Rod Block Monitor"

In M+SAR Section 5.3.2, the licensee confirmed that [[

]] In addition, the licensee confirmed that the generic disposition in the M+LTR for the RBM topic is applicable to BFN.

The NRC staff concludes that the generic M+LTR disposition is applicable to BFN because [[

3.6 BFN M+SAR Section 6.0, "Electrical Power and Auxiliary Systems"

3.6.1 BFN M+SAR Section 6.1, "AC Power"

The alternating current (AC) power supply includes both offsite and onsite power. The onsite power distribution system consists of transformers, buses, and switchgear. AC power to the distribution system is provided from the transmission system or from onsite diesel generators.

The M+LTR (Reference 17) describes that there is no change in the thermal power from the reactor or the electrical output from the station that results from the MELLLA+ operating domain expansion. [[

]]

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

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

]]

3.6.2 BFN M+SAR Section 6.2, "Direct Current (DC) Power"

The direct current (DC) power distribution system provides control and motive power for various systems/components within the plant. The M+LTR (Reference 17) describes that the MELLLA+ operating domain expansion does not change system requirements for control or motive power loads. [[]].

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

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

3.6.3 BFN M+SAR Section 6.3, "Fuel Pool"

3.6.3.1 BFN M+SAR Section 6.3.1, "Fuel Pool Cooling"

The licensee confirmed that the generic M+LTR (Reference 17) treatment of the spent fuel pool is applicable to BFN. Specifically, reactor power does not increase as a result of MELLLA+ operation. [[

]]

Therefore, the NRC staff concludes that the generic M+LTR resolution is applicable to BFN because fuel pool cooling [[

]]

3.6.3.2 BFN M+SAR Section 6.3.2, "Crud Activity and Corrosion Products"

The M+LTR (Reference 17) describes that [[

]] No further evaluation of the crud and corrosion products in the spent fuel pool is required for MELLLA+ operating domain expansion.

The licensee confirmed that [[
]] Therefore,
the NRC staff concludes that that the generic M+LTR resolution is applicable to BFN because
the BFN units [[
]]

3.6.3.3 BFN M+SAR Section 6.3.3, "Radiation Levels"

The M+LTR (Reference 17) describes that [[
]] and no further evaluation
of the radiation levels in the spent fuel pool is required for MELLLA+ operating domain
expansion.

The licensee in the M+SAR confirmed that [[
]]
Therefore, BFN meets all M+LTR dispositions for radiation levels in the spent fuel pool.

The NRC staff concludes that no further evaluation of the radiation levels in the spent fuel pool
is required for the BFN units for operation in the MELLLA+ domain.

3.6.3.4 BFN M+SAR Section 6.3.4, "Fuel Racks"

The M+LTR (Reference 17) describes that the MELLLA+ operating domain expansion does not
increase the core power level. [[
]] No further evaluation of the fuel racks is
required for MELLLA+ operating domain expansion.

The licensee confirmed in Section 6.3.4 of the M+SAR that, consistent with the generic
disposition, the MELLLA+ operating domain, expansion does not increase the BFN core power
level. Therefore, BFN meets all M+LTR dispositions for fuel racks.

The NRC staff concludes that no further evaluation of the fuel racks is required for the BFN units
for operation in the MELLLA+ domain.

3.6.4 BFN M+SAR Section 6.4, "Water Systems"

The licensee confirmed that the generic M+LTR (Reference 17) treatment of the water systems
topic is applicable to BFN. 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 BFN because the
performance of water systems is unaffected by the MELLLA+ operating domain.

3.6.5 BFN M+SAR Section 6.5, "Standby Liquid Control System"

The SLCS is a manually operated system that pumps a sodium pentaborate solution into the vessel to provide neutron absorption and achieve a subcritical reactor condition in the situation where some or all of the control rods cannot be inserted.

See discussion in Section 3.2.3 of this SE for SLCS evaluation in the MELLLA+ operating domain.

3.6.6 BFN M+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 HVAC systems that consist mainly of heating, cooling supply, exhaust, and recirculation units in the turbine building, reactor building, radwaste building, control building, diesel generator building, drywell, and the ECCS pump rooms. The licensee confirmed that the generic M+LTR (Reference 17) treatment of the HVAC topic is applicable to BFN. Specifically, the BFN HVAC systems, the process temperatures, and heat loads from motors and cables in the 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 worst case conditions.

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

3.6.7 BFN M+SAR Section 6.7, "Fire Protection"

The fire protection topic in the M+LTR (Reference 17) states that because the decay heat does not change for the MELLLA+ operating domain expansion, there are no changes in vessel water level response, operator response time, PCT, and peak SPT and containment pressure. Therefore, the MELLLA+ operating domain expansion does not affect any features of the fire protection design. Although fire responses, operator actions, and safe shutdown systems are plant-specific, MELLLA+ does not change the design requirements of fire events or requirements for operator actions or safe shutdown systems.

The licensee confirmed in Section 6.7 of the M+SAR that for BFN, the above parameters do not change, and operator response times are not affected by MELLLA+. Therefore, no further evaluation of BFN fire protection is required for MELLLA+ operating domain expansion.

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

3.6.8 BFN M+SAR Section 6.8, "Other Systems Affected"

The licensee confirmed in Section 6.8 of the M+SAR that the generic M+LTR (Reference 17) treatment of the other systems affected topic is applicable to BFN. Specifically, the licensee performed a review to assure that the M+SAR included all systems that may be affected by the implementation on MELLLA+. The licensee has confirmed that the BFN systems evaluated
[[]] were reviewed for MELLLA+ operating

domain expansion to ensure that all significantly affected systems were addressed in the BFN M+SAR.

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

3.7 BFN M+SAR Section 7.0, "Power Conversion Systems"

3.7.1 BFN M+SAR Section 7.1, "Turbine-Generator"

The licensee confirmed that the generic M+LTR (Reference 17) treatment of the turbine-generator topic is applicable to BFN. Specifically, there is no change in the BFN reactor power level, reactor operating pressure, or main electrical output of the generator as a result of MELLLA+ operation. The licensee further maintained that there is no change to the BFN missile avoidance and protection analysis.

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

3.7.2 BFN M+SAR Section 7.2, "Condenser and Steam Jet Air Ejectors"

The licensee confirmed that the generic M+LTR (Reference 17) treatment of the condenser and steam jet air ejector is applicable to BFN. Specifically, the licensee stated that there is no change in the BFN power level, reactor operating pressure, or MS flow rates as a result of MELLLA+ operation. The licensee stated that [[

]]

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

3.7.3 BFN M+SAR Section 7.3, "Turbine Steam Bypass"

The turbine steam bypass system provides a means of accommodating excess steam generated during normal plant maneuvers and transients. The licensee confirmed that the generic M+LTR (Reference 17) treatment of the turbine steam bypass is applicable to BFN. Specifically, there is no change in the BFN 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 BFN because the turbine steam bypass is unaffected by operation in the MELLLA+ operating domain.

3.7.4 BFN M+SAR Section 7.4, "Feedwater and Condensate Systems"

The licensee confirmed that the generic M+LTR (Reference 17) treatment of the FW and condensate topic is applicable to BFN. Specifically, there is no change in the BFN FW pressure, temperature, and flow rates. Since FW flow is unchanged in the MELLLA+ domain, system resistance and operating pressures in the MELLLA+ operating domain are also unchanged.

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

3.8 BFN M+SAR Section 8.0, "Radwaste Systems and Radiation Sources"

Regulatory Requirement

The BFN LAR requests operation in the MELLLA+ domain, which expands the operating boundary without increasing the maximum licensed power level, FW flow, steam flow, activation source terms, or volume of liquid radioactive waste or fission product concentrations in the reactor coolant. Any potential radiological consequences affecting occupational workers or members of the public depend on the liquid, gaseous and solid waste management, radiation sources in the reactor core, and radiation sources in the reactor coolant.

The NRC staff's review included an evaluation of any increases in radiation sources and the effects these increases may have on plant area dose rates, plant radiation areas, and accessibility to plant areas. The staff evaluated how occupational doses may be affected if access to plant vital areas is needed following an accident. The staff also considered the effects of the proposed MELLLA+ on plant effluent levels and radioactive waste generation, and the impact that any increase in these may have on offsite radiation doses to any member of the public. The NRC's acceptance criteria for occupational and public radiation doses are based on 10 CFR Part 20, "Standards for Protection Against Radiation," and 10 CFR Part 50, Appendix I, "Numerical Guides for Design Objectives and Limiting Conditions for Operation to Meet the Criterion 'As Low As Is Reasonably Achievable' for Radioactive Material in Light-Water-Cooled Nuclear Power Reactor Effluents."

The following regulatory requirements and guidance documents are applicable to the review of Section 8 of the M+SAR (Attachment 6 to the LAR (Reference 1)):

1. 10 CFR Part 20, "Standards for Protection Against Radiation"
 - a. 10 CFR 20.1101, "Radiation Protection Programs"
 - b. 10 CFR 20.1201, "Occupational Dose Limits for Adults"
 - c. 10 CFR 20.1301, "Dose Limits for Individual Members of the Public"
2. 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities"
 - a. 10 CFR 50.36a, "Technical specifications on effluents from nuclear power reactors"
3. 10 CFR Part 50, Appendix I, "Numerical Guides for Design Objectives and Limiting Conditions for Operation to Meet the Criterion 'As Low As Is Reasonably Achievable' for Radioactive Material in Light-Water-Cooled Nuclear Power Reactor Effluents."
4. NUREG-0737, "Clarification of TMI 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."

The regulations in 10 CFR parts 20 and 50, recited here, are summarized in Section 2.4 of this SE.

NUREG-0737, Item II.B.2 (Reference 23), requires the performance of radiation shielding design reviews to ensure that the design permits adequate access to important areas and provides for protection of safety equipment from radiation following an accident.

Radiation Protection Technical Evaluation

The BFN LAR requests operation in the MELLLA+ domain, which expands the operating boundary without increasing the maximum licensed power level, FW flow, steam flow, activation source terms, volume of liquid radioactive waste, or fission product concentrations in the reactor coolant. Any potential radiological consequences affecting occupational workers or members of the public depend on the liquid, gaseous and solid waste management, radiation sources in the reactor core, and radiation sources in the reactor coolant.

3.8.1 BFN M+SAR Section 8.1, "Liquid and Solid Waste Management"

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 for BFN is not considered to be significant. The licensee performed an evaluation using an MCO of 0.3 wt%, which is considered to be a conservatively high MCO because of the efficiency of the new steam dryers that were installed as a result of the BFN EPU. The MCO of 0.3 percent was used to bound the expected MCO as a result of operating in the MELLLA+ operating domain. Overall, the amount of MCO and fission and corrosion products that will pass through the system to the condenser will be small and will result in a negligible increase in the loading of the condensate demineralizers. As a result of the negligible increase in loading of the condensate demineralizers, the RWCU filter demineralizer backwash frequency is not expected to change. Therefore, the NRC staff concludes that waste volumes will not be significantly affected by operation in the MELLLA+ operating domain.

3.8.2 BFN M+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 to members of the public in the offsite environment is within existing limits and ALARA. The gaseous release rate is dependent on the fuel cladding performance, main condenser air inleakage, charcoal adsorber inlet dew point, and charcoal adsorber temperature. None of these parameters for BFN are affected by operation in the MELLLA+ domain. In addition, the BFN-specific radiolytic gas flow rate of 0.045 cfm/MWt does not change as a result of operation in the MELLLA+ operating domain. Therefore, the performance of the recombiner and the BFN offsite release rate will be unaffected. Accordingly, the NRC staff concludes that the operation of the gaseous waste management system will not be affected by operation in the MELLLA+ operating domain.

3.8.3 BFN M+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. BFN operating in the MELLLA+ operating domain expansion does not involve a change in the current licensed

maximum reactor thermal power. Therefore, 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 BFN M+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 nitrogen-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. Therefore, the concentration of radiation sources in the reactor water is not expected to be significantly impacted by operating in the MELLLA+ operating domain.

When conducting operation in the MELLLA+ operating domain, the moisture content of the steam leaving the reactor vessel may increase by up to 0.1 wt% at times when operating at the minimum CF in the MELLLA+ operating domain. 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 increased when operating in the MELLLA+ operating domain. However, BFN has replaced the steam dryers, which improved MCO performance, thus allowing MS MCO and resulting radioactivity to remain within the analyzed limits of the CLTP level. Therefore, the NRC staff concludes that the concentrations of the fission and activated corrosion products in the reactor coolant and steam will remain bounded by the BFN EPU source term and unchanged during MELLLA+ conditions.

3.8.5 BFN M+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). During operation, the primary source of radiation in the turbines is dependent upon power level. However, the radiation shielding that is provided around radioactive waste systems, MS lines, and the turbine are conservatively designed to minimize any effect on occupational doses from increased source terms. In addition, the BFN reactor power level, MS flow rate at 100 percent rated thermal power, and rated CF do not change as a result of operating in the MELLLA+ operating domain. Therefore, the radiation levels from the coolant activation products will not vary significantly. The moisture content of the moisture separator outlet may increase at certain times while operating in the MELLLA+ operating domain, which can affect radionuclide concentrations in the coolant. However, BFN has replaced the steam dryer, which improved MCO performance, thus allowing MS MCO and radiation levels to remain within the analyzed limits of the CLTP level. In addition to the improved steam dryer performance, BFN will monitor the MCO to ensure it is controlled within the MELLLA+ analytical assumed conservative MCO value of 0.3 wt percent, which was used for determining normal operation radiation levels.

The post-shutdown radiation levels are dominated by the accumulated contamination of some fission and activated corrosion products. The BFN 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 overall radiological effects of the increase in MCO are a function of the coolant radiochemistry and the levels of activated corrosion products. To address any increases in the radiological effects of the MCO, BFN controls access to radiation areas by maintaining appropriate health physics and ALARA controls in accordance with the regulations in

10 CFR Part 20. Therefore, the NRC staff concludes that the increase in radiation sources associated with operation 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.

The in-plant post-accident radiation levels depend primarily upon the post-accident source term. The post-accident source term consists 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 BFN maximum licensed power. Operating in the MELLLA+ operating domain does not change the maximum licensed power; therefore, the NRC staff concludes that 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.

3.8.6 BFN M+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 offsite doses to members of the public. As a result of operation in the MELLLA+ operating domain, the reactor power and steam flow rate do not change. The MCO in the MS may increase in the MELLLA+ operating domain for short periods of time during the operating cycle. However, the licensee's improved MCO performance will ensure that the MS MCO remains within the current licensed thermal power analyzed limits. Furthermore, 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, plant gaseous emissions, and gamma shine. 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. Therefore, the NRC staff concludes that the contribution to offsite doses will be negligible, and doses to the public will remain a small percentage of the dose limits in 10 CFR Part 20 and ALARA.

3.8.7 Technical Conclusion for Section 8

The NRC staff has concluded, based on the above evaluation, that the changes proposed by the licensee do not impact the licensee's ability to operate the facility within the requirements of 10 CFR Part 20 and 10 CFR Part 50, Appendix I. The licensee has taken the necessary steps to ensure that any increases in radiation doses will be maintained ALARA. As a result, the licensee continues to provide reasonable assurance that the requirements of 10 CFR Part 20 and 10 CFR Part 50, Appendix I, will continue to be met. Therefore, the staff finds the proposed changes acceptable.

3.9 BFN M+SAR and AMSAR Section 9.0, "Reactor Safety Performance Evaluations"

Section 9.0, "Reactor Safety Performance Evaluations," of the AMSAR (Attachment 8 to the LAR (Reference 1)) evaluates the following topics and their associated subjects on a plant-specific basis:

- 9.1, "Anticipated Operational Occurrences"
- 9.2, "Design Basis Accidents and Events of Radiological Consequence"
- 9.3, "Special Events" (ATWS Overpressure and ATWS PCT and Oxidation)

3.9.1 AMSAR Section 9.1, "Anticipated Operational Occurrences"

The UFSAR for BFN contains plant-specific design-basis analyses to evaluate the effects of a wide range of AOOs that might occur at the plant. Since these analyses are performed on a cycle- and core-configuration-specific basis during the standard reload analyses, the licensee provided demonstration analyses of the potentially limiting events.

The NRC staff reviewed this section to ensure the potentially limiting events are identified and considered for explicit analysis in the MELLLA+ domain, the AOO results in the MELLLA+ operating domain are realistic, and the results meet SAFDLs as required by GDC 10.

The methodologies used to analyze the AOOs are as follows:

- MICROBURN-B2 (Reference 29)
- COTRANSA2 (Reference 73)
- XCOBRA (Reference 74)
- XCOBRA-T (Reference 75)
- RODEX2 ((Reference 76) and (Reference 77))

The NRC staff's review of the applicability of the AOO codes and methods in MELLLA+ is found in Appendix E of this SE.

In the AMSAR, Section 9.1.1, the licensee identified the following potentially limiting AOOs in terms of thermal margin (i.e., Δ CPR) under the proposed MELLLA+ conditions at BFN:

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

The NRC staff reviewed the limiting events discussed in the M+LTR (Reference 17) and the transients in the licensee's UFSAR. The NRC staff determined the licensee adequately identified the limiting events since they are consistent with those identified in the M+LTR and approved SER (Reference 41) and since the staff did not identify any AOO event in the UFSAR that needed additional consideration.

For each limiting event, the AMSAR specifies that the analyses be performed at 100 percent CLTP and at both the minimum MELLLA+ flow condition and the ICF condition. This ensures that the extension to the MELLLA+ operating domain results in acceptable Δ CPR values that do not compromise the integrity of the fuel. The staff determined that the licensee met this requirement since these flow conditions were analyzed as seen in Table 9-1 in the AMSAR.

The NRC staff reviewed the limiting AOO results in Tables 9-1 through 9-5 of the AMSAR and in Figures 9-1 through 9-9 of the AMSAR. The staff did not identify any results that would question the methodology's ability to predict results in the MELLLA+ domain. However, the figures in the AMSAR were provided for the 100 percent power and 105 percent flow case since these were the overall limiting Δ CPR cases. While these cases were depleted with a MELLLA+ core, they do not show the plant response at the MELLLA+ knee (i.e., 100 percent power and 85 percent flow). Therefore, the NRC staff requested the licensee provide plots of the limiting AOO cases

at the MELLLA+ knee to understand the overall impact of MELLLA+. The staff reviewed the figures in the RAI response (Reference 6) and did not identify any results that would call into question the methodology's ability to predict results in the MELLLA+ domain.

Results of the licensee's AOO analyses are shown in Table 3.9.1-1 (reproduced from Table 9-1 of the AMSAR). The limiting event is FWCF, which shows a Δ CPR of 0.35 for the ICF/CLTP case and 0.30 for the 85 percent flow/CLTP case. For this two case, as well as for TTNB and LRNB, the Δ CPR response was more limiting at ICF than at 85 percent flow. Therefore, the licensee determined that operating in the MELLLA+ domain does not result in an increased OLMCPR for the demonstration AOO analysis. The NRC staff reviewed these results and determined that the MELLLA+ impact is not significant and is consistent with staff experience with other plants in MELLLA+. Therefore, there is no degradation in the licensee's ability to use the AOO methodology in the MELLLA+ operating domain.

To clarify how the licensee would treat the AOOs at MELLLA+ domain statepoints on a cycle-specific basis, the NRC staff questioned if these statepoints were explicitly reanalyzed each cycle in SRXB-RAI 8 (Reference 6). The licensee stated that in each reload, the plant parameters document is provided to its fuel vendor (AREVA), which includes planned plant changes each cycle. Each change is reviewed for its impact on an event, and if there is an impact, the event is potentially reanalyzed. It is noted that this process is unchanged compared to its current reload process. For the limiting events that are analyzed each cycle (e.g., LRNB), the licensee and vendor determine if there are any changes that would require explicit analysis at the MELLLA+ domain statepoints. Since the licensee is reviewing all changes each reload and this process is not impacted by MELLLA+, the NRC staff determined that the licensee's approach for consideration of the MELLLA+ domain statepoints in the AOO analyses is acceptable.

Table 3.9.1-1: AOO Event Results Summary

Event	Parameter	Unit	CLTP ICF (105% Rated Core Flow)	CLTP 85% Rated Core Flow
TTNB	Peak Neutron Flux	% Initial	315	254
	Peak Heat Flux	% Initial	122	117
	Peak Vessel Pressure	psia	1319	1314
	Δ CPR (TSSS)	N/A	0.31	0.27
LRNB	Peak Neutron Flux	% Initial	324	272
	Peak Heat Flux	% Initial	119	123
	Peak Vessel Pressure	psia	1314	1322
	Δ CPR (TSSS)	N/A	0.31	0.27
FWCF	Peak Neutron Flux	% Initial	322	257
	Peak Heat Flux	% Initial	125	122
	Peak Vessel Pressure	psia	1275	1272
	Δ CPR (TSSS)	N/A	0.35	0.30
LFWH	Δ CPR	N/A	0.14	--
CRWE	Δ CPR	N/A	0.29	0.23

The licensee identified the potentially limiting AOO events for explicit consideration in the MELLLA+ domain. The plant response to the AOO did not call into question the methodology's ability to predict results in the MELLLA+ domain, and the licensee demonstrated that there is acceptable margin to the fuel design limits for these AOOs. Therefore, the NRC staff determined the AOOs are adequately evaluated in the MELLLA+ operating domain.

3.9.2 BFN M+SAR and AMSAR Section 9.2, "Design Basis Accidents and Events of Radiological Consequence"

Regulatory Requirements

The evaluation of the release of fission products (source term) into containment is used for determining the acceptability of both the plant site and the effectiveness of engineered safety features. In the past, power reactor licensees have typically used Technical Information Document (TID)-14844 (Reference 78), "Calculation of Distance Factors for Power and Test Reactor Sites," as the basis for DBA source terms. The DBA offsite radiological dose consequences are evaluated against the guideline dose values in terms of whole body and thyroid dose stated in 10 CFR Section 100.11, "Determination of Exclusion Area Boundary, Low Population Zone, and Population Center Distance," which refers to TID-14844. Regulatory guidance for the review of DBAs based on TID-14844 is provided in RG 1.195, "Methods and Assumptions for Evaluating Radiological Consequences of Design Basis Accidents at Light-Water Nuclear Power Reactors."

In December 1999, the NRC issued a new regulation, 10 CFR 50.67, "Accident source term," which provided a mechanism for licensed power reactors to replace the traditional accident source terms used in their DBA analyses with alternate source terms (AST) and established the TEDE as the new acceptance criteria. The TEDE acceptance criteria of 10 CFR 50.67(b)(2) replaces the previous whole body and thyroid dose guidelines of 10 CFR 100.11 to determine the exclusion area boundary (EAB), low population zone (LPZ), and population center distance. In addition, holders of operating licenses using an AST under 10 CFR 50.67 shall meet the requirements of Appendix A to 10 CFR Part 50 and GDC 19 regarding control room access and occupancy. This applies not only to the analyses performed in the application, which may only include a subset of the plant analyses, but also to all future design-basis analyses. Regulatory guidance for the implementation of the AST is provided in RG 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors."

On September 27, 2004, the NRC issued Amendment Nos. 251, 290, and 249 for BFN Units 1, 2, and 3, respectively (Reference 79), approving the implementation of the AST in accordance with 10 CFR 50.67 following the guidance provided in applicable sections of RG 1.183. The regulatory requirements for which the NRC staff based its acceptance of the analyses are the accident dose criteria in 10 CFR 50.67, as supplemented in Regulatory Position 4.4 of RG 1.183, and GDC 19, as supplemented by SRP Section 6.4, "Control Room Habitability System." The NRC staff utilized the regulatory guidance provided in SRP Section 15.0.1, "Radiological Consequence Analysis Using Alternative Source Terms," in performing this review. The NRC staff also considered relevant information in the BFN UFSAR and TSs.

The regulatory requirements that the NRC staff used in the requested licensing action are based on the reference values in 10 CFR 50.67, the accident-specific guideline values in Regulatory Position 4.4 of RG 1.183, and Table 1 of SRP Section 15.0.1.

3.9.2.1 Design Basis Events

The NRC staff reviewed the impact of the proposed MELLLA+ on DBA radiological consequence analyses approved with an AST, as documented in Chapter 14 of the UFSAR (Reference 49). The specific DBA analyses reviewed are as follows:

- UFSAR Section 14.6.2, "Control Rod Drop Accident (CRDA)"
- UFSAR Section 14.6.3, "Loss-of-Coolant Accident (LOCA)"
- UFSAR Section 14.6.4, "Refueling Accident"
- UFSAR Section 14.6.5, "Main Steam Line Break Accident (MSLBA)"

In its application, the licensee indicated postulated DBA events have been evaluated and analyzed to show that NRC regulations are met. DBA events have either been previously analyzed at 102 percent of rated thermal power of 3,952 MWt at 4,031 MWt, which bounds the operation in the MELLLA+ operating domain or is not dependent on core thermal power. The evaluation/analysis was based on the methodology, assumptions, and analytical techniques described in the BFN current licensing basis, regulatory guides, and previous SEs. The NRC staff confirmed that the current licensing basis dose consequence analyses remains bounding at the proposed MELLLA+ power level at the rated thermal power of 3,952 MWt.

Two postulated CRDA events govern the analysis of radiological consequences for BFN. For Event 1, the release path is via the 1 percent condenser/turbine leakage with the MSIVs isolated. For Event 2, the release is at normal power, and the release path is via the condenser.

For Event 1, the plant is not operating in the MELLLA+ operating domain as shown by the power/flow map, and therefore, there is no effect on the results. Since BFN may operate with portions of the offgas system bypassed, Event 2 represents the bounding radiological consequences. The CRDA release is dependent on the source terms and maximum peaking factor. Operation in the MELLLA+ operating domain does not affect the AST CRDA source term, and the peaking factor remains bounding. There are no changes to removal, transport, or dose conversion assumptions for this event. Therefore, the licensee concluded that the BFN CRDA evaluation for the MELLLA+ operating domain is bounded by the analysis for the current licensed operating domain.

The postulated LOCA could release radioactive material directly into the primary containment as the result of nuclear system pipe breaks inside the drywell. The most severe nuclear system effects and the greatest release of radioactive material to the primary containment results from a complete circumferential break of one of the recirculation loop pipelines. The radiological consequences of the LOCA are generally based on a time-phased release of the reactor core source term inventory and various TS source terms. Operating in the MELLLA+ operating domain does not affect the reactor core source term inventory or release phase timing. Therefore, the licensee concluded, and the staff agrees, that the BFN LOCA evaluation for the MELLLA+ operating domain is bounded by the analysis for the current licensed operating domain.

The postulated refueling accident, or fuel handling accident (FHA), that could result in the release of radioactive materials directly to the secondary containment is an event that can occur when the primary containment is open. The greatest potential for the release of radioactive material exists when the primary containment head and reactor vessel head have been removed. With the primary containment open and the reactor vessel head off, radioactive material released as a result of fuel failure is available for transport directly to the reactor building. The radiological consequences of the FHA are primarily determined using a fraction of the reactor core source term inventory modified by a power peaking factor and maximum radial peaking factor and a decay period of 24 hours. Operating in the MELLLA+ operating domain does not affect these parameters. Therefore, the licensee concluded, and the staff agrees, that the BFN FHA evaluation for the MELLLA+ operating domain is bounded by the analysis for the current licensed operating domain.

The postulated MSLBA accident that could release radioactive materials outside the secondary containment is the result of postulated breaches in the nuclear system process barrier. The DBA is a complete severance of one MS line outside the secondary containment. The analysis of record for the worst-case MSLBA radiological consequences is at hot standby conditions, which is outside of the MELLLA+ operating domain. Additionally, the source terms used to compute radiological consequences are dependent on the relative amount of water and steam released. The moisture content of the moisture separator outlet may increase at certain times while operating in the MELLLA+ operating domain. However, considering the RSD improved MCO performance, the resulting MS MCO will remain within the CLTP results. Therefore, the licensee concluded, and the staff agrees, that the BFN MSLBA evaluation for the MELLLA+ operating domain is bounded by the analysis for the current licensed operating domain.

Using the licensing basis in the current BFN UFSAR, in addition to the information in the February 23, 2018, application, the NRC staff verified that the existing BFN UFSAR Chapter 14 radiological analyses and release assumptions bound the conditions for the proposed expanded operating domain as they relate to the radiological consequences of DBA analyses.

3.9.2.1.1 BFN AMSAR Section 9.2.1.1, "Control Rod Drop Accident"

Attachment 18 to the LAR (Reference 1), ANP-3552 NP BFN Unit 3 Cycle 19 Representative Reload Analysis (EPU MELLLA+), reports that a CRDA evaluation was performed for both A and B sequence startups consistent with the withdrawal sequence, including rod worth minimization strategies. CRDA evaluation was performed for both A and B sequence startups consistent with the withdrawal sequences specified by TVA. This CRDA analysis was performed based on the approved AREVA generic CRDA methodology as described in XN-NF-80-19(P)(A), Volume 1, and Supplements 1 and 2 (Reference 80). This CRDA analysis demonstrated that the maximum deposited fuel rod enthalpy is less than 280 calorie/gram (cal/g), and the estimated number of fuel rods that exceeds the fuel damage threshold of 170 cal/g is less than the number of failed rods supported by the BFN CRDA AST analysis based on the ATRIUM 10XM fuel.

Additionally, the licensee reevaluated in Attachment 32 to the LAR (Reference 1) the CRDA analysis based on Draft Regulatory Guide (DG)-1327 (Reference 81). This assessment included the pellet-cladding mechanical interaction criteria that addresses fuel failures due to pellet clad mechanical interaction, high cladding temperature failure threshold, and rod failure assessment. The results indicate there is margin to the pellet-cladding mechanical interaction failure and the maximum total enthalpy thresholds. 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 reevaluated CRDA assessment meet GDC 28.

3.9.3 BFN M+SAR and AMSAR Section 9.3, "Special Events"

3.9.3.1 BFN M+SAR and AMSAR Section 9.3.1, "Anticipated Transient Without Scram"

The licensee provided analysis of the required ATWS events in Section 9.3.1 of the M+SAR and Section 9.3.1 of the AMSAR. The M+SAR contains the analysis of the ATWS events to demonstrate suppression pool temperature, containment pressure, peak cladding temperature (and oxidation), and ATWS with core instability. The AMSAR contains the analysis of ATWS events for vessel pressure.

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 1,500 psig.
- Core coolability is maintained (i.e., PCT below 2,200 °F).
- The peak suppression pool temperature is less than the design limit (281 °F for BFN).
- The peak containment pressure is less than the containment design pressure (56 psig for BFN).

Additionally, the NRC staff reviewed the limiting event determination, the sequence of events, the analytical models and their 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+SER pertaining to the ATWS analysis. The licensee addressed these limitations in Appendix B of the M+SAR and AMSAR.

1. M+SER L&C 12.17, which specifies that at least 2 plant-specific ATWS calculations (MSIVC and PRFO) be performed, with an additional LOOP calculation required if the RHR capability is affected by LOOP.
2. M+SER L&C 12.18.a, which specifies that plant-specific TRACG calculations in the event that the heat capacity temperature limit (HCTL) is exceeded in the ODN ATWS calculations.
3. M+SER 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 percent flow ODN calculation.
4. M+SER 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+SER L&C 12.18.d, which specifies that operation in the MELLLA+ domain is 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+SER 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 (Reference 82), or may differ from the original analyses in a manner yielding more conservative results.
7. M+SER L&C 12.18.f, which specifies that the licensee tabulate and discuss the key input parameters and uncertainty treatment.
8. M+SER L&C 12.23.1, which is included as part of L&C 12.18.d.
9. M+SER L&C 12.23.2, which requires that all plant-specific ODN and TRACG key calculation parameters be provided for staff verification.
10. M+SER 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+SER L&C 12.23.4, which specifies review of the Emergency Procedure Guideline (EPG)/Severe Accident Guideline (SAG) parameters for applicability to the MELLLA+ domain and requires confirmation that the ATWS analyses are consistent with the Emergency Operating Procedure (EOP) operator actions.

12. M+SER L&C 12.23.5, which specifies that a power/flow ratio of less than 52.5 MWt/Mlbm/hr at minimum allowable CF rate at 120 percent OLTP.
13. M+SER 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+SER 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 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+SER 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+SER L&C 12.23.11, which specifies that plant-specific applications must justify the use of plant-specific suppression pool temperature limits for the ODPN and TRACG calculations that are higher than the HCTL limit for emergency depressurization.
17. M+SER 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 21). Therefore, an ATWS event cannot be classified as either an AOO or a DBA.

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 MS line isolation valves or turbine trip with bypass available if unmitigated unstable power oscillations can 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 presented 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 ATWS 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

There is no change in core power, decay heat, pressure, or steam flow because 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 increases the severity of ATWS events. Many ATWS events are possible. However, only a small number of events is expected to be limiting in terms of the ATWS acceptance criteria described in the Regulatory Evaluation section above. L&C 12.17 of the approved M+SER specifies that the following ATWS events must be considered as the limiting events for operation up to and including the MELLLA+ domain:

- MSIVC
- 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, which leads to an increase in core power due to coolant density reactivity feedback. In addition, during these events, the recirculation pumps are tripped. 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 maintained 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 evaluated and 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.

ATWS Methodologies Used for BFN MELLLA+ Analysis

The GEH transient code ODYN (Reference 83) is approved as a licensing basis code for ATWS analyses in the MELLLA+ domain per the M+SER. The AREVA transient code COTRANSA2 (Reference 84) is approved for ATWS analysis, and review of the COTRANSA2 methodology in the MELLLA+ domain is found in Appendix E of this SE. 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 BFN MELLLA+, COTRANSA2 is used exclusively for the short-term ATWS vessel overpressure analyses, and ODYN is used for the long-term ATWS analyses for suppression pool temperature, containment pressure, and cladding PCT/coolable geometry.

ODYN ATWS Analysis

The licensee performed plant-specific analyses for the MSIVC and PRFO ATWS events, as required by L&C 12.17. If the RHR heat exchanger effectiveness is affected by LOOP, L&C 12.17 requires that an analysis for the LOOP event is necessary. As stated in M+SAR, the

LOOP event results in a reduction in the RHR suppression pool cooling capability. Therefore, the LOOP event was performed with the MSIVC and PRFO. The NRC staff determined that since the licensee performed the LOOP analysis, the L&C was met.

The long-term ATWS ODYN calculations were initiated from 100 percent CLTP and 85 percent reactor CF, 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 BOC, middle-of-cycle (MOC), and 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 concludes that these conditions are acceptable because they satisfy the requirements for ATWS analyses by providing reasonably limiting assumptions.

The licensee provided the key input parameters for the ATWS analysis in Table 9-1 of the M+SAR. The NRC staff compared these parameters to the plant's licensing basis and the requirements in the M+LTR and determined the licensee sufficiently modeled BFN for the analyses.

It is also noted that the licensee is not changing the required operator actions to respond to an ATWS (i.e., water level reduction, SLC system boron injection, and RHR suppression pool cooling) as defined in the BFN Emergency Operating Instructions (EOIs). EOPs are designated as EOIs at BFN. These operator actions were evaluated in Section 3.10.6 of this SE.

The analyses were performed using nominal and bounding fuel parameter sensitivities for ATRIUM 10XM fuel. The bounding fuel parameter sensitivities conservatively account for differences between ATRIUM-10 and ATRIUM 10XM fuel for potential biases in inputs for ATRIUM 10XM, and to conservatively account for features/parameters that were not specifically modeled for ATRIUM 10XM fuel. The licensee selected the limiting fuel parameter cases as the licensing basis to conservatively bound the performance of ATRIUM 10XM. The staff's analysis of the fuel parameters and ranges used in the sensitivity studies are described in Appendix D of this SE.

Table 3.9.3.1-1 shows key results for the ODYN ATWS analysis. Even with conservative assumptions for fuel parameter sensitivity studies (direct energy deposition, gap conductance, and thermal and hydraulic channel losses), the ODYN analyses demonstrate that the ATWS acceptance criteria are satisfied for BFN MELLLA+. The calculated peak suppression pool temperature, peak containment pressure, and peak cladding temperature remain within the design limits for all cases analyzed. Since the results from the ODYN analysis meet the ATWS acceptance criteria, the NRC staff concludes that the results are acceptable.

The licensee provided tables of the sequence of events for each ATWS scenario (i.e., PRFO, MSIVC, and LOOP) in Tables 9-4 through 9-6 of the M+SAR. Additionally, the licensee provided pressure, suppression pool temperature, and PCT results in Figures 9-4 and 9-5 of the M+SAR. The staff compared trends of these results to a similar plant's MELLLA+ ATWS results. From this comparison and staff's experience with ATWS results, the staff determined that the BFN ATWS results are consistent with the expected characteristics of the ATWS events.

Table 3.9.3.1-1: Key Results for Licensing Basis OLYN ATWS Analysis

ATWS Acceptance Criterion	Limiting ATWS Event	MELLLA+ (limiting fuel parameter sensitivity)	Design Limit
Peak Suppression Pool Temperature (°F)	LOOP at EOC	[[281.0
Peak Containment Pressure (psig)	LOOP at EOC		56
Peak Cladding Temperature (°F)/Coolable Geometry	PRFO at EOC]]	2200

¹ This temperature is below the BFN HCTL (~176 °F); therefore, no additional TRACG analysis is necessary to model emergency depressurization. This addresses L&C 12.18a of the M+SER.

The licensee reviewed the limiting ATWS events and determined that the results of ATWS events are consistent with the expected plant responses, the analytical models and their applicability are acceptable for use in MELLLA+ (see Appendix E), and the values of parameters used in the analytical model are acceptable. Therefore, the NRC staff concludes that the ATWS analysis for the suppression pool temperature, containment pressure, and core coolable geometry are acceptable.

COTRANSA2 ATWS Vessel Pressure Analysis

The COTRANSA2 ATWS analysis was performed to demonstrate vessel pressure remains below the ASME Service Level C limit of 1,500 psig. The NRC staff reviewed the COTRANSA2 ATWS overpressure analyses provided in Section 9.3.1.1 of the AMSAR. The applicability of the COTRANSA2 methodology in MELLLA+ is reviewed in Appendix E of this SE. The analysis provided is a demonstration analysis of ATWS events to demonstrate that no change in overpressure relief capacity is required. As stated in Section 9.3.1.1 of the AMSAR, the licensee will analyze the ATWS event for each reload.

The NRC staff reviewed the demonstration analysis to ensure the limiting ATWS cases were evaluated, there were no unexpected results, and the results meet the acceptance criteria.

The approved M+LTR required analysis of four ATWS events: PRFO, MSIVC, inadvertent opening of a relief valve (IORV), and LOOP.

The licensee qualitatively dispositioned the IORV and LOOP events. For the IORV, the licensee discussed that this is non-limiting for overpressure because the reactor is not isolated during this event; therefore, vessel overpressure is not an issue. Since there is not another valve upstream of the IORV that could isolate the vessel at BFN, the NRC staff determined that this justification is acceptable. For the LOOP event, the licensee discussed that the fast opening of the bypass valves will reduce the pressure wave created by the reactor isolation. Since the vessel will have a vent path in the LOOP event, the NRC staff determined that it will be bounded by the PRFO and the MSIVC event for overpressure, and the qualitative disposition is acceptable.

The MSIVC and the PRFO lead to a complete isolation of the vessel. Therefore, the licensee explicitly analyzed both the cases. The key input parameters for the analyses are provided in Table 9-6 of the AMSAR. The NRC staff compared these parameters to the plant's licensing

basis and the requirements in the M+LTR and determined the licensee sufficiently modeled BFN for the analyses. Additionally, the licensee performed the analyses at the statepoints required by the M+LTR (i.e., required power and flow combinations and cycle exposures). Therefore, the staff determined that the licensee appropriately followed the M+LTR.

The results of the ATWS overpressure analysis are provided in Tables 9-7 and 9-8 of the AMSAR. The key parameters for the events are provided in Figures 9-11 through 9-19 of the AMSAR. The limiting ATWS event for overpressure was the PRFO event that resulted in a peak pressure of 1,498 psig. This value is within the acceptance criteria (1,500 psig) but is close to the limit. This result is discussed later in this section.

The NRC staff reviewed the results in Figures 9-11 through 9-19 of the AMSAR and compared trends of these results to a similar plant's MELLLA+ ATWS results. Additionally, the staff reviewed the differences in MELLLA and the MELLLA+ results provided in Table 9-7 (~30 psi difference). From the comparison and experience with ATWS results, the NRC staff determined that the BFN ATWS results are consistent with the expected characteristics of the ATWS event.

The NRC staff investigated the peak vessel pressure result (1,498 psig) in more detail because of the appearance of low margin to the acceptance criteria (1,500 psig). The L&Cs in the M+LTR allow the licensee to statistically analyze the SRV's opening setpoint in the ATWS overpressure event using the NRC's historical 95 percent worst case result/95 percent confidence approach (95/95 approach). The licensee reported that using a statistical approach, the overpressure results bound the 99.7 percent worst case result with 95 percent confidence. The staff determined that bounding the result at that confidence level would require a minimum 1,000 runs/trials per the binomial probability function (see Equation 22.3 in NUREG-1475, Revision 1). Given 1,000 trials, the licensee would have the option to use the 39th ranked result to meet the 95/95 requirements. Therefore, there is more margin than the reported value in Table 9-7.

As was discussed previously, the licensee will analyze the ATWS event for each reload. Therefore, the analysis in the AMSAR is not intended to bound all future cycles; any cycle-specific changes that could impact the event and potentially impact overpressure relief capacity will be addressed for each reload. If the licensee finds that the current overpressure relief capacity cannot be met for a given reload design, it would have to take the appropriate licensing actions to address the issue. This process is no different than the licensee's current reload process to ensure each event continues to meet the requisite acceptance criteria.

Since the staff determined that the key input parameters were acceptable, the results are consistent with the expected characteristics of the ATWS event, the results meet the acceptance criteria, there is unreported margin in the results, and the licensee will reanalyze this event each reload cycle, the NRC staff finds the ATWS overpressure analyses acceptable.

There is a note in Table 9-7 of the AMSAR where the licensee discusses the results that include a generic penalty for TCD, but a refined methodology could provide additional margin (4 psi) to the reported limit. Since the NRC staff found the analysis acceptable without this additional margin, the refined methodology or its impact was not evaluated.

3.9.3.2 BFN M+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 disposition to ensure BFN MELLLA+ plant conditions and associated analysis falls within the bounds of generic disposition. The NRC staff determined that since there is no expected change in core power, operating pressure, decay heat, and steam flow for BFN MELLLA+, the generic disposition is acceptable, and the Section 9.3.2 M+LTR evaluation is applicable to this application.

3.9.3.3 BFN M+SAR Section 9.3.3, "ATWS with Core Instability"

Regulatory Basis

The NRC staff's evaluation of BFN 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. The regulation in 10 CFR 50.46(b) defines three specific core coolability criteria: (1) peak clad temperature shall not to exceed 2,200 °F, (2) maximum cladding oxidation shall not to exceed 17 percent of the total cladding thickness before oxidation, and (3) maximum hydrogen generation shall not exceed 1 percent 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 is met.

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 anticipated transient without scram – instability (ATWS-I) events is not one of inadequate coolant but of high local energy deposition and cladding heat flux. In the event that cladding temperatures exceed 2,200 °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 criteria to be satisfied under these circumstances. This position has been previously stated by the NRC staff in its evaluation of NEDO-32047A (Reference 85).

Applicable Limitations and Conditions

The M+LTR (Reference 17) and associated SER provide the following L&Cs relevant to BFN analyses for ATWS-I:

- M+SER L&C 12.3.d, which specified new plant-specific analyses to demonstrate ATWS-I performance for cores with non-GE fuel;
- M+SER L&C 12.19, which specified that plant-specific ATWS-I analyses 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+SER L&C 12.23.6, which specified that bounding ATWS-I analyses be provided for MELLLA+ applications involving non-GE fuel;
- M+SER L&C 12.23.7, which is included in L&C 12.23.6; and
- M+SER 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 CF rate and increasing core power, which require that this region of the power flow operating map be avoided during normal operation and AOOs. An 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 MELLLA+ domain leads to a higher power level following a recirculation pump trip relative to operation in pre-MELLLA+ 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.

Once the oscillations grow to a sufficiently large amplitude, 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.

One potentially limiting event for ATWS-I is the turbine trip with bypass (TTWBP) event because it results in a trip of both recirculation pumps, as well as a reduction of the FW temperature to the condenser temperature due to loss of extraction steam from the turbine. This leads to a relatively high-power level at a low flow rate (i.e., natural circulation flow), which favors the development of unstable oscillations and possible violation of the ATWS acceptance criteria unless the oscillations are suppressed in a timely manner by operator actions.

Another potentially limiting ATWS-I event is the 2RPT event. This event involves a smaller decrease in FW temperature over time compared to the TTWBP, which leads to less unstable conditions. However, where a turbine trip generates an automatic scram signal with very short delay, a recirculation pump trip requires the operators to identify the trip and initiate manual scram. In the event of failure to scram, the 2RPT event becomes an ATWS event. Due to the additional time involved in operator identification of the recirculation pump trip and initiation of

manual scram, a longer overall time may elapse in the 2RPT event between the recirculation pump trip and eventual operator actions to reduce water level compared to the TTWBP event. This longer duration allows more time for the FW temperature to drop during the event. Additionally, the longer duration allows more time for the oscillations to grow and develop, giving more opportunity for failure to rewet to occur in the highest power locations of the core. Furthermore, the vessel pressures during the 2RPT are less than during the TTWBP event, which may be destabilizing under certain conditions. These effects may cause the 2RPT event to be more limiting in terms of the ATWS acceptance criteria than the TTWBP, despite the difference in FW temperature in these two events.

The approved M+LTR (Reference 17) requires that plant-specific ATWS-I calculations be performed for the TTWBP event, as that report considered the TTWBP event to be limiting. However, as discussed above, the possibility exists for the 2RPT ATWS-I event to be limiting instead. Therefore, the NRC staff performed its review considering that either event may be limiting and that the limiting event must be justified for the plant-specific configuration of BFN under MELLLA+ conditions.

ATWS-I Calculations for BFN MELLLA+

Calculations for ATWS-I in the BFN SAR were performed with TRACG04. TRACG04 is not approved for long-term ATWS calculations, including ATWS with depressurization and ATWS with core instability. ODDYN is the approved licensing basis code for ATWS, consistent with the NRC SE for NEDC-33006P (Reference 41). 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 SEs for NEDO-32047 (Reference 85) and NEDO-32164 (Reference 86), 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 properly used TRACG04 for best-estimate ATWS-I calculations, 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 if the TH conditions indicate that transition to nucleate boiling is possible. ATWS-I results using two different T_{min} correlations were presented in the BFN SAR — the Modified Shumway T_{min} correlation and the homogeneous nucleation plus contact temperature (HN+CT) T_{min} correlation. The modified form of the Shumway correlation [

]] and assumes zirconium thermophysical properties for the cladding (which is the default provided 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 have indicated that the modified Shumway correlation may not adequately capture T_{min} under ATWS-I conditions (Reference 87). As described in (Reference 87), the NRC staff determined that a T_{min} based on the HN+CT model gives acceptable agreement with the experimental data and is a reasonable model for ATWS-I analyses. Therefore, the NRC staff made its evaluations based primarily on the HN+CT model.

The licensee presented TRACG04 results for simulated TTWBP and 2RPT events in BFN initiated from minimum allowable CF at EPU power levels, which is the most limiting operating 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 (Reference 17). 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 BFN cycle that was analyzed. Two different TRACG channel grouping (i.e., nodalization) schemes were used in the analyses — one that allows regional mode oscillations and one that allows core-wide mode oscillations only. With the regional mode channel grouping, core-wide oscillations are possible as well, and the oscillations will tend to develop intrinsically in whichever mode is more unstable. L&C 12.19 only specifies that regional mode analyses be performed, as this mode is expected to be limiting. Figures 1 and 2 of Supplement 1 to the LAR (Reference 2), give a visual depiction of these channel grouping schemes. Figures 3 and 4 in the same document demonstrate that the oscillations in the BFN ATWS-I analyses occurred in the regional mode, indicating that this mode is the most unstable mode for BFN.

Nominal or reasonably conservative values were used for most parameters for the ATWS-I calculations. This is consistent with the historical approach for ATWS (Reference 85) and (Reference 86) in which best-estimate calculations are performed. The licensee performed ATWS-I analyses for both the 2RPT and TTWBP scenarios. The FW temperature reduction rate used in each event is discussed later in this section. For the TTWBP calculations, a 120-second delay was assumed after the initiation of the event before manual water level reduction was initiated. This is consistent with the BFN EOLs. For the 2RPT event, the time between initiation of the 2RPT and initiation of manual water level reduction was assumed to be 180 seconds. This includes an assumed 60 seconds for the operator to initiate manual scram following the 2RPT and identify failure to scram (ATWS), plus an additional 120 seconds to perform manual water level reduction per the EOLs.

In Attachment 33 to the LAR, the licensee provided its justification for the assumed FW temperature versus time during the TTWBP event. For this event, the licensee assumed a 14-second delay from the time of the turbine trip to the initiation of FW temperature reduction. After this delay, a FW temperature reduction rate of 1.1 °F per second was applied for a duration of 60 seconds, followed by a rate of 1.4 °F per second throughout the remainder of the event. These values were developed based on the plant-specific configuration of the FW system in BFN, as well as a justification based on measured plant data for a turbine trip event at BFN Unit 3. The licensee provided details of this justification in Attachment 33 to the LAR (Reference 1). The NRC staff issued SRXB-RAI 5 (Reference 6) to request additional justification and to address the staff's concerns with the approach used in determining the FW temperature reduction rate, as well as to justify the FW temperature reduction rate assumed for the 2RPT event.

In the M+SAR and response to SRXB-RAI 5a, the licensee justified the assumed 14-second delay following the turbine trip before the FW temperature begins to decrease. This was based on the mass of water from the outlet of the high-pressure FW heater to the reactor vessel, with this value taken from the long-term containment response analysis performed for the BFN EPU LAR, divided by the post-RPT FW flow rate determined by scaling the nominal FW flow rate by the power reduction following RPT. This relies on the assumption that the delay in coasting down the thermal power from rated to natural circulation power and any additional delay required for the FW flow rate to respond to this after the RPT is minimal.

The NRC staff requested additional justification in SRXB-RAI 5b to ensure that the assumed delay time remains bounding relative to the measured turbine trip data for BFN Unit 3. In the response to SRXB-RAI 5b, the licensee used the measured turbine trip data and scaled the values to correspond to the expected FW flow rate during an ATWS from the MELLLA+ domain.

This resulted in a delay time estimate of 23.4 seconds, which is conservatively bound by the 14-second time delay assumed in the ATWS-I analyses. The licensee performed an additional adjustment to account for the higher average FW flow rate during this stage of the event due to the time delay associated with core and FW flow responses following the turbine trip. This adjustment resulted in a delay time estimate of 20 seconds. For further assurance, the NRC staff requested comparisons to simulator results in SRXB-RAI 5d. The licensee performed the requested BFN simulator runs of TTWBP with ATWS from the 100 percent power per 85 percent flow operating point. These simulator runs indicated a delay time of longer than 20 seconds as seen in Figure 3.9.3.3-1, below. The NRC staff reviewed the information presented and determined that the 14-second delay time before FW temperature reduction is reasonable and conservative based on measured data and is supported by simulator runs.

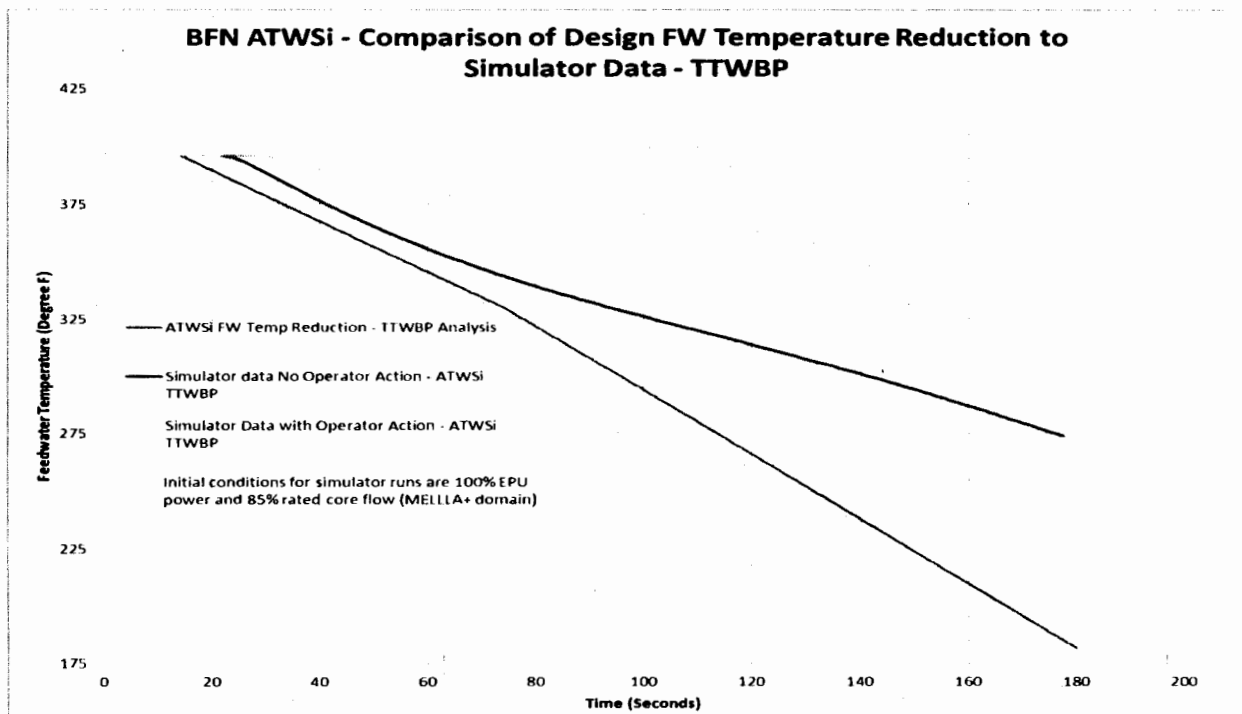


Figure 3.9.3.3-1 Feedwater Temperature Reduction Turbine Trip with Bypass

The licensee extracted data on FW temperature and FW flow rate during the BFN Unit 3 turbine trip event to determine appropriate FW temperature response during the postulated TTWBP ATWS event. During this event following the turbine trip and subsequent scram, the FW flow was kept near zero except for three short intervals in which FW injection was used to control the RPV water level. By contrast, in the postulated TTWBP ATWS event, the FW flow would remain continuously active until the operators take action to manually reduce water level within 120 seconds. However, by analyzing the rate of FW temperature decrease during each of the three FW injection intervals, and by scaling these rates by the ratio of FW flow rate expected during the TTWBP ATWS event compared to the BFN Unit 3 event flow rate, the licensee arrived at an estimate of the FW temperature response suitable for the TTWBP ATWS event. The NRC staff determined that this scaling was appropriate because the temperature of FW entering the vessel at a given point in time depends on the integrated mass of FW that has entered the vessel since the turbine trip. This is due to the fact that FW that was closer to the vessel at the time of turbine trip (and loss of extraction steam) is at a higher temperature than FW that was further upstream and closer to the condenser at the time of the turbine trip. Therefore, the rate of FW temperature reduction depends on the rate of FW flow.

The FW temperature reduction rates determined in this manner from the three FW injection intervals during the Unit 3 event were 0.4 °F/second (over a 25-second interval), 1.1 °F/second (over a 59-second interval), and 1.4 °F/second (over a 40-second interval), in chronological order. In SRXB-RAI 5c, the NRC staff requested additional explanation for the assumed FW temperature reduction rates and justification for why the assumed values are bounding for ATWS-I. In the response to SRXB-RAI 5c, the licensee justified that the observed trend of increasing FW temperature reduction rate over time is the expected behavior due primarily to the significant amount of steam remaining in the moisture separators at the time of the turbine trip, which provides additional FW heating after the trip and slows the reduction in FW temperature early on after the trip. As this additional energy in the steam is transferred to the FW and is depleted, the FW temperature reduces more quickly. Latent energy contained in the entrapped steam in the FW heater shells, extraction steam lines, and cascaded drain flow has an additional albeit smaller effect on FW temperature following the turbine trip.

The licensee discarded the 0.4 °F/second value and applied the 1.1 °F/second value for the first 60 seconds after the 14-second initial delay time, followed by a 1.4 °F/second rate thereafter. The NRC staff notes that using an initial rate of 1.1 °F/second is more conservative than 0.4 F/second. However, the licensee did not appear to directly consider the integrated FW flow or scale the FW injection time intervals from the BFN Unit 3 event data in its determination of a 60-second time interval for the 1.1 °F/second reduction rate. Additionally, the NRC staff identified concerns that the Unit 3 event data may have led to nonconservatively low FW temperature reduction rates relative to the postulated TTWBP ATWS event due to the relatively long (several minutes long) intervals of near-zero FW flow in the BFN Unit 3 event in-between the intervals of FW injection. These intervals would allow the latent heat from the sources listed in the previous paragraph to have additional time to transfer to the FW and increase the FW temperature prior to the next FW injection interval, which could result in significantly less FW temperature reduction for a given amount of FW mass injected to the vessel, as compared to the TTWBP ATWS-I event, which does not have these intervals of near-zero FW flow. This may be especially significant in the early stages of the event after the turbine trip. However, the NRC staff notes that additional factors may impact how the FW temperature evolves as a result of the near-zero FW flow intervals. For example, heat loss from the FW heater system to the surroundings would increase the rate of FW temperature reduction determined from the data, potentially counteracting or outweighing the effects discussed above.

In its response to SRXB-RAI 5c, the licensee provided a comparison of the assumed FW temperature reduction to the results of TTWBP ATWS-I simulator runs and demonstrated that the assumed FW temperature response in the ATWS-I analyses conservatively bounded the simulator predictions by a significant margin. Additionally, in Figure SRXB RAI-5-c-3, the licensee plotted the assumed ATWS-I FW temperature reduction and the measured FW temperature response during the BFN Unit 3 event against the integrated FW mass, which entered the vessel since the turbine trip. This plot indicates that the assumed FW temperature is significantly lower than the measured temperature at any given value of integrated FW mass. As discussed above, the relatively long near-zero FW flow intervals in the BFN Unit 3 event data could potentially introduce a positive (less conservative) bias in the FW temperature versus integrated FW flow behavior; however, the large conservative margin between the assumed and measured FW temperature behavior as shown by this plot provides reasonable assurance that the assumed FW temperature behavior is conservative overall.

Based on its review the information provided in the LAR (Reference 1) and the SRXB-RAI 5 response (Reference 6), the NRC staff has concluded that the FW temperature response used

in the TRACG analyses for the TTWB ATWS-I event is based on reasonable physical principles and conservatively bounds both measured data and simulator results for BFN. On this basis, the NRC staff finds the assumed FW temperature response for the ATWS-I TTWBP analyses to be acceptable.

In SRXB-RAI 5e, the NRC staff requested justification for the FW temperature response during the 2RPT ATWS event. In its RAI response, the licensee indicated that the assumed FW temperature response used in TRACG for the 2RPT ATWS event is consistent with the 2RPT TRACG runs supporting the DSS-CD reactor TH stability analyses presented in the SAR. This temperature response is based on a relationship between turbine steam flow and FW enthalpy, incorporating a lag into the FW enthalpy response with a 60-second time constant. For additional confirmation of validity, the licensee provided a comparison of the assumed FW temperature response with simulator runs for a 2RPT ATWS event, demonstrating that the assumed response conservatively bounds the simulator predictions.

Because this FW temperature response is included in the approved DSS-CD methodology and because the NRC staff determined that this approach remains applicable for BFN, the NRC staff finds the assumed FW temperature response for the ATWS-I 2RPT analyses to be acceptable.

The licensee determined that the 2RPT event is more limiting than the TTWBP event for BFN MELLLA+ based on the TRACG calculations. Results for several sensitivity cases for the 2RPT event were provided in Table 9-10 of the M+SAR, including combinations of the following sensitivities: nominal or bounding fuel parameter values, Modified Shumway or HN+CT T_{min} model, and quench model enabled or disabled. The staff issued SRXB-RAI 2 and SRXB-RAI 3 to examine the results of the TTWBP cases and ensure that this event remains bounded by the 2RPT event under every sensitivity scenario considered. In its RAI response, the licensee presented the requested TTWBP results. The NRC staff reviewed the RAI response and determined that the 2RPT results bound the TTWBP results under all sensitivity scenarios, including the T_{min} , fuel parameter and quench sensitivity studies considered in the LAR. These sensitivity studies, as well as further explanation of why the 2RPT event was limiting, are discussed in more detail later in this section.

Because the 2RPT results bound the TTWBP under all nominal and sensitivity studies considered, and because the NRC staff determined that the assumptions used for the 2RPT and TTWBP events, including assumed FW temperature reduction rate were acceptable, the NRC staff concludes that the 2RPT event is the limiting ATWS-I event for BFN MELLLA+. Therefore, the remainder of the evaluations and discussions for ATWS-I in this section consider the 2RPT event only.

The nominal ATWS-I results predict the occurrence of large-amplitude oscillations for roughly 150 seconds 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.

ATWS-I ATRIUM 10XM Fuel Parameter Sensitivity Analyses

The licensee provided an additional set of 2RPT results that accounts for the uncertainties of using TRACG04 with ATRIUM 10XM fuel (see discussion in Appendix D of this SE). A fuel parameter sensitivity study was performed by varying relevant modeling parameters within appropriate ranges of uncertainty. A summary of the process of selecting parameters,

determining sensitivity ranges, and determining limiting values for the ATWS and ATWS-I analyses is documented in Reference 2, with the NRC staff's evaluation of this process described in Appendix D of this SE. Based on this evaluation, the NRC staff confirms that the fuel parameter sensitivities acceptably account for ATRIUM 10XM performance during the ATWS-I analyses. Application of the limiting fuel parameter sensitivities results in a maximum PCT of [[

]]. The sensitivity results are discussed further in the next section.

ATWS-I T_{min} Sensitivity Analyses

Previous MELLLA+ reviews (e.g., Peach Bottom Atomic Power Station MELLLA+ (Reference 88)) 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 2,200 °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 87). Therefore, additional sensitivity calculations were requested by the NRC staff to examine the calculated behavior using the HN+CT model for T_{min} as proposed by Bjornard and Griffith (Reference 89), which provides a lower estimation of T_{min} , and therefore, predicts the occurrence of failure-to-rewet at lower cladding temperatures.

The licensee included sensitivity results using the HN+CT T_{min} model in Table 9-10 of the M+SAR. Using the HN+CT T_{min} model, nominal fuel parameter values, and quench model enabled, the maximum PCT for the 2RPT case was calculated to be [[

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Note that 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 coolant but of high local energy deposition and cladding heat flux. In the event that cladding temperatures exceed 2,200 °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 criteria to be satisfied under these circumstances. This position has been previously stated by the NRC staff in its evaluation of NEDO-32047-A (Reference 85).

The NRC staff examined an additional sensitivity case that used the HN+CT T_{min} model, quench model enabled, and bounding fuel parameter values. In this case, the increased oscillation growth rate meant that a cladding temperature of 2,200 °F was exceeded [[

]]. In SRXB-RAI 4, the NRC staff requested further explanation for why the fuel parameter sensitivities [[

]]. In its RAI response (Reference 6), the licensee presented detailed plots and discussion demonstrating that the primary reason for this [[

sensitivity study with a delayed operator action time of 240 seconds [[

]] An additional

operator actions had not prevented this from occurring. After examining this information in the RAI response, the NRC staff concurs with the licensee that the bounding fuel parameter sensitivities have only a small impact on the local cladding temperature given a particular set of local power and hydraulic conditions, and that the primary impact of the bounding fuel parameter sensitivities on PCT [[

]] if

fuel parameter sensitivities on the ATWS-I events allowed the NRC staff to more precisely evaluate the acceptability of BFN MELLLA+ for ATWS-I in terms of the nominal and bounding fuel parameter sensitivities.

]]. This understanding of the impact of bounding

Historically, the staff used the 10 CFR 50.46 criterion of 2,200 °F as a surrogate to demonstrate coolable geometry. 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 coolant but of high local energy deposition and cladding heat flux. In the event that cladding temperatures exceed 2,200 °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, in addition to the HN+CT T_{min} case with nominal fuel values, the NRC staff evaluated the sensitivity case using the HN+CT T_{min} model and limiting fuel parameter sensitivity values against these conditions.

Based on the evaluation described in Appendix D, 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. Therefore, the NRC staff evaluated the ATWS-I event based primarily on the nominal fuel parameter case with the HN+CT model, while also evaluating the bounding fuel parameter case with the HN+CT model as an additionally conservative case.

Under the conservative limiting fuel parameter assumptions, the NRC staff concludes that core coolability can still be reasonably expected to be maintained, given the relatively small number of fuel rods predicted to exceed 2,200 °F. This conclusion is consistent with previous staff evaluations for NEDO-32047-A (Reference 85) 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-I case in which localized cladding temperatures above 2,200 °F were predicted in as many as 88 bundles. For BFN 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 2,200 °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.

NRC TRACE/PARCS Sensitivity/Confirmatory Modelling Results

The TRACG results shown in the M+SAR and the licensee's response to SRXB RAI-4 (Reference 6) indicate that the PCT typically occurs at or shortly after the operator action time.

This is consistent with the NRC staff's prior experience. The assumed FW temperature response profiles lead to a FW temperature of 265.6 °F at the time of operator actions (120 seconds) in the TTWBP event and a FW temperature of 345 °F at the time of operator actions (180 seconds) in the 2RPT event. This significantly lower FW temperature in the TTWBP event may potentially promote more severe oscillations, more extensive failure to rewet, and higher PCT relative to the 2RPT event. However, as discussed previously, additional effects, including differences in system pressure in each event and the time required for oscillations to grow and develop, may contribute to a lower PCT in the TTWBP event than the 2RPT event. As a result, the limiting event is difficult to determine *a priori* and must be determined through analyses or other justification by considering the plant-specific configuration of BFN.

The confirmatory TRAC/RELAP Advanced Computing Engine (TRACE)/Purdue Advanced Reactor Core Simulator (PARCS) calculations performed by the NRC staff found the TTWBP event to be more limiting than the 2RPT event in terms of PCT, which was the opposite trend than shown in the TRACG calculations. However, a reasonable degree of difference in results is to be expected for these codes due to differences in the field equations, solution techniques, and constitutive relations. Furthermore, reasonable code-to-code differences may exist in the initial steady-state power distribution, initial flow rate distribution, and various assumptions impacting the dynamics of the vessel and balance of plant. All of these factors may lead to differences in the predictions of oscillation timing, growth rate, mode behavior, dryout-rewet behavior, and other phenomena. These phenomena are important for determining which ATWS-I event is more limiting.

For example, if the two codes predict different timings for failure to rewet relative to the operator action time, the PCT predicted by each code may differ greatly. This is because of the strong ability of operator actions to mitigate the occurrence of failure to rewet and the resulting PCT rise. This could be attributed to numerous different factors. Due in part to differences in radial power distribution between the codes, TRACE predicted that the in-phase mode was dominant during each event, while TRACG predicted that the out-of-phase mode was dominant. The NRC staff's experience indicates that the out-of-phase mode typically leads to larger flow rate variation and earlier failure to rewet in the limiting bundles than does the in-phase mode. A competing effect is that TRACG conservatively increased the hot rod peaking factors to put the limiting node at 95 percent of the LHGR limit at the initial statepoint, whereas TRACE used realistic values extracted from MICROBURN-B2 and did not apply this conservative peaking factor increase. This had the effect of delaying or preventing dryout and failure to rewet relative to TRACG.

The NRC staff issued SRXB-RAI 4 to obtain further clarification on what caused the 2RPT event to be more limiting than the TTWBP event in the TRACG analyses, given that the 2RPT event had a less limiting FW temperature response. In its RAI response, the licensee provided a sensitivity study with nominal and delayed operator actions for the TTWBP event. With delayed operator actions, the PCT quickly rose and exceeded 2,200 °F within a short timeframe after the time of peak PCT in the nominal operator action case. Therefore, although failure to rewet occurred in the TTWBP event with nominal operator action timing, the nominal operator action time of 120 seconds was sufficiently early to mitigate the majority of PCT rise that would have otherwise occurred. By contrast, in the 2RPT event, the oscillations had additional time to develop and led to more extensive failure to rewet before the PCT was reduced by operator actions.

Based on the above discussions, the NRC staff finds the TRACG prediction that the 2RPT event is the most limiting ATWSI event for BFN MELLLA+ to be reasonable, and that this finding highlights the importance of analyzing or otherwise justifying the limiting ATWSI event based on the actual plant-specific configuration.

To support the conclusion stated in this section, the NRC staff performed a plant-specific confirmatory analysis using TRACE/PARCS. The confirmatory study is summarized below. For full discussion of the confirmatory studies, see the NRC Memorandum from the Office of Nuclear Regulatory Research to the Office of Nuclear Reactor Regulation (NRR) (Reference 87).

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 rely 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 evaluated the 2RPT initiated ATWS-I event for BFN assuming nominal values for all parameters. The 2RPT is initiated at 0.1 second using a time-based trip. The 2RPT leads to a reduction in the CF rate.

The NRC staff's confirmatory analysis of a 2RPT ATWS-I indicates good agreement when compared to reference analyses submitted by the licensee in the M+SAR. The staff compared major plant parameters during the transient and found good agreement in all significant parameters such as level, power, and flow.

Two minor differences were observed between the two calculations. First, the TRACE calculations showed a lower PCT because a small thermal margin (MCPR approximately 1.04) was predicted, whereas the licensee calculations showed cyclic dryout/rewet. This difference is likely due to differences in the predicted oscillation mode (core-wide versus regional) but is minor and the NRC staff's calculations support the same conclusion that PCT remains relatively low during 2RPT ATWS-I.

Second, TRACE predicts MSIV closure late in the transient whereas the licensee calculations do not. This is due to a nuance of the TRACE simulation of the manual operator actions to drop and recover level during the event. In the licensee's calculations, as would be expected during an actual event, the MSIVs do not close because the EOIs direct the operators to bypass the MSIV closure trip if the main condenser is available. The NRC staff's calculation did not defeat this trip, resulting in MSIV closure late in the TRACE calculations. However, differences in the predicted level and MSIV closure during the late transient have no bearing on the analysis conclusion with respect to fuel damage consequences.

Considering the good agreement between the licensee and NRC staff analyses, and the large PCT margins that have been demonstrated, the staff concludes that the confirmatory analyses support the licensee's conclusion that 2RPT ATWS-I does not lead to excessive fuel damage consequences that could compromise core coolability.

For the TTWBP, calculations are initiated from 100 percent rated thermal power at 85 percent rated flow at a point early in the cycle with a strong downward axial peaking and a low first harmonic eigenvalue separation. In both cases, the operators intervene to mitigate the event with actions to reduce water level and inject boron through the SLCS at 120 seconds. The cases differ in terms of the feedwater temperature reduction (FWTR) transient following the turbine trip. In the first case, the FWT transient following the turbine trip is shown by a BFN-specific curve where the FWT changes at various rates during different periods. This curve has the FW temperature decrease at a rate that is generally faster than 1.0 °F/second. In the second case, the NRC staff performed a sensitivity study with a slower FWT curve. The approximate rate of decrease is 0.5 °F/second in the sensitivity calculation.

The results of the NRC staff's confirmatory analysis indicate that the TTWB ATWS-I scenario produces more adverse conditions compared to an ATWS-I initiated by a 2RPT. The reason is that when the BFN-specific FWT curve is considered, the core inlet subcooling becomes higher in the TTWB ATWS-I case when compared to the 2RPT ATWS-I case. Even accounting for a more rapid operator response time to lower level and inject boron, the TTWBP ATWS-I case shows power oscillations with a higher amplitude and ultimately a higher PCT. However, the NRC staff found that large PCT margin is maintained with the PCT reaching approximately 840 °K compared to a fuel damage threshold of 1,478 °K.

A sensitivity calculation performed by the staff indicates that a slower FWT reduction produces less adverse conditions in terms of power and PCT. Additionally, when the TTWBP case using the BFN-specific FW temperature curve failed early, the sensitivity calculation demonstrates the expected plant response for the full transient calculation time, including the restoration of feed flow. The results confirm that reactor power is expected to continue to decrease with the power oscillations suppressed. This provides further evidence that running the TTWBP with the BFN-specific FWT curve beyond approximately 200 seconds would not be expected to yield more limiting results.

The NRC staff completed a confirmatory analysis for an ATWS-I event that is initiated by a 2RPT and compared the results of a base case and a sensitivity case where the inlet orifice loss coefficients for all the channels are reduced by 10 percent. The base case and sensitivity cases are run with MSIV closure defeated.

The results of the NRC staff's confirmatory analysis indicate that reducing the core inlet orifice loss coefficients is destabilizing. This is the expected trend. The confirmatory analysis with TRACE shows that lower inlet losses lead to the reactor power oscillation amplitude growing more quickly once the reactor becomes unstable. As a result, thermal margins are eroded more quickly, but TRACE predicts that the MCPR remains above unity, leading to mild PCT consequences.

Differences between TRACE and TRACG predictions of dryout can be explained by differences in the assumed hot rod peaking factor between the two methods. The TRACE result is a realistic peaking factor that is tuned to match the maximum peak power from the licensee's nuclear design analysis. The TRACG hot rod peaking factor, on the other hand, is based on the maximum linear heat generation rate limit. This means that the hot rod peaking in the TRACG calculation is as much as 20 percent higher than the TRACE calculation, which would essentially erase the known conservatism in the TRACE critical power prediction.

The TRACE calculations, however, show that peak power and limit-cycle oscillation magnitude remain unchanged by differences in the channel inlet loss. This result is also expected because

the non-linear effects that constrain the gross core average power remain unchanged between the base and sensitivity cases. As a result, the ultimate power oscillation magnitude is essentially the same in both cases. Therefore, while the TRACE calculations do not show any fuel heatup, the NRC staff can infer that if heatup were predicted, the effect may be to slightly advance the timing of that heatup in the sensitivity case, but that ultimately, the PCT consequences would be largely similar. This inference agrees with the licensee's sensitivity calculations that show the PCT increases by less than 25 °K when a lower inlet orifice loss is assumed in the TRACG calculations.

The NRC staff completed a confirmatory analysis for an ATWS-I event initiated by a 2RPT and compared the results of a base case and a sensitivity case where the gap conductance for all the rod groups in all channels is increased by 25 percent.

The results of the NRC staff's confirmatory analysis indicate that the transient is largely insensitive to the gap conductance. The calculation results indicate that the power instability may occur slightly earlier in the case with a higher gap conductance. This is the expected trend. The confirmatory analysis with TRACE shows that the MCPR remains above unity; therefore, TRACE predicts that the PCT is approximately 570 °K in both cases.

The NRC staff's results show a very minor difference in the power behavior and resultant PCT consequences for an increase of 25 percent in the gap conductance. The trends observed in the NRC staff's calculations agree with the results of the licensee's calculations that show an increase in the gap conductance produces a small (7 °K) reduction in the predicted PCT.

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

Conclusions on ATWS

The NRC staff concludes that the plant design and operator actions adequately address ATWS events and meet the requirements of 10 CFR 50.62. This conclusion is based on the following:

- The licensee'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 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 and that unacceptable plant conditions, as defined during the rulemaking, will not occur because of ATWS events.

3.10 BFN M+SAR Section 10.0, "Other Evaluations"

3.10.1 BFN M+SAR Section 10.1, "High Energy Line Break"

3.10.1.1 BFN M+SAR Section 10.1.1, "Steam Lines"

As discussed in M+SAR (Attachment 6 of (Reference 1)) Section 10.1.1, the licensee confirmed that the generic disposition in the M+LTR for the HELB steam lines topic is applicable to BFN. Specifically, MELLLA+ operation has 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 disposition is applicable to BFN because MELLLA+ has no effect on the mass and energy releases from an HELB in a steam line.

3.10.1.2 BFN M+SAR Section 10.1.2, "Balance-of-Plant Liquid Lines"

As discussed in M+SAR (Attachment 6 of (Reference 1)) Section 10.1.2, the licensee confirmed that the generic disposition in the M+LTR for the HELB BOP liquid lines topic is applicable to BFN. Specifically, a review of the heat balances produced for BFN MELLLA+ operation confirmed there is no effect on the liquid line conditions at the postulated FW and RWCU break locations. A review of the BFN design basis confirms that there are no additional high energy lines at BFN and that MELLLA+ does not create any new HELB locations. In addition, a small increase (6.8 percent) in mass release and a 3.8 percent in energy release for operation in the MELLLA+ domain due to RWCU line breaks do not result in any reactor building room or general area exceeding the current pressure or relative humidity profiles used for structural or equipment qualification.

The NRC staff concludes that the generic M+LTR disposition is applicable to BFN because the current operating conditions, which are CFs and FW temperature lower than the MELLLA+ domain from an HELB in BOP liquid lines, are bounding.

3.10.1.3 BFN M+SAR Section 10.1.3, "Other Liquid Lines"

As discussed in M+SAR (Attachment 6 to the LAR (Reference 1)) Section 10.1.3, the licensee confirmed that the generic disposition in the M+LTR for the HELB other liquid lines topic is applicable to BFN. Specifically, a review of the BFN design basis confirms that there are no additional high energy lines beyond those covered by M+SAR Section 10.1.2.

The NRC staff concludes that the generic M+LTR disposition is applicable to BFN because there are no other high energy lines that require evaluation.

RWCU Line Break

The licensee evaluated RWCU line break under M+LTR Section 10.1.3, "Other Liquid Lines." The generic M+LTR treatment for HELB states [[

]] However, the licensee stated that the RWCU system (core inlet enthalpy) for the MELLLA+ 77.6 percent CLTP/55 percent reactor CF statepoint is not bounded by CLTP operation with final FW temperature reduction/ FWHOOS. The heat balances for this condition are provided in Table 1-3 of M+SAR (Attachment 6 to the LAR

(Reference 1)). New mass and energy releases for RWCU line breaks were analyzed for the more limiting MELLLA+ condition. The results indicated increases of 6.8 percent in mass release and 3.8 percent in energy release, approximately. The licensee stated that these small increases did not result in any reactor building room or general area exceeding the current governing pressure and relative humidity profiles used for structural or equipment qualification.

Based on the specific line break analysis conducted by the licensee, the NRC staff concludes that the deviation from generic M+LTR treatment was satisfactorily addressed.

Pipe Whip and Jet Impingement

Pipe whip and jet impingement loads resulting from high energy pipe breaks are a function of system pressure, temperature, and size, as well as proximity to relatively constant pressure sources connected to the line and the effect of friction or line area restrictions between the break and the constant pressure source.

The licensee in Section 10.1.3.2 of the M+SAR confirmed that MELLLA+ results in no systems (either inside or outside containment) experiencing an increase in operating pressure or operating temperature. Therefore, MELLLA+ results in no increase in pipe whip and jet impingement loads from the current design-basis analyses.

Therefore, the NRC staff concludes that pipe whip and jet impingement meet the M+LTR generic disposition.

HELB Flooding

The licensee stated that the most limiting HELB flooding event for BFN is a circumferential break of a FW line in the MS tunnel and that the event conservatively assumes that the entire condenser hotwell inventory is discharged from the break location. Since MELLLA+ does not result in any input parameters such as FW pressure and flow, hotwell capacity, and condensate and FW pumping capacity for the FW break event, the licensee stated flood levels are unchanged with the expansion of the BFN operating domain to MELLLA+.

The NRC staff finds the logic and reasoning provided by the licensee that the current basis for the most limiting HELB flooding is unaffected by MELLLA+ operating domain expansion acceptable.

3.10.2 BFN M+SAR Section 10.2, "Moderate Energy Line Break"

3.10.2.1 BFN M+SAR Section 10.2.1 - Moderate Energy Line Break – Flooding

The licensee confirmed that the generic M+LTR treatment of the moderate energy line break flooding is applicable to BFN. Specifically, auxiliary system flow rates and inventories, including operating modes, are not affected by MELLLA+ operating domain expansion.

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

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3.10.3 BFN M+SAR Section 10.3, "Environmental Qualification"

3.10.3.1 BFN M+SAR Section 10.3.1, "Electrical Equipment"

The licensee confirmed in Section 10.3.1 of the M+SAR (Attachment 6 to the LAR (Reference 1)) that the reactor power does not increase as a result of MELLLA+ operating domain expansion, and there is no change in radiation levels in any of the plant areas where safety-related equipment is located and no changes in decay heat. The licensee also confirmed that for BFN, there are no increases in reactor operating pressure, MS flow rate, or FW flow rate. The licensee also stated that [[

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

3.10.3.2 BFN M+SAR Section 10.3.2, "Mechanical Equipment with Non-Metallic Components"

The licensee confirmed in Section 10.3.2 of the M+SAR (Attachment 6 of LAR) that for BFN, normal process temperatures are not affected by MELLLA+ and there is no change in radiation levels in any of the plant areas where safety-related equipment is located. [[

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

3.10.3.3 BFN M+SAR Section 10.3.3, "Mechanical Component Design Qualification"

The licensee confirmed in Section 10.3.3 of the M+SAR (Attachment 6 of LAR) that for BFN, normal process temperatures, pressures, and flow rates are not affected by MELLLA+ and there is no change in radiation levels in any of the plant areas where safety-related equipment is located. [[

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

3.10.4 BFN M+SAR Section 10.4, "Testing"

As discussed in the M+SAR (Reference 41), 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 M+SAR (Attachment 6 to the LAR (Reference 1)) provides a brief description of plant-specific testing for implementation of the BFN MELLLA+. The following testing will be performed:

- 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 CF 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 since they will help confirm that plant operation is consistent with the analysis to support the operation in the proposed MELLLA+ domain.

3.10.5 BFN M+SAR Section 10.5, "Individual Plant Examination"

The NRC staff reviewed the regulatory and technical analyses performed by the licensee in support of its proposed expanded operating domain identified as MELLLA+ as they relate to the licensee's plant-specific probabilistic risk assessment (PRA) evaluation. The evaluation scope included all the elements of Section 10.5, "Individual Plant Examination," of the generic M+LTR. The evaluation included an assessment of the impacts on core damage frequency (CDF) and large early release frequency (LERF). The following sections provide the NRC staff's technical evaluation of the risk information provided by the licensee.

As discussed in M+SAR Section 10.5, "Individual Plant Examination," the licensee provides a discussion of an assessment of the risk increase, including CDF and LERF, associated with operation in the MELLLA+ operating domain. The topics addressed in this evaluation are: Initiating Event Categories and Frequency; Component and System Reliability; Operator Response; Success Criteria; External Events; Shutdown Risk; and PRA Quality.

3.10.5.1 BFN M+SAR Section 10.5.1, "Initiating Event Categories and Frequency"

The MELLLA+ expanded operating domain involves changes to the operating power/CF map and a small number of setpoints and alarms. The licensee explained that the MELLLA+ changes do not result in any new initiating events because the changes are not extensive and are not of the type that could result in a new initiating event. The power conversion systems, electrical systems, and other auxiliary systems are not changed as a result of MELLLA+ operation. MELLLA+ operation affects the core and some aspects of the NSSS, including moisture content in the steam, but it does not change thermal power or normal operating pressure, and the steam flow, FW flow, and FW temperature are essentially unchanged. Therefore, the licensee concluded that the MELLLA+ expanded operating domain does not significantly change the probability of an instability event, and thus, the effect of the stability region modifications on the PRA is negligible.

The NRC staff finds that the information provided by the licensee adequately addresses this element of Section 10.5, "Individual Plant Examination," of the generic M+LTR. Based on the information provided, the NRC staff concludes that the proposed changes to implement the MELLLA+ do not include changes to plant hardware or operating procedures that would create additional event categories or have a significant effect on initiating event frequencies.

3.10.5.2 BFN M+SAR Section 10.5.2, "Component and System Reliability"

The licensee stated in M+SAR (Attachment 6 to the LAR (Reference 1)) Section 10.5.2, "Component and System Reliability," that the expanded operating domain does not affect the system or component reliability, and therefore, does not impact CDF and LERF results. There is no change in the operating pressure and power, and the steam and FW flow rates are essentially unchanged. The MELLLA+ expanded operating domain does not require major plant hardware modifications. The licensee explained that the TSs ensure that plant and system performance parameters are maintained within the values assumed in the safety analyses. The TS setpoints, allowable values (AVs), operating limits, and requirements are selected such that the equipment parameter values are equal to or more conservative than the values used in the safety analysis.

The NRC staff finds that the information provided by the licensee adequately addresses this element of Section 10.5, "Component and System Reliability," of the generic M+LTR. Based on the information provided, the NRC staff concludes that the impact of the MELLLA+ expanded operating domain would be negligible on component and system reliability, and therefore, on the PRA, CDF, and LERF results.

3.10.5.3 BFN M+SAR Section 10.5.3, "Operator Response"

The licensee stated in M+SAR (Attachment 6 to the LAR (Reference 1)) Section 10.5.3, "Operator Response," that operator responses to anticipated occurrences, accidents, and special events under MELLLA+ operating conditions are principally the same as for current operating conditions. Operating under MELLLA+ conditions does not cause changes to any of the automatic safety actions. After the applicable automatic responses have initiated, the post-event operator actions for plant safety remain the same under MELLLA+ operating conditions. Since decay heat is unchanged, the time for boil-off is unchanged. Therefore, long-term core cooling is not affected by the MELLLA+ expanded operating domain.

The licensee discussed how plant conditions under an ATWS are potentially more severe post-reactor RPT during MELLLA+ operation. The licensee explained that at reduced flow rates with power at 120 percent of the OLTP, the post-RPT power level may be slightly higher during MELLLA+ operation. At the slightly higher power level, the RPV water level would potentially drop faster, and containment heatup would initially be faster. This circumstance could potentially reduce the time for operator response to the event. Therefore, the licensee evaluated the operator response time to an ATWS to assess the impact to the PRA. The licensee found the impact to the PRA is negligible because there was very little effect on operator response times and there were no changes to the human error probabilities associated with short-term ATWS response actions.

The NRC staff finds that the information provided by the licensee adequately addresses this element of Section 10.5, "Operator Response," of the generic M+LTR. Based on the information provided, the NRC staff concludes that the impact of the MELLLA+ expanded operating domain would be negligible on operator response, and therefore, on the PRA CDF and LERF results.

3.10.5.4 BFN M+SAR Section 10.5.4, "Success Criteria"

The licensee stated in M+SAR (Attachment 6 to the LAR (Reference 1)) Section 10.5.4, "Success Criteria," that systems success criteria credited in its PRA to perform the critical safety functions were analyzed based on the MELLLA+ expanded operating domain. The critical safety functions are as follows: reactivity control, overpressure control, vessel depressurization, reactor coolant makeup, and containment heat removal. The licensee explained the expanded operating domain involves changes to the operating power/CF map and a small number of setpoints and alarms. There is no change in the operating pressure and power, and the steam and FW flow rates are essentially unchanged. The MELLLA+ expanded operating domain does not impose any additional requirements on any of the safety, balance-of-plant, electrical, or auxiliary systems. Adequate MSRV capacity is provided to ensure that the ATWS overpressure requirement for MELLLA+ is satisfied. Therefore, MELLLA+ operating domain expansion will not affect PRA success criteria.

The NRC staff finds that the information provided by the licensee adequately addresses this element of Section 10.5, "Success Criteria," of the generic M+LTR. Based on the information provided, the NRC staff concludes that the impact of the MELLLA+ expanded operating domain would not affect the success criteria of the PRAs.

3.10.5.5 BFN M+SAR Section 10.5.5, "External Events"

As discussed in M+SAR Section 10.5.5, "External Events," the expanded operating domain is not expected to significantly affect the elements discussed above of an internal event PRA. The licensee evaluated the external hazards group and determined that MELLLA+ will not affect the external events hazard groups and determined there is no effect on the PRA external events risk. The licensee concluded the MELLLA+ operating domain expansion does not affect the external events analysis.

The NRC staff finds that the information provided by the licensee adequately addresses this element of Section 10.5.5, "External Events," of the generic M+LTR. Based on the information provided, the NRC staff concludes that the impact of the MELLLA+ expanded operating domain would not affect the external events PRAs.

3.10.5.6 BFN M+SAR Section 10.5.6, "Shutdown Risk"

The licensee stated in M+SAR (Attachment 6 to the LAR (Reference 1)) Section 10.5.6, "Shutdown Risk," that the expanded operating domain does not change the shutdown conditions; therefore, it has no effect on the plant PRA shutdown risks.

The NRC staff finds that the information provided by the licensee adequately addresses this element of Section 10.5, "Shutdown Risk," of the generic M+LTR. Based on the information provided, the NRC staff concludes that the impact of the MELLLA+ expanded operating domain would not change shutdown conditions modeled in the PRA, and therefore, has no effect on the plant PRA shutdown risks.

3.10.5.7 BFN M+SAR Section 10.5.7, "Probabilistic Risk Assessment Quality"

The licensee stated in M+SAR (Attachment 6 to the LAR (Reference 1)) Section 10.5.7, "PRA Quality," the base reference model used in this risk assessment is Revision 7 to the BFN Level 1 and Level 2 PRA average maintenance model. This model includes EPU-implemented plant modifications. The BFN full power internal events PRA models and documentation have been updated to reflect the current plant configuration and to reflect the accumulation of additional plant operating history and component failure data.

The BFN internal events PRA was subject to three peer reviews: (1) a full scope review, (2) a focused scope follow-on peer review for internal flooding, and (3) a focused scope peer review to evaluate specific aspects of the internal events PRA and assess existing findings and observations dispositions. The purpose of these reviews was to provide a method for establishing the technical adequacy of the BFN PRA for the spectrum of potential risk-informed plant licensing applications for which the BFN PRA may be used. These reviews provided a full-scope review of the technical elements of the internal events and internal flooding for at-power conditions. The licensee indicated there have been no changes made to the internal events model following these peer reviews that would constitute an upgrade, and thus, the model does not require another focused scope peer review.

The NRC staff concludes that the BFN Levels 1 and 2 PRAs provide a sufficient level of scope and detail to measure the potential changes of CDF and LERF due to implementation of MELLLA+ operating domain expansion.

3.10.5.8 Probabilistic Risk Assessment Conclusion

Based on the evaluation in Section above 10.5 of the M+SAR (Attachment 6 to the LAR (Reference 1)), the licensee indicates there is no change in the baseline CDF, and therefore, there is no incremental increase in CDF for BFN Units 1, 2, and 3. Based upon this evaluation, the licensee indicates there is no change in the baseline LERF, and therefore, there is no incremental increase in LERF for BFN Units 1, 2, and 3. The licensee applied NRC guidelines established in RG 1.174 to the calculated results produced from the Levels 1 and 2 PRAs. The licensee found the mean value for the CDF risk increase and the mean value for the LERF increase were within Region III, and thus, concludes the changes represent a very small change in risk. Based on the risk results from the plant-specific PRA evaluation, the licensee concludes, and the NRC staff agrees, that BFN operation within the proposed MELLLA+ operating domain is acceptable.

3.10.6 BFN M+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 is as follows:

- NUREG-0800 (Reference 21), “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants,” Chapter 18
- NUREG-1764 (Reference 90), “Guidance for the Review of Changes to Human Actions”
- NUREG-0711 (Reference 91), “Human Factors Engineering Program Review Model”

The NRC staff reviews the human performance aspects of LARs utilizing the review guidance in NUREG-1764, Revision 1, “Guidance for the Review of Changes to Human Actions.” 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 (HRA).

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

Section 10.5.3, “Operator Response,” of the M+SAR (Attachment 6 to the LAR (Reference 1)) states, in part:

The operator responses to anticipated occurrences, accidents, and special events for CLTP with MELLLA+ conditions are basically the same as for CLTP conditions.

The licensee stated in Section 10.6, “Operator Training and Human Factors,” of the M+SAR that two new operator actions are also required to operate in the MELLLA+ domain. The licensee stated in its supplement dated December 14, 2018 (Reference 5), that these new actions are associated with new TSs and renewed facility operating license (RFOL) condition restrictions to ensure operation is maintained within MELLLA+ analyzed conditions. The new actions are an extension of existing operator actions and will prohibit operation in the MELLLA+ region of the power to flow map when operating in SLO or when operating with FWT reduced to over 10 °F below the feedwater design temperature.

Operating Experience Review

The licensee stated that the BFN MELLLA+ LAR used the following NRC-approved GEH LTRs:

- NEDC-33006P-A (Reference 17), (M+LTR), Revision 3, and its associated SER
- NEDC-33173P-A (Reference 19), (Methods LTR), Revision 4, and its associated SER
- NEDC-33075P-A (Reference 18), (DSS-CD LTR), Revision 8, and its associated SER

In addition to using NRC-approved methodologies to develop the BFN MELLLA+ implementation at BFN, the licensee provided a list with summaries of industry precedents considered, including an amendment for the Monticello Nuclear Generating Plant, letter dated March 28, 2014 (Reference 92); an amendment for the Grand Gulf Nuclear Station, letter dated August 31, 2015 (Reference 93); an amendment for the Nine Mile Point Nuclear Station, Unit No. 2, letter dated September 2, 2015 (Reference 94); and amendments to Peach Bottom Atomic Power Station, letter dated March 21, 2016 (Reference 88). Subsequent to the submittal of the BFN MELLLA+ LAR, the MELLLA+ amendments for Brunswick Steam Electric Plant, Units 1 and 2 (Reference 95), were issued.

The licensee's supplement dated March 13, 2019 (Reference 9), addressed site-specific operating experience related to the implementation of MELLLA+ at BFN. Specifically, the supplement addressed the January 31, 2019 (Reference 96), BFN Integrated Inspection Report Green inspection finding regarding a high crew failure rate during 2018 wherein 4 of 15 operating crews failed their fifth operating cycle annual requalification simulator scenario examinations. Two of the four crew failures involved ATWS scenarios associated with MELLLA+ operating conditions.

A Performance Analysis was performed by BFN Operations and Operations Training to determine the specific causes of the failures. The Performance Analysis identified that the crew simulator failures were not due to programmatic training deficiencies. Rather, the failures were due to inattention to detail, misjudgment, and knowledge gaps attributed to the specific crews involved in the simulator failures.

During one crew simulator failure, the crew initially started an emergency depressurization with an ATWS but did not recognize that low pressure injection systems were capable of injection to the RPV. Subsequent injection to the RPV occurred prior to being authorized. This crew simulator failure was associated with a low-power ATWS such that the initial operator actions for ATWS mitigation were not required to be performed.

The second crew failure involving an ATWS scenario was attributed to a knowledge gap on the part of the Senior Reactor Operator (SRO) wherein the SRO misapplied an override step in the emergency operating instruction (EOI) flowchart during an ATWS. Upon receiving a report that reactor power was below the downscale setpoint, the SRO erroneously directed injection to the RPV prior to RPV water level reaching the level required to minimize core inlet subcooling, resulting in a failed critical task. However, the failure occurred subsequent to the crew successfully performing the initial operator actions for the ATWS, including termination of RPV injection and injection of SLC within the times assumed in the MELLLA+ ATWS analyses.

The NRC staff dispositioned the crew requalification simulator scenario examinations described above in the BFN Integrated Inspection Report issued January 31, 2019 (Reference 96). The NRC staff screened the issue as having very low safety significance (Green) because the deficiency was identified and corrected in a timely manner. As discussed above, only two of the four identified crew failures involved an ATWS scenario, and in one case, the initial operator actions for ATWS mitigation were not required to be performed. For the single crew failure that did require performing the initial operator actions for the ATWS, the crew successfully performed the actions within the required immediate operator action time limits. Therefore, the NRC staff finds that the excessive crew failures reported in the subject inspection report have been appropriately evaluated and corrected by the licensee such that they do not adversely impact the implementation of the MELLLA+ operating domain at BFN.

The NRC staff concludes that the licensee's analysis using NRC-approved methodology, industry precedents, and site-specific operating experience is acceptable to address relevant 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 BFN. 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 requires revisions for the planned configuration change.

Implementation of MELLLA+ at BFN does not replace any existing automatic functions with manual actions or vice versa. However, a new automatic function, ABSP, is being added by the power range neutron monitoring (PRNM) system modification 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 amendments.

Task Analysis

The supplement dated December 14, 2018, stipulates that operators are required to initiate SLC and the lowering of reactor water level to mitigate ATWS instability events within 120 seconds. In addition, the supplement dated January 16, 2019 (Reference 7), states that an operator response time of 180 seconds is required from a dual recirculation pump trip (2RPT) to the initiation of SLC and the reduction of reactor water level following an ATWS. These actions are controlled as immediate actions per an abnormal operating instruction (AOI) hard card located in the control room.

BFN operations with the MELLLA+ improvements do not change the required operator actions or significantly reduce the time for operator actions. The licensee's supplement dated December 14, 2018 (Reference 5), addressed the potential for reduced operator time to respond due to the possibility of slightly higher reactor power during ATWS post-RPT. BFN performed an HRA evaluation regarding the operator actions taken during ATWS post-RPT under MELLLA+ conditions. The evaluation considered potential changes in cues and indications, procedures, training, timing analysis, cognitive analysis, the application of human performance shaping factors, and stress. The evaluation determined that operator cues and indications and operator timing are unchanged. The time specified for successful completion of the ATWS actions remained the same, and there was no impact on the allotted time to complete the ATWS actions, although there is a potential for slightly higher ATWS power post-RPT when operating in the MELLLA+ domain. The evaluation also determined that operator stress levels were not increased, and performance shaping factors such as environmental factors, complexity of response, and equipment accessibility were not impacted. The evaluation concluded that the ATWS EOs and operator training regarding ATWS actions were unchanged for operation in the MELLLA+ domain.

The licensee utilized the HRA qualitative evaluation results to perform a sensitivity study wherein the human error probability (HEP) was increased by 10 percent for each of the affected human failure events in the PRA model. The licensee reported that the results of the sensitivity

study showed a very small increase to core damage frequency and large early release frequency probabilities on the order of 1E-8, confirming that the HEPs for actions taken during ATWS post-RPT under MELLLA+ conditions in the PRA model are bounded by the current assumptions in the baseline PRA.

The supplement dated December 14, 2018 (Reference 5), also stated that the two new operator actions are an extension of existing operator actions to prohibit operation in the MELLLA+ region of the power to flow map when operating in SLO or when operating with FWT reduced greater than 10 °F below the feedwater design temperature. However, the new restrictions are not associated with any ATWS or 2RPT ATWS-I operator mitigation actions. Therefore, operator response time and operator mitigation actions for ATWS or 2RPT ATWS-I are not impacted or changed. Plant procedures will include cues to indicate that the new operating parameter restrictions have been met such that operator action is required. As there have been no fundamental 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 consistent with existing operator actions and are proceduralized. In addition, the actions are straightforward and do not require changes to physical interfaces.

Staffing

The licensee stated that no new or additional operator actions are required to mitigate ATWS events with operation in the MELLLA+ domain at BFN, nor are there any new or additional qualifications required to perform the associated actions within the unchanged time constraints. Therefore, operation in the MELLLA+ domain is not expected to increase operator workload. Because no additional staffing, 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," of this SE, 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 BFN 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 BFN involves changes to the main control room computer display of the power/flow map. In addition, the PRNM system requires hardware and software changes through implementation of the DSS-CD solution, including an ABSP. Therefore, some limited 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 2019 (Reference 97), 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 amendments, upon approval, the licensee stated that necessary changes to procedures will be consistent with existing BFN 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 BFN MELLLA+ implementation plan. Per Section 10.6, "Operator Training and Human Factors," of the M+SAR (Attachment 6 to the LAR (Reference 1)), BFN simulator changes and fidelity validation will be performed in accordance with ANSI/American Nuclear Society (ANSI/ANS) 3.5-2009.

As part of the implementation of the amendments, upon approval, the licensee is required to update the BFN training program in accordance with 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 plant dynamics do not substantially change for MELLLA+ operation, enhanced simulator 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 BFN simulator has been updated to reflect the MELLLA+ analysis to support the implementation of the amendments. 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 with the MELLLA+ analysis.

The supplement dated July 23, 2018 (Reference 3), provided the average operating crew response time for SLC pump initiation as 44 seconds and the average time for commencing reactor water level reduction by reducing FW flow as 41 seconds from the recognition of an ATWS. The supplement dated December 14, 2018 (Reference 5), provided the raw results for the 16 BFN operating shift crews and showed the longest recorded crew response time as 70.75 seconds, which is well within the required 120-second response time.

The supplement dated January 16, 2019 (Reference 7), reported the raw and average operating crew times associated with the additional 60-second response time requirement following the 2RPT (total of 180 seconds). The longest crew response time from the raw results provided in the December 14, 2018, supplement was added to the longest crew response time in the January 16, 2019, supplement to determine a conservatively representative total ATWS with 2RPT response time of 101.7 seconds to reduce reactor water level and 112.0 seconds to initiate SLC. These conservatively determined operator response times remain well within the 180-second requirement.

The licensee stated that for both sets of BFN operating shift crew response time validation results, all crews completed the actions well within the required times with no problems or failures identified and that no remediation was required.

In addition, during the February 27, 2019, simulator audit (Reference 97), the NRC staff observed ATWS scenarios for timing validation. The demonstrations showed the operators successfully initiating FW flow reduction and SLC initiation well within the allowed time. The NRC staff also observed operating crew tabletop exercises involving TSs and plant procedures regarding OPRM upscale function inoperable (transfer to automatic backup stability protection) and loss of OPRM upscale function and automatic backup stability protection. These demonstrations support a finding of reasonable assurance that the actions are feasible and can be reliably performed 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 BFN MELLLA+ implementation plan will determine the changes required to implement MELLLA+ consistent with TVA 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+.

Based on the above, the NRC staff finds the licensee's treatment of human performance monitoring strategy to be acceptable.

Overall, the NRC staff finds the proposed amendments to be acceptable with respect to operator training and human factors.

3.10.7 BFN M+SAR Section 10.7, "Plant Life"

3.10.7.1 BFN M+SAR Section 10.7.1, "Irradiated Assisted Stress Corrosion Cracking"

Section 10.7.1 of the M+SAR (Attachment 6 to the LAR (Reference 1) addresses the potential impact of MELLLA+ on susceptibility of reactor vessel internal (RVI) components to irradiated assisted stress corrosion cracking (IASCC) and associated implementation of plant programs and activities for managing the effects of aging on RVI components during the period of extended operation. The licensee identified that the decrease in projected neutron fluence for the MELLLA+ application decreases the potential for IASCC for several RVI components.

The NRC staff noted that any potential for an increase in the susceptibility of RVI components to IASCC would require one or more of the following:

- an increase in component tensile stress due to a significant increase in the loadings on RVI components for normal operating conditions;
- a change in the reactor cooling water environment to more oxidizing conditions; and/or
- a significant increase in projected neutron fluence exposure (i.e., an increase in projected irradiation damage to the RVI materials caused by high-energy neutrons.)

For MELLLA+ conditions, the NRC staff confirmed that none of these factors would be adversely impacted for determining the susceptibility of RVI components to IASCC. The staff verified that operation in the MELLLA+ domain will have no adverse impact on the aging of any RVI component during plant life. The NRC staff determined that the licensee's continued implementation of applicable EPRI BWRVIP inspection and evaluation guidelines during the period of extended operation, as specified in applicable aging management program documents and site procedures, would continue to provide for acceptable aging management of RVI components for MELLLA+ conditions. This includes aging effects due to IASCC, as well as all other aging mechanisms that may be applicable for BWR internals. The NRC staff's detailed evaluation and acceptance of the licensee's continued implementation of the BWRVIP inspection and evaluation guidelines to manage aging effects associated with IASCC and other aging mechanisms for EPU conditions during the period of extended operation are addressed in Section 2.1.3 of its SE for the BFN EPU amendments, dated August 14, 2017 (Reference 16). The NRC staff's evaluation and findings documented in its EPU SE will continue to be applicable and valid for MELLLA+.

3.10.7.2 BFN M+SAR Section 10.7.2, "Flow Accelerated Corrosion"

The licensee confirmed that the generic disposition in the M+LTR for the flow-accelerated corrosion (FAC) topic is applicable to BFN. Specifically, for BFN, 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 M+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 CF. [[

licensee stated that:

]] The

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 BFN by utilization of the Electric Power Research Institute generic program, CHECWORKS™. BFN-specific parameters are entered into this program to develop requirements for monitoring and maintenance of specific system components. No changes are required to the BFN-specific parameters that are entered into the CHECWORKS™ 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 BFN equipment is not affected by the MELLLA+ operating domain expansion.

Because FAC under MELLLA+ operating conditions is bounded by the current plant operation, the NRC staff concludes that the generic M+LTR disposition is applicable to BFN. Therefore, the NRC staff concludes the licensee's disposition of the effects of FAC due to MELLLA+ is acceptable.

3.10.8 BFN M+SAR Section 10.8, "NRC and Industry Communications"

The licensee confirmed that the generic disposition in the M+LTR for the NRC and industry communications topic is applicable to BFN. Specifically, the M+LTR states that NRC and industry communications could affect the plant design and safety analyses. However, the evaluations and calculations included in the M+SAR, along with any supplements, demonstrate that operating in the MELLLA+ domain can be accomplished within the applicable design criteria. Because these evaluations of plant design and safety analyses had already included NRC and industry communications of related topics, it is not necessary to review prior communications as part of the MELLLA+ review.

The NRC staff concludes that the licensee's disposition of NRC and industry communications is acceptable because there is no additional information to suggest BFN falls outside the applicability scope of the original M+LTR.

3.10.9 BFN M+SAR Section 10.9 "Emergency and Abnormal Operating Procedures"

Per Section 10.9.1, "Emergency Operating Procedures," of the M+SAR (Attachment 6 to the LAR (Reference 1)), EOPs are designated as EOIs at BFN. The licensee stated in Section 10.9.2, "Abnormal Operating Procedures," of the M+SAR that abnormal operating procedures are designated as AOIs at BFN.

The licensee stated that EOIs and AOIs can be affected by operating in the MELLLA+ domain. The EOIs include variables and limit curves, which define conditions where operator actions are indicated. The EOIs are symptom-based. AOIs include event-based operator actions.

The licensee also stated that the EOIs and AOIs 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 amendments.

The NRC staff concludes that the licensee's processes for updating procedures and training operators are consistent with the requirements in 10 CFR 50.59 and 10 CFR 50.120, and are sufficient with respect to addressing the impact of MELLLA+ implementation on the EOIs and AOIs.

4.0 RENEWED FACILITY OPERATING LICENSE AND TECHNICAL SPECIFICATION CHANGES

The licensee in Section 1 of the enclosure to its LAR (Reference 1) proposed changes to the following RFOL and TSs:

- Prohibiting operation in the MELLLA+ domain when operating in the following plant configurations:
 - reactor recirculation system SLO
 - more than a 10 °F reduction in FW temperature below the design FW temperature
- Replacing TS 3.3.1.1 conditions/required actions/completion times for Conditions I and J with new conditions/required actions/completion times for Conditions I, J, and K, due to implementation of DSS-CD stability solution.
- Eliminating SR 3.3.1.1.17, which is no longer required by the proposed DSS-CD stability solution.
- Revising TS Table 3.3.1.1-1 Function 2.b to change the allowable value for APRM flow-biased simulated thermal power – high trip function and to add a new note to implement automated backup stability region setpoints when Function 2.f is inoperable.
- Revising TS Table 3.3.1.1-1 Function 2.f to set the new operability power level for OPRM upscale and to add a new note due to implementation of the DSS-CD stability solution.
- Adding new Condition B and corresponding Required Action B.1 and a completion time to TS 3.4.1 to reflect the fact that SLO is prohibited in the MELLLA+ region and to require immediate action required when the MELLLA+ domain is entered with one recirculation loop in operation. The previous Condition B regarding no recirculation loops in operation is re-designated as Condition C.
- Revising Administrative Controls TS 5.6.5.a to require certain content in the COLR and updating the applicable references in Subsection 5.6.5.b. due to implementation of DSS-CD stability solution.
- Adding new Administrative TS 5.6.7, which specifies the contents of a new report required by new TS 3.3.1.1 Required Action I.3.

The licensee provided marked up and retyped pages of the RFOL and TSs in Attachments 1 and 2 to the LAR. The licensee in its letter dated November 25, 2019 (Reference 12) revised the proposed license condition associated with the feedwater temperature. Further in letter dated December 19, 2019 (Reference 13), the licensee proposed to add a license condition regarding implementation of MELLLA+ to correct identified errors in the Framatome RODEX-2A and RODEX-4 codes.

The detailed evaluation of these proposed changes is discussed in corresponding technical subsections of Section 3.0 and Appendices A, B, and C of this SE. The following list is provided for cross-referencing:

4.1 License Conditions

Feedwater Temperature License Condition

Renewed Facility Operating License No. DPR-33 for Unit 1 is amended by addition of License Condition 2.C(22) as follows:

(22) Maximum Extended Load Line Limit Analysis Plus (MELLLA+) Special Consideration

The licensee shall not operate the facility within the MELLLA+ operating domain with more than a 10°F reduction in feedwater temperature below the design feedwater temperature.

Renewed Facility Operating License No. DPR-52 for Unit 2 is amended by addition of License Condition 2.C(22) as follows:

(22) Maximum Extended Load Line Limit Analysis Plus (MELLLA+) Special Consideration

The licensee shall not operate the facility within the MELLLA+ operating domain with more than a 10°F reduction in feedwater temperature below the design feedwater temperature.

Renewed Facility Operating License No. DPR-68 (Unit 3) is also amended by addition of License Condition 2.C(18) as follows:

(18) Maximum Extended Load Line Limit Analysis Plus (MELLLA+) Special Consideration

The licensee shall not operate the facility within the MELLLA+ operating domain with more than a 10°F reduction in feedwater temperature below the design feedwater temperature.

The NRC staff concludes that the proposed license conditions are acceptable since they meet the intent of M+SER L&C 12.5.b and are supported by the licensee's M+SAR. See Appendix B of this SE for detailed discussion.

MELLLA+ Implementation License Condition

Renewed Facility Operating License No. DPR-33 for Unit 1 is amended by addition of License Condition 2.C(23) as follows:

- (23) Maximum Extended Load Line Limit Analysis Plus (MELLLA+) Implementation

Prior to the first implementation of MELLLA+, TVA shall perform reload safety analyses using codes that have been corrected for the errors described in TVA letter CNL-19-125, dated December 19, 2019.

Renewed Facility Operating License No. DPR-52 for Unit 2 is amended by addition of License Condition 2.C(23) as follows:

- (23) Maximum Extended Load Line Limit Analysis Plus (MELLLA+) Implementation

Prior to the first implementation of MELLLA+, TVA shall perform reload safety analyses using codes that have been corrected for the errors described in TVA letter CNL-19-125, dated December 19, 2019.

Renewed Facility Operating License No. DPR-68 (Unit 3) is also amended by addition of License Condition 2.C(19) as follows:

- (19) Maximum Extended Load Line Limit Analysis Plus (MELLLA+) Implementation

Prior to the first implementation of MELLLA+, TVA shall perform reload safety analyses using codes that have been corrected for the errors described in TVA letter CNL-19-125, dated December 19, 2019.

The NRC staff concludes that the proposed license conditions are acceptable since the identified errors in the Framatome codes will be corrected prior to implementation of the MELLLA+ amendments.

4.2 Technical Specification Changes

The NRC staff reviewed the following TS changes proposed for BFN Units 1, 2, and 3 in the licensee's LAR and concludes that they are acceptable based on the detailed discussions of the corresponding sections of the SE:

- TS 3.3.1.1, Reactor Protection System (RPS) Instrumentation

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

- I.1 Initiate action to implement the Manual Backup Stability Protection (BSP) Regions defined in the COLR. (Completion Time: Immediately)

AND

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

AND

- I.3 Initiate action to submit an OPRM report 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 18), and finds these changes to be consistent with Section 8 of the approved LTR. The proposed required actions are also the same actions provided in the Sample BWR-4 TSs of Appendix A of the approved LTR. The sample TSs within LTR Appendix A are applicable to BFN Units 1, 2, and 3 because these plants are General Electric Type 4 BWR plants. These required actions are, therefore, acceptable.

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

Required Action J.1 is changed 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 TSs of Appendix A to the approved DSS-CD (Reference 18) and are consistent with Section 8 of the approved LTR. These required actions are, therefore, acceptable.

- TS 3.3.1.1, Required Action K.1

A new Required Action K.1 is added 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 percent 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 TSs of Appendix A to the approved DSS-CD. Reducing power level to less than 18 percent RTP will place the plant into a condition to which LCO 3.3.1.1 does not apply for OPRM upscale functions because Function 2.f of Table 3.3.1-1 is only required at power levels greater than or equal to 18 percent RTP. This is consistent with Section 8 of the approved DSS-CD LTR. These required actions are, therefore, acceptable.

- TS 3.3.1.1, SR 3.3.1.1.17

This SR is deleted. 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 18). This change is consistent with the Sample BWR-4 TSs of Appendix A and with Section 8 of the approved LTR. 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 is acceptable.

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

The allowable value for Function 2.b in Table 3.3.1.1-1, "Flow Biased Simulated Thermal Power – High," is changed from $\leq .55W + 65.5\%$ Reactor Thermal Power (RTP) to $\leq .61W + 68.3\%$ RTP. In addition, a note (e) is added to address the OPRM upscale function inoperable condition.

The revised allowable value formula reflects the changed curve for determining the Flow Biased Simulated Thermal Power trip setpoint based on power level and recirculation drive flow. These revised setpoints were calculated in accordance with the Section 5.3.1 of BFN M+SAR (Attachment 6 to the LAR (Reference 1)) and are consistent with Section 5.3.1 of the approved M+LTR (Reference 17).

The OPRM functions are described in the TS Bases B.3.3.1.1 Function 2.f, "Oscillation Power Range Monitor (OPRM) Upscale," contained within Attachment 4 to the LAR (Reference 1).

The changes to the OPRM upscale function settings and the addition of note (e) are consistent with the Sample BWR-4 TSs and TS Bases of Appendix A and with Section 8 of the approved DSS-CD LTR (Reference 18). The NRC staff finds that these changes are, therefore, acceptable.

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

The specified condition associated with Function 2.f of Table 3.3.1.1-1 is changed from Mode 1 to $\geq 18\%$ RTP and a new note (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 Requirement (SR) 3.3.1.17, which required a periodic verification that the OPRM is not bypassed when the APRM simulated thermal power is $\geq 23\%$ and recirculation flow is $\leq 60\%$, is deleted. This surveillance is no longer required because DSS-CD functions automatically arm when predetermined conditions are met.

Note (e) was deleted (refer to COLR for OPRM period-based detection algorithm (PBDA) setpoint limits) from the allowable value column due to implementation of DSS-CD.

The change to the specified condition for Function 2.f is consistent with the requirement that the DSS-CD must be operable above a power level 5 percent below the lower RTP boundary of the DSS-CD armed region. Since the lower boundary of the DSS-CD armed region is 23 percent, as defined by the MCPR threshold power level in TS 3.2.2, the NRC staff finds this revised condition of $\geq 18\%$ RTP is acceptable. The NRC staff also finds the addition of note (f) is consistent with the Sample BWR-4 TSs of Appendix A and with Section 8 of the approved DSS-CD LTR (Reference 18) and is, therefore, acceptable.

- TS 3.4.1, Recirculation Loops Operating

This TS LCO is revised to stipulate that single recirculating loop operation is not allowed when the reactor is in the MELLLA+ operating domain. A new note is added to LCO 3.4.1 to prohibit operation in the MELLLA+ region with a single recirculation loop in operation. 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 LCO reinforces the requirement that two recirculation loops with matched flows shall be in operation when the reactor is operating in the MELLLA+ region. This revised LCO is acceptable because the licensee is not requesting to operate the plant in the MELLLA+ domain during SLOs. The new note is:

-----NOTE-----
Single recirculation loop operation is prohibited in the MELLLA+ operating domain.

A new Condition B with Action B.1 is being added to LCO 3.4-1 and a completion time to TS 3.4.1 to reflect the fact that SLO is prohibited in the MELLLA+ region and to require immediate exit from the MELLLA+ domain if one SLO is in operation. The new condition is:

- B. Operation in the MELLLA+ operating domain with a single recirculation loop in operation.

The new required action is:

- B.1 Initiate action to exit the MELLLA+ operating domain. (Completion Time: Immediately)

The existing Condition B is renumbered to Condition C, and the reference to "Condition A" is changed to "Condition A or B."

These changes are consistent with the premise that single recirculating loop operation is not allowed when the reactor is in the MELLLA+ operating domain. These changes are consistent with the M+LTR (Reference 17), and therefore, are acceptable. This revised action condition is consistent with the premise 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 because the licensee is not requesting to operate the plant in the MELLLA+ domain during SLOs.

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

Item a.(4) of this TS is replaced to reflect new COLR setpoint requirements associated with the DSS-CD reactor trip function. The NRC staff finds that this change is consistent with the Sample BWR-4 TSs of Appendix A and with Section 8 of the approved DSS-CD LTR (Reference 18) and is, therefore, acceptable.

Subsection b of this TS is revised to reflect the change in the approved analytical method associated with the DSS-CD methodologies. The PBDA will no longer be credited in the safety analysis. A new reference to the DSS-CD LTR is included with this change. The NRC staff confirmed that the correct reference to the approved methodology was provided, and therefore, finds that this change is acceptable.

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

A new TS is added to stipulate when a report required by Condition I, Required Action I.3, of LCO 3.3.1.1, RPS Instrumentation, shall be submitted and what the contents of this report

shall be. The NRC staff finds that this new TS is consistent with the Sample BWR-4 TSs of Appendix A and with Section 8 of the approved DSS-CD LTR (Reference 18) and is, therefore, acceptable.

5.0 TECHNICAL EVALUATION CONCLUSION

The NRC staff reviewed the licensee's analyses related to the effect of the proposed amendments for BFN Units 1, 2, and 3, to operate in the MELLLA+ domain. The NRC staff concludes from this review that the broadening of the BFN operating domain by lowering the flow at high power levels without additional limitations would reduce the safety margin. However, the licensee has proposed the following solutions in the M+SAR that are technically acceptable to satisfy the regulatory criteria while operating in the MELLLA+ domain:

- Operating with more than a 10 °F reduction in FW temperature below the design FW temperature will not be allowed in the MELLLA+ domain.
- SLO is not allowed in the MELLLA+ domain.

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 support operation at CLTP, which also acts 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 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 EOI operator actions. Therefore, the applicable ATWS regulatory criteria (i.e., demonstrating core coolability is maintained) are satisfied during ATWS-I events for BFN. The NRC staff review considered plant-specific information (e.g., EOPs) and specific aspects of the TRACG codes use as they were applied in the context of the BFN 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 staff's conclusions that the proposed LAR is acceptable is based upon its determinations that:

- All L&Cs from the approved methodology topical reports have been addressed.
- The generic assessments are applicable to BFN.

- The plant-specific assessments meet the regulatory criteria and, where calculations were necessary, the appropriate input assumptions and methods were used.
- TS and license condition changes are appropriate and necessary to ensure safe operations in the expanded CF region.

Based on the considerations noted above and the discussion contained in this SE, the NRC staff concludes that the proposed amendments for BFN Units 1, 2, and 3 are acceptable.

6.0 STATE CONSULTATION

In accordance with the Commission's regulations, the appropriate official for the State of Alabama was notified of the NRC's proposed issuance of the amendments on October 9, 2019. 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 (83 FR 26116; June 5, 2018). 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 TVA letter to U.S. NRC, "Proposed Technical Specifications (TS) Change TS-510 – Request for License Amendments – Maximum Extended Load Line Limit Analysis Plus," dated February 23, 2018 (ADAMS Accession No. ML18079B140).
- 2 TVA letter to U.S. NRC, "Proposed Technical Specifications (TS) Change TS-510 – Request for License Amendments – Maximum Extended Load Line Limit Analysis Plus, Supplement 1," dated March 7, 2018 (ADAMS Accession No. ML18067A495).
- 3 TVA letter to U.S. NRC, "Proposed Technical Specifications (TS) Change TS-510 – Request for License Amendments – Maximum Extended Load Line Limit Analysis Plus - Supplement 2, Operator Training Results," dated July 23, 2018 (ADAMS Accession No. ML18205A498).
- 4 TVA letter to U.S. NRC, "Proposed Technical Specifications (TS) Change TS-510 - Request for License Amendments - Maximum Extended Load Line Limit Analysis Plus - Supplement 4, Response to Request for Additional Information," dated December 13, 2018 (ADAMS Accession No. ML18347B381).
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Principal Contributors:

Office of Nuclear Reaction Regulation		
J. Borromeo (Lead)	M. Biro	C. Basravaju
R. Mathew	J. Dozier	F. Saba
J. Hughey	R. Alvarado	M. Panicker
A. Sallman	R. Stattel	A. Smith
M. Smith	A. Wysocki (ORNL)	N. Karipineni
M. Hamm	D. Woodyatt	E. Dickson
C. Sydnor	J. Tsao	
Office of Nuclear Regulatory Research		
A. Bielen	K. Gibson	C. Gingrich
N. Hudson	J. Staudenmeier	P. Yarsky

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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, "General Electric Boiling Water Reactor Detect and Suppress Solution – Confirmation Density"
- D. Fuel Parameter Sensitivities
- E. AREVA Codes Used for Browns Ferry MELLLA+ Application and Evaluation for MELLLA+ Applicability
- F. 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 disposition of the limitations given in the final safety evaluation for NEDC-33173P-A (Reference 19).

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 BFN equilibrium core for all calculations involving GEH methods, including the MELLLA+ stability and ATWS evaluations.

The comparable AREVA methods used in the BFN MELLLA+ LAR are CASMO4/MICROBURN-B2. In Appendix E of this SE, the NRC staff determined that the use of CASMO4/MICROBURN-B2 in the BFN MELLLA+ domain is acceptable.

Therefore, the NRC staff determined the licensee meets the limitation and condition.

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 while this limitation is specific to GEH/Global Nuclear Fuel (GNF) methods, the intent of the limitation and condition is met. The licensee uses the POWERPLEX core monitoring system based on the NRC-approved CASMO-4/MICROBURN-B2 methodology. In Appendix E of this SE, the NRC staff determined that the use of CASMO4/MICROBURN-B2 in the BFN MELLLA+ domain is acceptable. Furthermore, the uncertainties associated with POWERPLEX CMS are used in the statistical analyses that are performed for SLMCPR and LHGR, which meet the intent of this limitation. Therefore, the NRC staff determined that the licensee meets the limitation and condition.

Limitation 9.3, Power-to-Flow Ratio

Plant-specific EPU and expanded operating domain applications will confirm that the core thermal power to CF 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 O in Figure 1-1 of the AMSAR) exceeds 50 MWt/Mlbm/hr (with a value of 54.4 MWt/Mlbm/hr). 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 the AREVA methodology at BFN, including the applicability of power distribution uncertainties, is addressed in the application. The NRC staff reviewed the CASMO-4/MICROBURN-B2 methodology, which is used to determine the power distribution uncertainties, in Appendix E of this SE and determined it was acceptable to use for BFN in the MELLLA+ operating domain. The NRC staff also reviewed the CASMO-4/MICROBURN-B2 power distribution uncertainties in Section 3.2.2 of this SE and determined that the uncertainties were acceptable for the BFN MELLLA+ operating domain. Therefore, the NRC staff determined the licensee meets this limitation and condition.

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 NRC staff determined that this limitation is not applicable to BFN since the NRC did not impose an adder on 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 NRC staff reviewed the CASMO-4/MICROBURN-B2 power distribution uncertainties and the necessity for a 0.03 penalty applied to the SLMCPR in Section 3.2.2 of this SE. In that section of the SE, the NRC staff determined that no penalty was necessary and that the intent of the limitation and condition is met.

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 they complied with this limitation. The corresponding factors to the R-factors in AREVA methods are K-factors. These K-factors are determined with their existing ACE/ATRIUM 10XM methodology documented in ANP-10298PA (Reference 98). The NRC staff reviewed this methodology and has determined that the K-factors are calculated based on the actual calculated condition within each axial lattice of each fuel bundle, including the void conditions and their impact on rod peaking. Therefore, the NRC staff determined the licensee meets this limitation.

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 they complied with this limitation. The licensee's 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 determined that the licensee meets this limitation.

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 they complied with this limitation. The NRC staff confirmed that calculations for the maximum and minimum CF at rated EPU power as discussed in Section 3.4.3 of this SE.

[[

]] Therefore, the staff determined the

licensee meets this limitation.

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) O₂ rods.

The licensee stated they complied with this limitation. The NRC staff determined that compliance with the T-M acceptance criteria for AOOs has been demonstrated and documented using the most recent NRC-approved method in BAW-10247PA (Reference 26), including the use of RODEX4. The NRC staff reviewed those methods and determined the licensee meets this limitation.

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 they complied with this limitation. The licensee stated they will perform the T-M calculations each cycle. Additionally, the licensee provided the demonstrated acceptable transient T-M response in the demonstration reload analysis provided in the LAR (Reference 1). Therefore, the NRC staff determined the licensee meets this limitation.

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, in Appendix E of this SE, the AREVA methodology and determined this limitation is not applicable to BFN MELLLA+.

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 99) is used for all GEH analyses. The NRC staff determined the licensee meets this limitation from review of the analyses performed with GEH methods.

The licensee stated that the limitation is not applicable for AREVA methods. The licensee is using RODEX4 (Reference 26) for the TM evaluation. The NRC staff reviewed the RODEX4 methodology for applicability in BFN MELLLA+ in Appendix E of this SE and determined that the method is acceptable. Therefore, the NRC staff determined the limitation is not applicable.

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 for AREVA methods. The licensee is using RODEX4 (Reference 26) for the TM evaluation. The NRC staff reviewed the RODEX4 methodology for applicability in BFN MELLLA+ in Appendix E of this SE and determined that the method is acceptable. Therefore, the NRC staff determined the 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 TCD on AREVA methods is described in the LOCA analysis in the application. The NRC staff reviewed the LOCA analysis in Section 3.4.3 of this SE and determined TCD was sufficiently accounted for in BFN analysis. Therefore, the determined this limitation is not applicable to BFN MELLLA+.

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 void reactivity coefficient information for AREVA methods is provided in Attachment 28 to the LAR (Reference 1). The NRC staff reviewed the information and determined that the void reactivity coefficient bias and uncertainties used in AREVA methods are representative of the ATRIUM 10XM fuel loaded into the BFN core for MELLLA+ operation. Therefore, the NRC staff determined the intent of the limitation and condition is met.

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 Attachment 28 to the LAR (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 the information and determined 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 determined the intent of the limitation and condition is met.

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 the limitation is applicable and that the required information regarding the cycle-specific bypass voiding calculations is included in the cycle-specific RSAR for BFN MELLLA+ operation. Section 2.1.2 and 5.1.5 of the AMSAR (Attachment 8 to the LAR (Reference 1)) and Attachment 28 to the LAR documented bypass voiding calculation for the MELLLA+ operating domain. Attachment 28 to the LAR provides a cycle-specific demonstration that the bypass voiding limitation is met. The NRC staff reviewed, in Appendix E of this SE, the information and determined the methodology is sufficient to predict the bypass voiding and the demonstration analysis meets this limitation. Therefore, the NRC staff determined this limitation is met.

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 BFN since there is significant conservatism in the DSS-CD methodology due to the 20 percent and 50 percent MCPR penalties on the flow reduction and oscillation in the methodology. Additionally, under MELLLA+ conditions the licensee predicted a negligible amount of bypass voiding at the D level LPRM (see Table 2-2 of the AMSAR). The NRC staff determined that since there is negligible bypass voiding for BFN MELLLA+ conditions, the conservatism in the DSS-CD methodology is sufficient to account for the calibration errors. Therefore, the NRC staff determined that the licensee sufficiently addressed this limitation.

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 they complied with this limitation. The void-quality correlation used in AREVA methods for BFN MELLLA+ (the $\left[\frac{1}{1 + 0.01 \left(\frac{1}{\text{OLMCPR}} \right)} \right]$ void correlation) is addressed in Section 9.0 of the AMSAR and Attachment 28 to the LAR. The NRC staff reviewed the void-quality correlation in Section 3.2.2 and determined that the correlation is sufficient for BFN MELLLA+ operating domain. The NRC staff determined that no additional OLMCPR penalty was necessary since there was sufficient experimental data to validate the void fraction correlations for AREVA ATRIUM 10XM fuel, including void fraction levels close to 100 percent. Therefore, the NRC staff determined this limitation is met.

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 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 100), as required by this limitation. Therefore, the NRC determined the licensee met this limitation.

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 BFN is not a mixed vendor core and contains only AREVA 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 BFN MELLLA+.

For BFN MELLLA+, the licensee provided details on the applicability of GE methods to ATRIUM 10XM fuel in Attachment 35 to the LAR (Reference 1). The licensee demonstrated 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 determine that ATRIUM 10XM fuel has been modeled in a satisfactory manner using GEH methods for BFN MELLLA+. Therefore, the NRC staff determined the licensee met the intent of this limitation.

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, and 11x11 lattices.

The acceptability of the modified epithermal slowing down models in TGBLA06 has not been demonstrated for application to these or other geometries for expanded operating domains. Significant changes in the Gd rod optical thickness will require an evaluation of the TGBLA06 radial flux and Gd depletion modeling before being applied. Increases in the lattice Gd loading that result in nodal reactivity biases beyond those previously established will require review before the GE methods may be applied.

The licensee stated that the limitation is applicable because BFN contains a full core of AREVA fuel and is analyzed with AREVA methodology. Based on information provided in Attachment 34 to the LAR (and as discussed in the review of the previous limitation), the NRC staff determined there is sufficient information to demonstrate that TGBLA06 is acceptable for the modeling of ATRIUM 10XM fuel in BFN MELLLA+. Therefore, the NRC staff determined the licensee sufficiently addressed this limitation.

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 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 stated they comply with this limitation. The licensee will provide the information required by the limitation after MELLLA+ implementation at BFN. Therefore, the NRC staff determined this limitation is met.

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 it complies with this limitation. The NRC staff reviewed the information provided in the Section 2.1.2 of the AMSAR and determined that all information has been provided as required in this limitation. Thus, the NRC staff determined this limitation is met.

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 the final safety evaluation for NEDC-33006P (Reference 17).

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 for GEH methods. The licensee stated the GEXL database covers the ranges of mass fluxes and power/flow ratios for BFN MELLLA+. The NRC staff reviewed the ranges and determined that the licensee meets this limitation.

The licensee complied with this limitation for AREVA methods. 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 provided in Attachment 28 to the LAR (Reference 1). The NRC staff determined that the ranges and conservative actions are applicable for the BFN MELLLA+ operating domain. Therefore, the NRC staff determined the limitation 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 sufficiently. Therefore, the NRC staff determined the limitation is met.

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 NRC staff determined the LAR analyses comply with all operating condition changes that were implemented at BFN in support of EPU and MELLLA+. Therefore, the NRC staff determined the limitation was met.

Limitation 12.3b

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

The licensee stated that this limitation is applicable. The NRC staff determined all events that were included in the BFN MELLLA+ application have been analyzed or dispositioned adequately for MELLLA+. Therefore, the NRC staff determined this limitation was 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. The NRC staff determined plant-specific calculations (including ATWS) have been performed using BFN MELLLA+ conditions. Therefore, the NRC staff determined this limitation was 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 BFN MELLLA+ conditions and AREVA ATRIUM 10XM fuel. The NRC staff has reviewed the calculations provided in the M+SAR and AMSAR and has determined that the design features and performance of ATRIUM 10XM fuel have been adequately accounted for in both GEH and AREVA methods. Therefore, the NRC staff determined this limitation was 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 dispositioned or reduced in scope in LTR NEDC-33006P will be demonstrated to be applicable, or new analyses based on the specific core configuration or bounding core conditions will be provided.

The licensee stated that this limitation is applicable. The NRC staff determined plant-specific calculations have been performed using BFN MELLLA+ conditions and AREVA ATRIUM 10XM fuel. Therefore, the NRC staff determined this limitation was 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, NEDO-33075-A (Reference 18). The NRC staff determined plant-specific calculations have been performed using BFN MELLLA+ conditions and AREVA ATRIUM 10XM fuel. Therefore, the NRC staff determined this limitation was 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. The NRC staff determined BFN MELLLA+ adopted the approved DSS-CD stability solution, including an automated backup stability solution. Therefore, the NRC staff determined this limitation was 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 M+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 BFN-specific RSAR (which is the AREVA equivalent to the SRLR) was submitted in Attachment 18 to the LAR (Reference 1). Therefore, the NRC staff determined this limitation was 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. The NRC staff determined the BFN TSs were updated to support the BFN MELLLA+ operation including the associated equipment out-of-service limitations. Therefore, the NRC staff determined this limitation was 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. The licensee proposed a license condition to limit FW temperature reduction to 10 °F below the design FW temperature in the MELLLA+ operating domain. Additionally, the NRC staff determined the licensee performed the necessary evaluations to determine that this license condition is acceptable for BFN MELLLA+. Therefore, the NRC staff determined this limitation was 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 and will provide the power/flow map in the COLR. Therefore, the NRC staff determined this limitation was met.

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 BFN statepoint corners and have been calculated using the off-rated CF uncertainty where appropriate (Section 3.2.2 of this SE). Therefore, the NRC staff determined this limitation was 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. Therefore, the NRC staff determined this limitation was 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 evaluation of fracture toughness is included in Section 3.3.2.1 of this SE.

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 evaluation of other than Category "A" materials in the Reactor Coolant Pressure Boundary is included in Section 3.3.5.2 of this SE.

Limitation 12.10a, LOCA-Off-rated Multiplier

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

The licensee stated that the limitation is applicable. The NRC staff confirmed that calculations for the maximum and minimum CF at rated EPU power have been performed and that the limiting (licensing basis) PCT has been provided, consistent with AREVA methods. [[

]] Therefore, the NRC staff determined this limitation was met.

Limitation 12.10b

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

The licensee stated that it complied with this limitation. [[

]] the NRC staff determined no cycle-specific validation is required. Therefore, the NRC staff determined this limitation was met.

Limitation 12.10c

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

The licensee stated that this limitation is applicable. The NRC staff confirmed that [[
]] Therefore, the staff determined this limitation was met.

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 they complied with this limitation. [[
]], the staff determined [[
]] Therefore, the NRC staff determined this limitation was 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 Attachment 12 to the LAR (Reference 1) as required by this limitation. Therefore, the NRC staff determined this limitation was 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 AMSAR using the approved AREVA LOCA methodology. Therefore, the NRC staff determined this limitation was 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. The NRC staff determined the important plant parameters found in Attachment 12 to the LAR and used in the LOCA analyses are acceptable for BFN MELLLA+. Therefore, the NRC staff determined this limitation was 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.

The licensee stated that this limitation is applicable. The NRC staff determined small break analyses are performed for the **[[]]** and are reported in Attachment 12 to the LAR. The NRC staff determined the small breaks analyzed are sufficient to cover the sensitivities required by this limitation. Therefore, the NRC staff determined this limitation was 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. The NRC staff determined a large number of break sizes were evaluated at different flow rates and are reported in Attachment 11 to the LAR. Therefore, the NRC staff determined this limitation was 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 5 percent will occur above the D-level. The licensee shall provide in the plant-specific submittal the operator actions and procedures that will mitigate the impact of the bypass voiding on the TIPs and the core simulator used to monitor the fuel performance. The plant-specific submittal shall also provide discussion on what impact the bypass voiding greater than 5 percent will have on the NMS [neutron monitoring system] as defined in Section 5.1.1.5. The NRC staff will evaluate on plant-specific bases acceptability of bypass voiding above D level.

The licensee stated that this limitation is applicable. The NRC staff determined bypass boiling was evaluated for BFN MELLLA+ operation and found to be negligible at the D level LPRM, which is below the 5 percent acceptance criteria. Therefore, the NRC staff determined this limitation was 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 M+SAR shall provide a discussion of the analyses performed and the results.

The licensee stated they complied with this limitation. 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 BFN RBM setpoints. Therefore, the NRC staff determined this limitation was 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 NRC staff determined the RHR capability is impacted by MELLLA+; therefore, the licensee is required to perform LOOP calculations. The licensee provided these calculations in Section 9.3.1 of the M+SAR. Therefore, the NRC staff determined this limitation was 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 since the calculated suppression pool temperature using ODYN remains below the HCTL curves. The NRC staff confirmed that the ODYN analysis in Section 9.3.1 of the M+SAR results in suppression pool temperatures below HCTL. Therefore, the NRC staff determined this limitation was met.

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 not applicable since the calculated suppression pool temperature using ODYN remains below the HCTL curves. The NRC staff confirmed that the ODYN analysis in Section 9.3.1 of the M+SAR results in suppression pool temperatures below HCTL. Therefore, the NRC staff determined this limitation was 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.

In the AMSAR, the licensee stated that the requirement to calculate PCT for the initial overpressure phase is applicable. The NRC staff reviewed Section 9.3 of the AMSAR and determined the licensee appropriately determined the PCT. Therefore, the NRC staff determined this limitation was met.

In the M+SAR, the licensee stated that the requirement to calculate PCT for the emergency depressurization phase is not applicable because TRACG calculations are not required. The NRC determined this part of the limitation is not applicable because no TRACG calculations were required for emergency depressurization.

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 OLYN ATWS calculations in the M+SAR and the COTRANSA2 calculations in the AMSAR and determined that the input parameters, calculation assumptions, and equipment out-of-service conditions are reflective of the allowed BFN plant configuration in MELLLA+ and that the important parameters are included in the analyses. Therefore, the NRC staff determined this limitation was 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 in the M+SAR and the CONTRANSA2 ATWS analyses in the AMSAR and determined that the input parameters are conservative for ATWS. Therefore, the NRC staff determined this limitation was 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 9-6 in the AMSAR. Therefore, the NRC staff determined this limitation was 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 TGBLA06/PANAC11 and TRACG04.

The licensee stated that this condition is applicable. The licensee 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 determined this limitation was 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 determined this is acceptable because no generic ATWS-I solution has been approved at this time. Plant-specific ATWS-I analyses were performed for BFN 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 evaluation of Individual Plant Examination is included in Section 3.10.5 of this SE.

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 5×10^{20} n/cm² ($E > 1$ MeV), 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 evaluation of this limitation is included in Section 3.10.7.1 of this SE.

12.23 Limitations from the ATWS RAI Evaluations

Limitation 12.23.1

See limitation 12.18.d.

Note that this limitation is the same as 12.18.d. The licensee stated that this condition is applicable. The NRC staff has reviewed the licensee's ODYN ATWS calculations in the M+SAR and the COTRANSA2 calculations in the AMSAR and determined that the input parameters, calculation assumptions, and equipment out-of-service conditions are reflective of the allowed BFN plant configuration in MELLLA+ and that the important parameters are included in the analyses. Therefore, the NRC staff determined this limitation was met.

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 NRC staff determined the licensee provided the key calculation parameters in Section 1.1.3, Section 9.3.1, and Table 9-1 of the M+SAR, and Section 9.3 and Table 9-6 of the AMSAR, which satisfies this condition. Therefore, the NRC staff determined this limitation was 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 NRC staff determined the licensee used a 95/95 approach, incorporating the SRV setpoint uncertainty, to demonstrate acceptable result to the ATWS acceptance criteria in Section 9.3 of the AMSAR. The NRC staff also determined the plant-specific SRV performance complies with its licensing basis values. Therefore, the NRC staff determined this limitation was 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 emergency procedure guidelines/SAG parameters and confirmed that they remain applicable for BFN MELLLA+. The NRC staff also reviewed the ATWS analyses and determined that they account for all relevant EOLs, including water level control, SLC system injection, and RHR suppression pool cooling. Therefore, the NRC staff determined this limitation was 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 BFN, the power/flow ratio at the minimum allowable CF rate in the MELLLA+ domain at 120 percent OLTP is 45.4 MWt/MLbm/hr, per Table 1-3 of the M+SAR. Therefore, the NRC staff determined this limitation was 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 NRC staff determined the licensee performed bounding plant-specific ATWS-I analyses fully accounting for the design features and performance of ATRIUM 10XM fuel under MELLLA+ operation conditions for BFN. Therefore, the NRC staff determined this limitation was 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 condition is applicable. The NRC staff determined the licensee performed bounding plant-specific ATWS-I analyses fully accounting for the design features and performance of ATRIUM 10XM fuel under MELLLA+ operation conditions for BFN. Therefore, the NRC staff determined this limitation was met.

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 determined this limitation was 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 evaluation of this limitation is provided in Section 3.4.2.6.1 and Section 3.9.3.1 of this SE.

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 evaluation of this limitation is provided in Section 3.4.2.6.1 and 3.9.3.1 of this SE.

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 NRC staff determined the suppression pool temperature limit is unchanged relative to pre-MELLLA+ operation and that the HCTL for emergency depressurization also remained below the suppression pool temperature limit. Therefore, the NRC staff determined this limitation was 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. The NRC staff determined in-channel water rod flow was explicitly modeled in TRACG04 for the ATWS and ATWS-I analyses. The licensee also stated that the AREVA methods that have the capability of modeling in-channel water flow would include that modeling capability in the analysis. Therefore, the NRC staff determined this limitation was 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. The NRC staff determined exit void fraction conditions were provided in Section 2.1.2 of the AMSAR. Therefore, the NRC staff determined this limitation was 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. SLMCPR values have been provided at each of the specified BFN statepoint corners and have been calculated using the off-rated CF uncertainty where appropriate (Section 3.2.2 of this SE). Therefore, the NRC staff determined this limitation was met.

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 condition is applicable. The NRC staff has reviewed the licensee's OLYN ATWS calculations in the M+SAR and the COTRANSA2 calculations in the AMSAR and determined that the input parameters, calculation assumptions, and equipment out-of-service conditions are reflective of the allowed BFN plant configuration in MELLLA+ and that the important parameters are included in the analyses. Therefore, the NRC staff determined this limitation was met.

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 disposition of the limitations in the final safety evaluation for NEDC-33075P (Reference 18)

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. BFN currently uses GEH Option III hardware. Therefore, the NRC staff finds that this limitation is applicable to BFN MELLLA+ and has been addressed adequately.

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 TSs, as discussed in Section 3.2.4 in this SE. Therefore, the NRC staff finds that this limitation is applicable to BFN MELLLA+ and has been addressed adequately as discussed in Section 3.2.4.

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 **[[]]** parameter. For detailed discussion and the NRC staff's evaluation see Section 3.2.4 of this SE.

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 BFN since the BFN stability solution licensing basis is unchanged from current. Therefore, the NRC staff finds that this limitation has been addressed adequately for BFN MELLLA+.

Limitations from NEDC-33147P, "DSS-CD TRACG Application"

As discussed under the evaluation for NEDC-33006P L&C 12.2 (Appendix B of this SE), the limitations of NEDC-33147P, Revision 2, which require additional justification to use un-approved codes, are no longer applicable because TRACG is now approved for DSS-CD stability solution calculations in NEDC-33075PA, Revision 8. Therefore, the NRC staff finds that this limitation has been addressed adequately.

APPENDIX D

Fuel Parameter Sensitivities

Overview

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 requirement review to down-select a set of fuel parameters that were the most 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):

- Direct energy deposition [[]]
- Gap conductance [[]]
- Thermal and hydraulic channel losses [[]]

For the TRACG ATWS-I analyses, these parameters were:

- Direct energy deposition [[]]
- Gap conductance [[]]
- Thermal and hydraulic channel losses [[]]
- Void history effects [[]]
- GEXL critical quality [[]]

The justification for each of these parameter ranges is provided in the following sections.

Direct Energy Deposition

The direct energy deposition sensitivity range was based on CASMO calculations for both ATRIUM-10 and ATRIUM 10XM fuel. This study found a [[]] energy deposition in ATRIUM 10XM fuel (occurring in the bottom lattice), and a [[]] energy deposition (occurring in the top lattice). AREVA attributed this [[]]

]] The [[]] range used for the ATWS sensitivity studies bounds the calculated range of energy deposition, with significant conservatism in the upper bound relative to the calculated results.

Gap Conductance

The gap conductance sensitivity range was determined based on a comparison of RODEX2-calculated gap conductance for ATRIUM 10XM and ATRIUM-10 assemblies, using

MICROBURN-B2 calculated power history values from [[]] cycle exposure points spanning a representative BFN ATRIUM 10XM equilibrium cycle. For each cycle exposure point, power histories were taken from fuel bundles spanning a wide range of bundle power and exposure conditions.

These power histories were used as inputs to the RODEX2 code, which was used to calculate gap conductance for ATRIUM 10XM and ATRIUM-10 fuel at LHGR values ranging from [[]]. The maximum ratio of ATRIUM 10XM to ATRIUM-10 gap conductance among all test points was [[]], while the minimum was [[]]. This is the basis for the gap conductance sensitivity range of [[]] used in the ATWS and ATWS-I analyses.

In general, ATRIUM 10XM has [[]]

]]

The NRC staff notes that this approach for determining the gap conductance sensitivity range includes substantial conservatism, in that it represents a reasonable estimate of the largest possible difference in gap conductance across the full range of bundle power, power history, and exposure expected to occur at any point in the cycle in any assembly, and then applies these bounding sensitivity values to all fuel nodes in the core simultaneously during the ATWS and ATWS-I sensitivity calculations. In reality, the difference in gap conductance would be substantially closer to 0% in the majority of fuel nodes, particularly for the ATWS-I calculations that were performed at BOC.

Thermal and Hydraulic Channel Losses

The ATWS and ATWS-I fuel parameter sensitivities included a +/- 10% sensitivity range for TH loss factors for ATRIUM 10XM fuel. This range was selected based on engineering judgment (i.e., it did not come directly from calculations using ISCOR, XCOBRA, or other codes) and was expected to bound possible variations in TH losses and intra-assembly flow rates when comparing ATRIUM 10XM to ATRIUM-10. This expectation was based, in part, on the close agreement between GEH and AREVA TH results, which indicated that a +/- 10% sensitivity range was conservatively large. After GEH selected this sensitivity range of +/- 10%, AREVA confirmed that this range was acceptable and reasonably conservative for use in the ATWS and ATWS-I sensitivity calculations.

Void History Effects

The [[]] perturbation in void reactivity coefficient was determined based on TGBLA/MCNP comparisons for GE14 and GNF2 lattices at various burnup, void, and temperature conditions. The void coefficient model used in TRACG is adjusted based on the average differences between TGBLA and MCNP predictions, and the uncertainty in this quantity (about 2% for both fuels combined, or about 3% considering GNF2 only) was increased by a multiplier of 1.5 to conservatively account for the possibility that inclusion of ATRIUM 10XM in the comparisons might increase the uncertainty. This gave a conservative upper limit of about 4.5% uncertainty, which GEH further increased to the range of 5% that was used for the void history sensitivity (i.e. [[]]).

The NRC staff notes that the void history sensitivity had a small effect on the figure of merit (PCT) for the ATWS-I analyses, relative to the other parameters that were varied in the sensitivity study. The [] perturbation in void reactivity coefficient led to a variation of [] in the PCT value for the reference 2RPT case. The NRC staff suspects that the change in PCT due to the void history sensitivity may be higher for the 2RPT cases experiencing failure-to-rewet using the homogeneous nucleation T_{min} model, due to the larger overall PCT excursion in these cases. However, the NRC staff expects that the effect of the void history sensitivity would remain relatively small compared to the other sensitivity parameters.

GEXL Critical Quality

The parameter range for GEXL critical quality was chosen to be []

AREVA used its own MICROBURN-B2 single-channel TH models to reproduce the operating conditions of the KATHY ATRIUM 10XM CPR data. Starting with an implementation of the GEXL97 correlation within MICROBURN-B2 (developed for ATRIUM-10 fuel), AREVA then calculated appropriate ATRIUM 10XM GEXL correlation coefficients for this application using a data fitting process recommended by GEH. This process involved leaving certain "complex" terms (such as those involving the boiling length and R-factors) unchanged relative to ATRIUM-10 fuel and adjusting certain "simple" terms to fit the ATRIUM 10XM KATHY data.

AREVA performed the data fit such that the correlation was as close to "best estimate" as possible (about [] conservative) over a reduced input range corresponding to low-flow, low-subcooling, bottom-peaked conditions characteristic of ATWS-I oscillations. Outside of this range, the data were fit in a more conservative manner (about [] conservative).

The GEXL results for several steady-state MICROBURN-B2 cases representative of low-flow ATWS-I conditions were then compared to the approved ACE correlation results for ATRIUM 10XM fuel, and it was required that the GEXL results were conservative relative to ACE.

The [] range used for the fuel parameter sensitivity study for GEXL was determined by starting with the [] minimum conservatism in the data fitting process (described above) and applying a [] CPR uncertainty value of roughly [] for the KATHY data at low-flow ATWS-I conditions. This suggested a lower bound of [] for the sensitivity calculation, which was adjusted to [] for additional conservatism. Similarly, the upper bound of [] was adjusted to []. The NRC staff finds that the [] range can reasonably be expected to bound the variation in CPR relative to the GEXL correlation predictions, with an apparent degree of conservatism added.

However, although a decrease in the critical power values calculated by GEXL would decrease the margin to dryout, and therefore, would most likely be conservative (in terms of increasing the PCT), the sensitivity results for GEXL shown in Table 9-11 of the BFN M+SAR indicate no (0 °K) impact on PCT for the GEXL sensitivity range of []. The NRC staff concludes that this lack of impact on PCT is likely due to the fact that the GEXL correlation directly impacts the occurrence of dryout but has a less significant effect on the heat transfer behavior within the transition boiling regime, and this effect diminishes as the cladding temperatures exceed T_{CHF} and approach T_{min} . Furthermore, GEXL has no effect on the heat transfer behavior in the film boiling regime. Therefore, since the PCT in the reference 2RPT

case was associated with the transition boiling regime, a reduced sensitivity (or no sensitivity) of PCT to the GEXL correlation is understandable and not unexpected.

Furthermore, the NRC staff concludes that, TRACG likely uses the Biasi correlation for the determination of boiling transition during ATWS-I oscillations. It seems likely that this effectively reduced the impact of the GEXL fuel parameter sensitivity study with respect to PCT, as PCT was presumably more strongly affected by the Biasi correlation during the ATWS-I oscillations.

Considering this, the NRC staff concludes that the use of the Biasi correlation is acceptable because it was developed using tube data, and therefore, it lacks dependence on fuel-geometry-specific details such as are present in the GEXL correlation. Because the Biasi correlation is not implemented in a fuel-specific manner, the use of fuel parameter sensitivities based on ATRIUM 10XM versus ATRIUM-10 would not be relevant for the Biasi correlation. Therefore, the NRC staff finds the application of the GEXL fuel parameter sensitivities to be acceptable, despite the limited use of GEXL during ATWS-I oscillations.

Conclusions for ATWS-I Fuel Parameter Sensitivity Results

In Table 9-11 of the BFN M+SAR, the licensee provided results for the 2RPT 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 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 NRC 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 conductance 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 BFN for MELLLA+.

Consideration of DSS-CD Fuel Parameter Values

For the DSS-CD analyses, only the "nominal" fuel parameter values were used, based on the previous values used by GEH for ATRIUM-10 fuel. However, in BFN M+SAR, the licensee justifies that the "20/50 bounding uncertainty" used in the DSS-CD analyses are sufficiently conservative to account for the expected variation in the calculated stability behavior for the DSS-CD confirmatory analyses. The NRC staff finds this justification acceptable because the "20/50 bounding uncertainty" is intended to encompass variability in fuel performance across different fuel designs and vendors, and ATRIUM-10 and ATRIUM 10XM are expected to behave sufficiently similarly in terms of the DSS-CD stability analyses that the use of the "20/50 bounding uncertainty" remains acceptable.

APPENDIX E

AREVA Codes Used for Browns Ferry MELLLA+ Application and Evaluation for MELLLA+ Applicability

This section provides a summary report on AREVA licensing methods and AREVA topical reports used for EPFOD analysis at BFN. This review of AREVA methods identified that there are no SER restrictions on power or flow for the AREVA topical reports. The review also indicated that there are no SER restrictions on the parameters most impacted by the increased power level at each CF rate in the MELLLA+ domain: steam flow, FW flow, jet pump M-ratio, and core average void fraction.

The determination of the applicability of the AREVA methods for the BFN EPFOD analysis included an evaluation of the core and reactor conditions experienced under EPFOD conditions to determine any challenges to the validity of the models. When the reactor power is increased, and/or the CF 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 EPFOD operating conditions:

- The reduction in the hot assembly radial peaking factor leads to a more uniform radial power distribution and consequently a more uniform CF 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 CF at CFs previously constrained by the MELLLA operating boundary.
- With the increase in the average assembly power for a given CF, the core pressure drop will increase slightly resulting in a decrease in the jet pump M-ratio for a given CF rate.
- Core average void fraction will increase.

The following is a list of approved codes that were used in the BFN EPFOD analysis:

1. CASMO4/MICROBURN-B2 is the approved code, described in EMF-2158(P)(A) (Reference 29), that is used in the steady state core simulator. CASMO4 is used to generate the lattice cross sections as a function of instantaneous void and temperature as well as the histories. MICROBURN-B2 performs 3D neutronic calculations and couples them to the TH solution.
2. SAFLIM3D, addressed in ANP-10307PA, Revision 0 (Reference 28), is the approved code for AREVA safety limit methodology for BWRs.
3. XCOBRA is the steady state detailed TH 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) (Reference 80).

4. XCOBRA-T is the approved code, described in XN-NF-84-105(P)(A) (Reference 75), transient TH analysis code. This code is used to perform analyses of transient heat transfer behavior in BWR assemblies.
5. COTRANSA2 is the approved transient coupled neutronic TH code used for transient analyses, including AOOs and is described in ANF-913(P)(A) (Reference 84). 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 frequency domain stability code, described in EMF-CC-074(P)(A) (Reference 101), used for exclusion region calculations and TH stability analysis.
7. RODEX2 code is approved, described in XN-NF-81-58(P)(A) (Reference 76), for T-M fuel performance. This code 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, described in BAW-10247PA (Reference 26), for T-M fuel performance of BWR fuel designs. RODEX4 is used in the T-M 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, described in (EMF-2361(P)(A)) (Reference 68), that calculates the system and hot channel blowdown transient. This code is part of the EXEM/BWR ECCS evaluation suite of codes.
10. HUXY is the approved code, described in XN-CC-33(A) (Reference 69), 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. This code 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 is used to generate the lattice cross sections as a function of instantaneous void and temperature as well as the histories. MICROBURN-B2 is used to perform 3D neutronic calculations and couple them to the TH solution.

The approving SER has the following limitations, which are implemented by AREVA as engineering guidelines:

1. The CASMO-4/MICROBURN-B2 code systems shall be applied in a manner that predicted results are within the range of the validation criteria (Tables 2.1 and 2.2) and measurement uncertainties (Table 2.3) presented in EMF-2158(P)(A) (Reference 29).
2. The CASMO-4/MICROBURN-B2 code system shall be validated for analyses of any new fuel design, which departs from current orthogonal lattice designs and/or exceed gadolinia and U-235 enrichment limits.

3. The CASMO-4/MICROBURN-B2 code system shall only be used for BWR licensing analyses and BWR core monitoring applications.
4. The review of the CASMO-4/MICROBURN-B2 code system should not be construed as a generic review of the CASMO-4 or MICROBURN-B2 computer codes.
5. The CASMO-4/MICROBURN-B2 code system is approved as a replacement for the CASMO-3G/MICROBURN-B code system used in NRC-approved AREVA BWR licensing methodology and in AREVA BWR core monitoring applications. Such replacements shall be evaluated to ensure that each affected methodology continues to comply with its SER restrictions and/or conditions.
6. AREVA shall notify any customer who proposes to use the CASMO-4/MICROBURN-B2 code system independent of any AREVA fuel contract that conditions 1 through 4 above must be met. AREVA's notification shall provide positive evidence to the NRC that each customer has been informed by AREVA of the applicable conditions for using the code system.

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

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

SAFLIM3D

SAFLIM3D is the safety limit methodology for BWRs used by AREVA. 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 (Reference 28) 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. BFN operation in the EPFOD does not impact the process; therefore, the NRC staff concludes that the use of SAFLIM3D in the BFN EPFOD is an acceptable extension of the existing approval.

XCOBRA

XCOBRA is the steady state detailed TH 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, described in XN-NF-80-19(P)(A) (Reference 80). 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 Exxon Nuclear Company (ENC) transient safety analyses are monitored. Whatever monitoring system is used should be specifically identified in plant submittals.

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

As discussed in Section 3.3.2 in this SE, the NRC staff concludes the EPFOD operating conditions for BFN 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 BFN EPFOD is an acceptable extension of the existing approval.

XCOBRA-T

XCOBRA-T is the transient TH analysis code. This code is used to perform analyses of transient heat transfer behavior in BWR assemblies.

The SER (Reference 75) for XN-NF-84-105(P)(A), "Transient Thermal Hydraulic Analysis Code," 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 approval was subsequently expanded in letter dated May 31, 2000 (Reference 102):
 - 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 FW 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 EPFOD operating conditions in BFN 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 (Reference 103). XCOBRA-T uses the Ohkawa-Lahey correlation, [[

]]

The NRC staff finds that operation in EPFOD does not impact significantly whether a transient will have downflow in the bypass region, which requires a very low CF rate. Thus, the applicability of XCOBRA-T in the EPFOD is not likely to be impacted by reverse bypass flow.

The NRC staff finds that operation in the EPFOD increases the core average void fraction, but BFN EPFOD conditions are bounded by experience of other operating plants that use approved AREVA methods successfully. In addition, AREVA has demonstrated [[

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Therefore, the NRC staff concludes that the use of XCOBRA-T in the BFN MELLLA+ EPFOD is an acceptable extension of the existing approval.

COTRANSA2

COTRANSA2 is the transient coupled neutronic TH's code used for transient analyses, including AOOs. The COTRANSA2 code is used to calculate BWR system behavior for steady-state and transient conditions. This behavior is then used to provide input to the XCOBRA-T and XCOBRA codes, from which critical power ratios are determined for limiting transients.

The approval SER for ANF-913PA (Reference 73) 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.

5. COTRANSA2 is approved for using in the Chapter 15 analyses, AOO analyses and ATWS overpressure analysis.

The COTRANSA2 SER restrictions are similar to those for XCOBRA-T, and a similar evaluation applies. AREVA has provided void data for up to 100 percent void and a sensitivity analysis showing little sensitivity to void bias. Other reactors in the fleet bound the EPFOD operating conditions in BFN. Further, none of the limitations in the original COTRANSA2 SER are violated by BFN operation in the EPFOD.

Therefore, the NRC staff concludes that the use of COTRANSA2 in the BFN EPFOD is an acceptable extension of the existing approval.

STAIF

STAIF is the frequency domain BWR thermal-hydraulic stability code, including reactivity feedback effects. The SER (Reference 104) 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 TH regions must have its own axial power shape to account for 3-D power distributions. For example, high power channels are likely to have more bottom peaked shapes.
4. The collapsed 1-D cross sections must represent the actual conditions being analyzed as closely as possible, including control rod positions.
5. The STAIF calculation must use the "shifted Nyquist" or complex pole search feature to minimize the error at low 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 BFN EPFOD 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 SER for the RODEX2 topical report XN-NF-81-58(P)(A) (Reference 76) has the following limitations, which are implemented by AREVA as engineering guidelines and computer code controls:

1. The NRC concluded that the RODEX2 fission gas release model was acceptable to burnups up to 60 MWd/KgU. This implies a burnup limit of 60 MWd/KgU (nodal basis). (This restriction no longer applies. The exposure limits for BWR fuel were increased to 54 MWd/kgU for an assembly and to 62 MWd/kgU for a rod.)
2. The creep correlation accepted by the NRC is the one with the designation MTYPE = 0.

XN-NF-85-92(P)(A) (Reference 105) justifies Gd fuel properties for up to 8 wt. percent Gd with RODEX2 methods, which covers the expected operation of BFN in MELLLA+ EPFOD.

The NRC staff finds operation in the EPFOD increases the core-average void fraction, but does not change the operating power; thus, fuel rod temperatures and conditions are similar. Therefore, operation in EPFOD is not expected to impact the validity of the RODEX2 models.

Therefore, the NRC staff concludes that the use of RODEX2 in the BFN EPFOD 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 UO₂ fuel pellet with a maximum Gd content of 10.0 weight percent.
3. Cold-worked stress-relieved (CWSR) Zircaloy (Zr)-2 fuel clad material

The SER for the topical report (BAW-10247PA) (Reference 26) has the following limitations:

1. Due to limitations within the fission gas release 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 (Reference 26) (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 EPFOD in BFN without changes.

In the approved RODEX4 Supplement 1 (BAW-10247, Revision 0, Supplement 1PA, Revision 0), AREVA has a corrosion model where the uniform oxidation rate is a two-stage model that is a function of both exposures, a corrosion enhancement factor (depends on reactor chemistry) and temperature at the metal-oxide interface, and therefore, LHGR. It is recognized that both nodular corrosion and diffusion-controlled uniform corrosion occur on BWR cladding. Nodular corrosion is treated as a thermal, 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 CWSR or recrystallized Zr-2 cladding and may be used for analyses where hydrogen content is required.

The NRC staff finds that, as with the RODEX2 evaluation above, operation in the EPFOD increases the core-average void fraction, but does not change the operating power; thus, fuel rod temperatures and conditions are similar. Therefore, operation in EPFOD 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 BFN EPFOD is an acceptable extension of the existing approval.

RELAX

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 SER for the topical report EMF-2361(P)(A) (Reference 68) 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 operation in the EPFOD. Thus, the NRC staff concludes that the use of RELAX in the BFN EPFOD is an acceptable extension of the existing approval.

HUXY

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

The approving SER in XN-CC-33A (Reference 69) 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-°F) 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. ENC 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 SER limitations and operation in the EPFOD does not affect them. Thus, the NRC staff concludes that the use of HUXY in the BFN EPFOD is an acceptable extension of the existing approval.

APPENDIX F

Acronyms and Initialisms

Term	Definition
1RPT	one reactor recirculation pump trip
2RPT	two reactor recirculation pump trip
ABSP	automated backup stability protection
AC	alternating current
AEC	Atomic Energy Commission
ALARA	as low as reasonably achievable
AMSAR	AREVA MELLLA+ Safety Analysis Report
ANS	American Nuclear Society
ANSI	American National Standards Institute
AOI	abnormal operating instruction
AOO	anticipated operational occurrence
APRM	average power range monitor
ART	adjusted reference temperature
ASME	American Society of Mechanical Engineers
AST	alternate source term
ATWS	anticipated transient without scram
ATWS-I	anticipated transient without scram - instability
B Case	Bounding Case
BLEU	blended low enriched uranium
BOC	beginning of cycle
BOP	balance of plant
BFN	Browns Ferry Nuclear Plant, Units 1, 2, and 3
BFN M+SAR	BFN MELLLA+ Safety Analysis Report
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
CGU	commercial grade uranium
CHF	critical heat flux
CLTP	current licensed thermal power

Term	Definition
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
CST	condensate storage tank
Cu	copper
CWSR	cold-worked stress-relieved
DB	design-basis
DBA	design-basis accident
DC	Direct Current
DSS-CD	detect and suppress solution - confirmation density
DSS-CD LTR	Detect and Suppress Solution - Confirmation Density Licensing Topical Report
ECCS	emergency core cooling system
EDG	emergency diesel generator
EMA	equivalent margin analysis
ENC	Exxon Nuclear Company, Inc.
EOC	end of cycle
EOI	emergency operating instruction
EOP	emergency operating procedure
EPFOD	extended power/flow operating domain
EPRI	Electrical Power Research Institute
EPU	extended power uprate
EQ	equipment qualification
ERG	emergency response guidelines
FAC	flow accelerated corrosion
FCTF	Fuel Cooling Test Facility
FHA	fuel handling accident
FIV	flow-induced vibration
FSTF	Full Scale Test Facility
FW	feedwater
FWCF	feedwater controller failure
FWHOOS	feedwater heater out of service
FWTR	feedwater temperature reduction
Gd	Gadolinium
GDC	General Design Criterion / Criteria
GE	General Electric

Term	Definition
GEH	GE-Hitachi Nuclear Energy Americas LLC
GHNE	GE-Hitachi Nuclear Energy
GL	Generic Letter
gpm	gallons per minute
HCTL	heat capacity temperature limit
HCVS	hardened containment vent system
HELB	high energy line break
HEP	human error probability
HN+CT	homogeneous nucleation plus contact temperature
HPCI	high-pressure coolant injection
HRA	human reliability analysis
HSBW	hot shutdown boron weight
HVAC	heating, ventilation and air conditioning
HWC	hydrogen water chemistry
Hz	Hertz
I&C	instrumentation and controls
IASCC	irradiated assisted stress corrosion cracking
lbf	pound-force
ICF	increased core flow
IGSCC	intergranular stress corrosion cracking
IORV	inadvertent opening of a relief valve
IRMS	intermediate range monitors
ISI	inservice inspection program
ISP	integrated surveillance program
L&C	limitation and condition
LAR	license amendment request
LCO	limiting condition for operation (local cladding oxidation)
LDR	Load Definition Report
LERF	large early release frequency
LFWH	Loss of Feedwater Heater
LHGR	linear heat generation rate
LOCA	loss-of-coolant accident
LOOP	loss of offsite power
LPCI	low pressure coolant injection
LPRM	local power range monitor
LPZ	low population zone
LRNB	load reject no bypass
LTR	Licensing Topical Report
LTS	long-term stability solution

Term	Definition
M+LTR	NEDC-33006P-A, Revision 3, GEH MELLLA+ Licensing Topical Report
M+SAR	MELLLA+ Safety Analysis Report
M+SER	Safety Evaluation Report for NEDC-33006P-A, Revision 3, GEH MELLLA+ LTR
MAPLHGR	maximum average planar linear heat generation rate
MASR	minimum alternating stress ratio
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, Revision 4, Applicability of GE Methods to Expanded Operating Domains LTR
Methods SER	Safety Evaluation Report for NEDC-33173P-A, Rev 4, Applicability of GE Methods to Expanded Operating Domains LTR
MOC	middle-of-cycle
MOV	motor-operated valve
MS	main steam
MSIV	main steam isolation valve
MSIVC	main steam isolation valve closure
MSL	main steam line
MSLBA	Main Steam Line Break Accident
MSRV	main steam safety relief valve
MWt	megawatts thermal
N-16	nitrogen-16
NEI	Nuclear Energy Institute
Ni	nickel
NMCA	Noble Metal Chemical Application
NPSH	net positive suction head
NPSHA	NPSH available-
NPSHR	NPSH required
NPSHReff	NPSHR effective
NPSHR 3%	NPSHR 3 percent
NRC	U.S. Nuclear Regulatory Commission
NRR	Office of Nuclear Reactor Regulation
NSSS	nuclear steam supply system
OLMCPR	operating limit minimum critical power ratio
OLTP	original licensed thermal power
OPRM	oscillating power range monitor

Term	Definition
P _a	peak containment accident pressure
P-T	pressure-temperature
PBA	period-based algorithm
PBDA	period-based detection algorithm
PBLE01	Plant-Based Load Evaluation Method 1
PBLE02	Plant-Based Load Evaluation Method 2
PCT	peak cladding temperature
PFR	partial flow reduction
PLFR	partial length fuel rod
PRA	probabilistic risk assessment
PRFO	pressure regulator failure open
PRMS	parameter "root-mean-square pressure"
PRNM	power range neutron monitoring
PSIG	pounds per square inch gauge
PULD	plant unique load definition
RAI	Request for Additional Information
RBM	rod block monitor
RCIC	reactor core isolation cooling
RCPB	reactor coolant pressure boundary
RCS	reactor coolant system
RFOL	Renewed Facility Operating License
RG	Regulatory Guide
RHR	residual heat removal
RIPD	reactor internal pressure difference
RPS	Reactor Protection System
RPT	reactor recirculation pump trip
RPV	reactor pressure vessel
RRS	reactor recirculation system
RS-001	Review Standard for Extended Power Uprate
RSAR	Reload Safety Analysis Report
RSD	replacement steam dryer
RSLB	recirculation suction line break
RT _{NDT}	reference temperature for nil ductility transition
RTP	rated thermal power
RVI	reactor vessel internals
RWCU	reactor water cleanup
RWE	rod withdrawal error
RWM	rod worth minimizer
S _{AD}	amplitude discriminator setpoint

Term	Definition
SAFDL	specified acceptable fuel design limit
SAG	severe accident guideline
SAR	safety analysis report
SBO	station blackout
SE	Safety Evaluation
SER	Safety Evaluation Report
SGTS	standby gas treatment system
SLC	standby liquid control
SLCS	standby liquid control system
SLMCPR	safety limit minimum critical power ratio
SLO	single loop operation
SPCB	Siemens Power Corporation B
SPT	suppression pool temperature
SRLR	Supplemental Reload Licensing Report
SR	Surveillance Requirement
SRM	Staff Requirements Memorandum /source range monitor
SRO	Senior Reactor Operator
SRP	Standard Review Plan
SRV	safety relief valve
SRXB	Reactor Systems Branch
SSC	system, structure, and component
SSLB	small steam line break
SSW	sacrificial shield wall
STP	simulated thermal power
T_{CHF}	critical heat flux temperature
TCD	thermal conductivity degradation
TEDE	total effective dose equivalent
TH	thermal hydraulic
TID	Technical Information Document
TIP	traversing in-core probe
TLO	two-loop operation
T-M	thermal-mechanical
T_{min}	time period lower limit/minimum stable film boiling temperature
TS	technical specification
TSV	turbine stop valve
TTNB	turbine trip no bypass
TTWBP	turbine trip with bypass
UFSAR	Updated Final Safety Analysis Report

Term	Definition
USE	upper shelf energy
USI	unresolved safety issue
Zr	zircaloy

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J. Barstow

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SUBJECT: BROWNS FERRY NUCLEAR PLANT, UNITS 1, 2, AND 3 - ISSUANCE OF
AMENDMENT NOS. 310, 333, AND 293 REGARDING MAXIMUM EXTENDED
LOAD LINE LIMIT ANALYSIS PLUS (L-2018-LLA-0048)
DATED DECEMBER 26, 2019

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